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Standard Technical Specifications Combustion Engineering Plants

Bases (Sections 3.4–3.9)

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PREFACE

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

(continued)

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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 ensure that these parameters will not be less conservative than were assumed in the analyses and thereby provide assurance that the minimum departure from nucleate boiling ratio (DNBR) will meet the required criteria for each of the transients analyzed.

The LCO 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 flow rates 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 $\geq$ [1.3]. This is the acceptance limit for the RCS DNB parameters. Changes to the facility that could impact these parameters must be assessed for their impact on the DNBR criterion. The transients analyzed for include loss of coolant flow events and dropped or struck control element assembly (CEA) events. A key assumption for the analysis of these events is that the core power

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

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] psig, core inlet temperature [500-580]°F, and reactor vessel inlet coolant flow rate [95-116]%.

The RCS DNB limits satisfy Criterion 2 of the NRC Policy Statement.

LCO

This LCO specifies limits on the monitored process variables—RCS 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 the DNBR criterion in the event of a DNB limited transient.

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

APPLICABILITY

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

A Note has been added to indicate the limit on pressurizer pressure may be exceeded during short term operational transients such as a THERMAL POWER ramp increase of > 5% RTP per minute or a THERMAL POWER step increase of > 10% RTP. These conditions represent short term perturbations where actions to control pressure variations might be

(continued)

BASES

APPLICABILITY (continued)

counterproductive. Also, since they represent transients initiated from power levels < 100% RTP, an increased DNBR margin exists to offset the temporary pressure variations.

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 this LCO, but violation of SLs merits a stricter, more severe Required Action. Should a violation of this LCO 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 by Required Action B.1, to restore DNB margin and eliminate the potential for violation of the accident analysis bounds.

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

B.1

If Required Action A.1 is not met within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 2 within 6 hours. In MODE 2, the reduced power condition eliminates the potential for violation of the accident analysis bounds.

(continued)

BASES

ACTIONS

B.1 (continued)

Six 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. If this parameter is 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.

D.1

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

The 6 hour Completion Time is a reasonable time that permits power reduction at an orderly rate in conjunction with even control of SG heat removal.

SURVEILLANCE REQUIREMENTS

SR 3.4.1.1

Since Required Action A.1 allows a Completion Time of 2 hours to restore parameters that are not within limits, the 12 hour Surveillance Frequency for 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.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.1.2

Since Required Action A.1 allows a Completion Time of 2 hours to restore parameters that are not within limits, the 12 hour Surveillance Frequency for 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.

SR 3.4.1.3

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

This SR 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 rate at normal operating conditions at power with all RCPs running.

SR 3.4.1.4

Measurement of RCS total flow rate by performance of a precision calorimetric heat balance once every [18] months. This allows the installed RCS flow instrumentation to be calibrated and verifies that the actual RCS flow rate is within the bounds of the analyses.

The Frequency of [18] months reflects the importance of verifying flow after a refueling outage where the core has been altered, which may have caused an alteration of flow resistance.

The SR is modified by a Note that states the SR is only required to be performed [24] hours after ≥ [90]% RTP. The Note is necessary to allow measurement of the flow rate at normal operating conditions at power in MODE 1. The

(continued)

BASES

SURVEILLANCE REQUIREMENTS	SR 3.4.1.4 (continued) Surveillance cannot be performed in MODE 2 or below, and will not yield accurate results if performed below 90% RTP.
REFERENCES	1. FSAR, Section [15].

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.2 RCS Minimum Temperature for Criticality

BASES

BACKGROUND

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

a. Operation within the existing instrumentation ranges and accuracies;

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 RCS loop 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

There are no accident analyses that dictate the minimum temperature for criticality, but all low power safety analyses assume initial temperatures near the [520]°F limit (Ref. 1).

The RCS minimum temperature for criticality satisfies Criterion 2 of the NRC Policy Statement.

LCO

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

(continued)

BASES

LCO (continued)

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 or below a T_{avg} of [535]°F. The no load temperature of 544°F is maintained by the Steam Dump Control System.

ACTIONS

A.1

If T_{avg} is below [520]°F, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 30 minutes. Rapid reactor shutdown can be readily and practically achieved within a 30 minute period. The allowed time reflects the ability to perform this action and to maintain the plant within the analyzed range.

SURVEILLANCE REQUIREMENTS

SR 3.4.2.1

T_{avg} is required to be verified $\geq$ [520]°F every 30 minutes. The 30 minute time period is frequent enough to prevent inadvertent violation of the LCO. While the Surveillance is required whenever the reactor is critical and temperature is below [535]°F, in practice the Surveillance is most appropriate during the period when the reactor is brought critical.

REFERENCES

1. FSAR, Section [15].

B 3.4 REACTOR COOLANT SYSTEM (RCS)

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

BASES

BACKGROUND

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

The PTLR contains P/T limit curves for heatup, cooldown, and inservice leak and hydrostatic (ISLH) testing, and data for the maximum rate of change of reactor coolant temperature (Ref. 1).

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

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

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

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. 4) and Appendix H of 10 CFR 50 (Ref. 5). The operating P/T limit curves will be adjusted,

(continued)

BASES

BACKGROUND (continued)

as necessary, based on the evaluation findings and the recommendations of Reference 3.

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

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

The criticality limit includes the Reference 2 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 consequence of violating the LCO limits is that the RCS has been operated under conditions that can result in brittle failure of the RCPB, possibly leading to a nonisolable leak or loss of coolant accident. In the event these limits are exceeded, an evaluation must be performed to determine the effect on the structural integrity of the RCPB components. The ASME Code, Section XI, Appendix E (Ref. 6), provides a recommended methodology for evaluating an operating event that causes an excursion outside the limits.

APPLICABLE SAFETY ANALYSES

The P/T limits are not derived from Design Basis Accident (DBA) Analyses. They are prescribed during normal operation to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and cause nonductile failure of the RCPB, an unanalyzed condition. Reference 1 establishes the

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

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

LCO

The two elements of this LCO are:

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

b. Limits on the rate of change of temperature.

The LCO limits apply to all components of the RCS, except the pressurizer.

These limits define allowable operating regions and permit a large number of operating cycles while providing a wide margin to nonductile failure.

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

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

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

(continued)

BASES

LCO (continued)

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

APPLICABILITY

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

During MODES 1 and 2, other Technical Specifications provide limits for operation that can be more restrictive than or can supplement these P/T limits. LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits"; LCO 3.4.2, "RCS Minimum Temperature for Criticality"; and Safety Limit 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.

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 outside the P/T limits must be corrected so that the RCPB is returned to a condition that has been verified by stress analyses.

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

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

Besides restoring operation to within limits, an evaluation is required to determine if RCS operation can continue. The evaluation must verify the RCPB integrity remains acceptable and must be completed before continuing operation. Several methods may be used, including comparison with pre-analyzed transients in the stress analyses, new analyses, or inspection of the components.

ASME Code, Section XI, Appendix E (Ref. 6), may be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline.

The 72 hour Completion Time is reasonable to accomplish the evaluation. The evaluation for a mild violation is possible within this time, but more severe violations may require special, event specific stress analyses or inspections. A favorable evaluation must be completed before continuing to operate.

Condition A is modified by a Note requiring Required Action A.2 to be completed whenever the Condition is entered. The Note emphasizes the need to perform the evaluation of the effects of the excursion outside the allowable limits. Restoration alone per Required Action A.1 is insufficient because higher than analyzed stresses may have occurred and may have affected the RCPB integrity.

B.1 and B.2

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

Either possibility indicates a need for more careful examination of the event, best accomplished with the RCS at

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

reduced pressure and temperature. With reduced pressure and temperature conditions, the possibility of propagation of undetected flaws is decreased.

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.

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

C.1 and C.2

The actions of this LCO, anytime other than in MODE 1, 2, 3, or 4, 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. Operation outside the P/T limits must be corrected so that the RCPB is returned to a condition that has been verified by stress analyses.

The Completion Time of "immediately" 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 a short period of time in a controlled manner.

Besides restoring operation to within limits, an evaluation is required to determine if RCS operation can continue. The evaluation must verify that 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.

ASME Code, Section XI, Appendix E (Ref. 6), may be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline.

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

The Completion Time of prior to entering MODE 4 forces the evaluation prior to entering a MODE where temperature and pressure can be significantly increased. The evaluation for a mild violation is possible within several days, but more severe violations may require special, event specific stress analyses or inspections.

Condition C is modified by a Note requiring Required Action C.2 to be completed whenever the Condition is entered. The Note emphasizes the need to perform the evaluation of the effects of the excursion outside the allowable limits. Restoration alone per Required Action C.1 is insufficient because higher than analyzed stresses may have occurred and may have affected the RCPB integrity.

SURVEILLANCE REQUIREMENTS

SR 3.4.3.1

Verification that operation is within the PTLR limits is required every 30 minutes when RCS pressure and temperature conditions are undergoing planned changes. This Frequency is considered reasonable in view of the control room indication available to monitor RCS status. Also, since temperature rate of change limits are specified in hourly increments, 30 minutes permits assessment and correction for minor deviations within a reasonable time.

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

This SR is modified by a Note that requires this SR be performed only during RCS system heatup, cooldown, and ISLH testing. No SR is given for criticality operations because LCO 3.4.2 contains a more restrictive requirement.

REFERENCES

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

2. 10 CFR 50, Appendix G.

(continued)

BASES

REFERENCES (continued)

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

4. ASTM E 185-82, July 1982.

5. 10 CFR 50, Appendix H.

6. ASME, Boiler and Pressure Vessel Code, Section XI, Appendix E.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.4 RCS Loops—MODES 1 and 2

BASES

BACKGROUND

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

The secondary functions of the RCS include:

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

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

c. Carrying the soluble neutron poison, boric acid;

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

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

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

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

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

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 that 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).

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 and temperature and Safety Limit (i.e., the departure from nucleate boiling ratio (DNBR) limit), assumes a maximum power level of 107% RTP. 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 that result in a minimum DNBR greater than or equal to the critical heat flux correlation limit.

RCS Loops—MODES 1 and 2 satisfy Criteria 2 and 3 of the NRC Policy Statement.

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 that is OPERABLE in accordance with the Steam Generator Tube Surveillance Program. 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

(continued)

BASES

LCO (continued)

MODE 3 if any SG level is ≤ [25]% as sensed by the RPS. The minimum water level to declare the SG OPERABLE is [25]%.

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.

The decay heat production rate is much lower than the full power heat rate. As such, the forced circulation flow and heat sink requirements are reduced for lower, noncritical MODES as indicated by the LCOs for MODES 3, 4, 5, and 6.

Operation in other MODES is covered by:

LCO 3.4.5, "RCS Loops—MODE 3";
LCO 3.4.6, "RCS Loops—MODE 4";
LCO 3.4.7, "RCS Loops—MODE 5, Loops Filled";
LCO 3.4.8, "RCS Loops—MODE 5, Loops Not Filled";
LCO 3.9.4, "Shutdown Cooling (SDC) and Coolant Circulation—High Water Level" (MODE 6); and
LCO 3.9.5, "Shutdown Cooling (SDC) and Coolant Circulation—Low Water Level" (MODE 6).

ACTIONS

A.1

If the requirements of the LCO are not met, the Required Action is to reduce power and bring the plant to MODE 3. This lowers power level and thus reduces the core heat removal needs and minimizes the possibility of violating DNB limits. 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 conditions in an orderly manner and without challenging safety systems.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.4.1

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

REFERENCES

1. FSAR, Section [ ].

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.5 RCS Loops – MODE 3

BASES

BACKGROUND

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

In MODE 3, reactor coolant pumps (RCPs) are used to provide forced circulation 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 provide redundant paths for decay heat removal. Only one RCP needs to be OPERABLE to declare the associated RCS loop OPERABLE.

Reactor coolant natural circulation is not normally used but 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.

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

RCS Loops – MODE 3 satisfy Criterion 3 of the NRC 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 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, a reduction in boron concentration is prohibited because an even concentration distribution throughout the RCS cannot be ensured. Core outlet temperature is to be maintained at least 10°F below the saturation temperature so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

In MODES 3, 4, and 5, it is sometimes necessary to stop all RCPs or shutdown cooling (SDC) pump forced circulation (e.g., to change operation from one SDC train to the other, to perform surveillance or startup testing, to 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 that is OPERABLE in accordance with the Steam Generator Tube Surveillance Program. An RCP is OPERABLE if it is capable of being powered and is able to provide forced flow if required.

APPLICABILITY

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

Operation in other MODES is covered by:

(continued)

BASES

APPLICABILITY (continued)	LCO 3.4.4, "RCS Loops—MODES 1 and 2"; LCO 3.4.6, "RCS Loops—MODE 4"; LCO 3.4.7, "RCS Loops—MODE 5, Loops Filled"; LCO 3.4.8, "RCS Loops—MODE 5, Loops Not Filled"; LCO 3.9.4, "Shutdown Cooling (SDC) and Coolant Circulation—High Water Level" (MODE 6); and LCO 3.9.5, "Shutdown Cooling (SDC) and Coolant Circulation—Low Water Level" (MODE 6).

ACTIONS

A.1

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

B.1

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

C.1 and C.2

If no RCS loop is in operation, except as provided in Note 1 in the LCO section, all operations involving a reduction of RCS 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 OPERABLE status and operation shall be initiated immediately and continued until one RCS loop is restored to OPERABLE status and operation. The immediate Completion Times reflect the importance of maintaining operation for decay heat removal.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.5.1

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

SR 3.4.5.2

This SR requires verification every 12 hours that the secondary side water level in each SG is ≥ [25]%. 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.

SR 3.4.5.3

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

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.

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

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. The LCO allows the two loops that are required to be OPERABLE to consist of any combination of RCS and SDC System loops. 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 provide redundancy for heat removal.

Note 1 permits all RCPs and SDC pumps to be de-energized ≤ 1 hour per 8 hour period. This means that natural circulation has been established using the SGs. The Note

(continued)

BASES

LCO (continued)

prohibits boron dilution when forced flow is stopped because an even concentration distribution cannot be ensured. Core outlet temperature is to be maintained at least 10°F below saturation temperature so that no vapor bubble may form and possibly cause a natural circulation flow obstruction. The 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 overpressure protection (LTOP) limits) must be observed and forced SDC flow or heat removal via the SGs must be re-established prior to reaching the pressure limit. The circumstances for stopping both RCPs or SDC pumps are to be limited to situations where:

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

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

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 is < [60]%; or

b. Secondary side water temperature in each SG is < [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.

Similarly, for the SDC System, an OPERABLE SDC train is composed of the OPERABLE SDC pump(s) capable of providing

(continued)

BASES

LCO (continued)

forced flow to the SDC heat exchanger(s). RCPs and SDC pumps are OPERABLE if they are capable of being powered and are able to provide flow if required.

APPLICABILITY

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

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops—MODES 1 and 2";
LCO 3.4.5, "RCS Loops—MODE 3";
LCO 3.4.7, "RCS Loops—MODE 5, Loops Filled";
LCO 3.4.8, "RCS Loops—MODE 5, Loops Not Filled";
LCO 3.9.4, "Shutdown Cooling (SDC) and Coolant Circulation—High Water Level" (MODE 6); and
LCO 3.9.5, "Shutdown Cooling (SDC) and Coolant Circulation—Low Water Level" (MODE 6).

ACTIONS

A.1

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

B.1

If only one required SDC train is OPERABLE and in operation, redundancy for heat removal is lost. 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 (≤ 200°F) rather than MODE 4 (200°F to 300°F). The Completion Time of 24 hours is reasonable, based on operating experience, to reach MODE 5

(continued)

BASES

ACTIONS

B.1 (continued)

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 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 reduced in this type of operation. The immediate Completion Times reflect the importance of 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 SR requires verification every 12 hours that one required loop or train is in operation. This ensures forced flow is providing heat removal. Verification includes flow rate, temperature, or pump status monitoring. The 12 hour Frequency has been shown by operating practice to be sufficient to regularly assess RCS loop status. In addition, control room indication and alarms will normally indicate loop status.

SR 3.4.6.2

This SR requires verification every 12 hours of secondary side water level in the required SG(s) $\geq$ [25]%. 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.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.6.3

Verification that the required pump is OPERABLE ensures that an additional RCS loop or SDC 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

None.

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 the removal of decay heat and transfer this heat either to the steam generator (SG) secondary side coolant or the component cooling water via the shutdown cooling (SDC) heat exchangers. While the principal means for decay heat removal 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 side water. As long as the SG secondary side 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 decay 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 decay heat removal.

The LCO provides for redundant paths of decay heat removal capability. The first path can be an SDC train that must be OPERABLE and in operation. The second path can be another OPERABLE SDC train, or through the SGs, 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)

BASES

APPLICABLE SAFETY ANALYSES (continued)

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

LCO

The purpose of this LCO is to require at least one of the SDC trains be OPERABLE and in operation with an additional SDC train OPERABLE or secondary side water level of each SG shall be ≥ [25]%. 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 to provide redundant paths for decay heat removal. However, if the standby SDC train is not OPERABLE, a sufficient alternate method to provide redundant paths for decay heat removal is two SGs with their secondary side water levels ≥ [25%]. Should the operating SDC train fail, the SGs could be used to remove the decay heat.

Note 1 permits all SDC pumps to be de-energized ≤ 1 hour per 8 hour period. The circumstances for stopping both SDC trains are to be limited to situations where pressure and temperature increases can be maintained well within the allowable pressure (pressure and temperature and low temperature overpressure protection) and 10°F subcooling limits, or an alternate heat removal path through the SG(s) is in operation.

This LCO is modified by a Note that prohibits boron dilution when SDC forced flow is stopped because an even concentration distribution cannot be ensured. Core outlet temperature is to be maintained at least 10°F below saturation temperature, so that no vapor bubble would form and possibly cause a natural circulation flow obstruction. In this MODE, the SG(s) can be used as the 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 decay heat removal,

(continued)

BASES

LCO (continued)

the reactor coolant temperature can be maintained subcooled, and boron stratification affecting reactivity control is not expected.

Note 2 allows one SDC train to be inoperable for a period of up to 2 hours provided that the other SDC 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.

Note 3 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 < [60]%; or

b. Secondary side 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 RCS circulation function provided by the SDC trains.

An OPERABLE SDC train is composed of an OPERABLE SDC pump and an OPERABLE SDC heat exchanger.

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.

APPLICABILITY

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

(continued)

BASES

APPLICABILITY (continued)

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops—MODES 1 and 2";
LCO 3.4.5, "RCS Loops—MODE 3";
LCO 3.4.6, "RCS Loops—MODE 4";
LCO 3.4.8, "RCS Loops—MODE 5, Loops Not Filled";
LCO 3.9.4, "Shutdown Cooling (SDC) and Coolant Circulation—High Water Level" (MODE 6); and
LCO 3.9.5, "Shutdown Cooling (SDC) and Coolant Circulation—Low Water Level" (MODE 6).

ACTIONS

A.1 and A.2

If the required SDC train is inoperable and any SGs have secondary side water levels < [25%], redundancy for heat removal is lost. Action must be initiated immediately to restore a second SDC train to OPERABLE status or to restore the water level in the required SGs. Either Required Action A.1 or Required Action A.2 will restore redundant decay heat removal paths. The immediate Completion Times reflect the importance of maintaining the availability of two paths for decay heat removal.

B.1 and B.2

If no SDC train is in operation, except as permitted in Note 1, all operations involving the reduction of RCS boron concentration must be suspended. Action to restore one 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 Times reflect the importance of maintaining operation for decay heat removal.

SURVEILLANCE REQUIREMENTS

SR 3.4.7.1

This SR requires verification every 12 hours that one SDC train is in operation. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing decay heat removal. The

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.7.1 (continued)

12 hour Frequency has been shown by operating practice to be sufficient to regularly assess degradation and verify operation is within safety analyses assumptions. In addition, control room indication and alarms will normally indicate loop status.

The SDC flow is established to ensure that core outlet temperature is maintained sufficiently below saturation to allow time for swapover to the standby SDC train should the operating train be lost.

SR 3.4.7.2

Verifying the SGs are OPERABLE by ensuring their secondary side water levels are ≥ [25%] ensures that redundant heat removal paths are available if the second SDC train is inoperable. The Surveillance is required to be performed when the LCO requirement is being met by use of the SGs. If both SDC trains are OPERABLE, this SR is not needed. The 12 hour Frequency has been shown by operating practice to be sufficient to regularly assess degradation and verify operation within safety analyses assumptions.

SR 3.4.7.3

Verification that the second SDC train is OPERABLE ensures that redundant paths for decay heat removal are available. 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 Surveillance is required to be performed when the LCO requirement is being met by one of two SDC trains, e.g., both SGs have < [25]% water level. 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 3.4.8 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. The secondary function of the reactor coolant is to act as a carrier for the soluble neutron poison, boric acid.

In MODE 5 with loops not filled, only 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 and to require that two paths be available to provide redundancy for heat removal.

APPLICABLE SAFETY ANALYSES

In MODE 5, RCS circulation is considered in determining 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 decay heat removal and for boron mixing.

RCS loops—MODE 5 (loops not filled) have been identified in the NRC Policy Statement as important contributors to risk reduction.

LCO

The purpose of this LCO is to require a minimum of two SDC trains be OPERABLE and one of these trains be in operation. An OPERABLE train is one that is capable 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.

(continued)

BASES

LCO (continued)

Note 1 permits the SDC pumps to be de-energized for ≤ 15 minutes when switching from one train to another. The circumstances for stopping both SDC pumps are to be limited to situations when the outage time is short [and the core outlet temperature is maintained > 10°F below saturation temperature]. The Note prohibits boron dilution or draining operations when SDC forced flow is stopped.

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 capable of providing forced flow to an OPERABLE SDC heat exchanger, along with the appropriate flow and temperature instrumentation for control, protection, and indication. SDC pumps are OPERABLE if they are capable of being powered and are able to provide flow if required.

APPLICABILITY

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

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops—MODES 1 and 2";
LCO 3.4.5, "RCS Loops—MODE 3";
LCO 3.4.6, "RCS Loops—MODE 4";
LCO 3.4.7, "RCS Loops—MODE 5, Loops Filled";
LCO 3.9.4, "Shutdown Cooling (SDC) and Coolant Circulation—High Water Level" (MODE 6); and
LCO 3.9.5, "Shutdown Cooling (SDC) and Coolant Circulation—Low Water Level" (MODE 6).

ACTIONS

A.1

If the required SDC train is inoperable, redundancy for heat removal is lost. Action must be initiated immediately to restore a second train to OPERABLE status. The Completion Time reflects the importance of maintaining the availability of two paths for heat removal.

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

If no SDC train is OPERABLE or in operation, except as provided in Note 1, all operations involving the reduction of RCS boron concentration must be suspended. Action to restore one SDC train to OPERABLE status and operation must be initiated immediately. 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.

SURVEILLANCE REQUIREMENTS

SR 3.4.8.1

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

SR 3.4.8.2

Verification that the required number of trains are OPERABLE ensures that redundant paths for heat removal are available and 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 indicated 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

None.

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 level, the required heaters and their backup heater controls, and emergency power supplies. Pressurizer safety valves and pressurizer power operated relief valves (PORVs) are addressed by LCO 3.4.10, "Pressurizer Safety Valves," and LCO 3.4.11, "Pressurizer Power Operated Relief 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 that could result in degraded ability for pressure control.

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

(continued)

BASES

BACKGROUND (continued)

(PORVs or pressurizer 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 Safety Limit of 2750 psig.

The requirement to have two groups of pressurizer heaters ensures 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 loss of single phase flow and decreased capability to remove core decay heat.

APPLICABLE SAFETY ANALYSES

In MODES 1, 2, and 3, the LCO requirement for a steam bubble is reflected implicitly in the accident analyses. No safety analyses are performed in lower MODES. All analyses performed from a critical reactor condition assume the existence of a steam bubble and saturated conditions in the pressurizer. In making this assumption, the analyses neglect the small fraction of noncondensable gases normally present.

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

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 their inclusion. The requirement for emergency power supplies is based on NUREG-0737 (Ref. 1). The intent is to keep 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 a coincident occurrence assumed in the accident analyses, maintaining hot, high pressure conditions over an extended time period is not evaluated in the accident analyses.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The pressurizer satisfies Criterion 2 and Criterion 3 of the NRC Policy Statement.

LCO

The LCO requirement for the pressurizer to be OPERABLE with water level < [60]% ensures that a steam bubble exists. Limiting the maximum operating water level preserves the steam space for pressure control. The LCO has been established to 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 [and capable of being powered from an emergency power supply]. 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 amount needed to maintain pressure is dependent on the ambient heat losses.

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. The LCO does not apply to MODE 5 (Loops Filled) because LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," applies. The LCO does not apply to MODES 5 and 6 with partial loop operation.

In MODES 1, 2, and 3, there is the need to maintain the availability of pressurizer heaters capable of being powered from an emergency power supply. In the event of a loss of offsite power, the initial conditions of these MODES gives

(continued)

BASES

APPLICABILITY (continued)

the greatest demand for maintaining the RCS in a hot pressurized condition with loop subcooling for an extended period. For MODE 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 in service and therefore the LCO is not applicable.

ACTIONS

A.1 and A.2

With pressurizer water level not within the limit, action must be taken to restore the plant to operation within the bounds of the safety analyses. To achieve this status, the unit must be brought to MODE 3, with the reactor trip breakers open, within 6 hours and to MODE 4 within [12] hours. This takes the plant 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 brings the plant to a MODE where the LCO is not applicable. The 12 hour time to reach the nonapplicable MODE is reasonable based on operating experience for that evolution.

B.1

If one required group of pressurizer heaters is inoperable, restoration is required within 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.

C.1 and C.2

If one required group of pressurizer heaters is inoperable and cannot be restored within the allowed Completion Time of Required Action B.1, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and to MODE 4

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

within [12] 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 the indicated level. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess the level for any deviation and verify that operation is within safety analyses assumptions. Alarms are also available for early detection of abnormal level indications.

SR 3.4.9.2

The 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 acceptable.

[SR 3.4.9.3

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

This Surveillance demonstrates that the heaters can be manually transferred to and energized by emergency power supplies. The Frequency of [18] months is based on a

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.9.3 (continued)

[typical fuel cycle and industry accepted practice. This is consistent with similar verifications of emergency power.]

REFERENCES

1. NUREG-0737, November 1980.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.10 Pressurizer Safety Valves

BASES

BACKGROUND

The purpose of the two spring loaded pressurizer safety valves is to provide RCS overpressure protection. Operating in conjunction with the Reactor Protection System, two valves are used to ensure that the Safety Limit (SL) 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 ± 1%. The safety valves discharge steam from the pressurizer to a quench tank located in the containment. The discharge flow is indicated by an increase in temperature downstream of the safety valves and by an increase in the quench tank temperature and level.

The upper and lower pressure limits are based on the ± 1%-tolerance requirement (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.

(continued)

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 < 15% power. Single failure of a safety valve is neither assumed in the accident analysis nor required to be addressed by the ASME Code. Compliance with this specification is required to ensure that the accident analysis and design basis calculations remain valid.

The pressurizer safety valves satisfy Criterion 3 of the NRC Policy Statement.

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 SL, to maintain accident analysis assumptions, and to comply with ASME Code requirements. The upper and lower pressure tolerance limits are based on the ± 1% tolerance requirements (Ref. 1) for lifting pressures above 1000 psig. The limit protected by this specification is the reactor coolant pressure boundary (RCPB) SL of 110% of design pressure. Inoperability of one or both valves could result in exceeding the SL if a transient were to occur. The consequences of exceeding the ASME pressure limit could include damage to one or more RCS components, increased leakage, or additional stress analysis being required prior to resumption of reactor operation.

APPLICABILITY

In MODES 1, 2, and 3, and portions of MODE 4 above the LTOP 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

(continued)

BASES

APPLICABILITY (continued)

included, although the listed accidents may not require both safety valves for protection.

The LCO is not applicable in MODE 4 when all RCS cold leg temperatures are ≤ [285]°F and MODE 5 because LTOP protection is provided. Overpressure protection is not required in MODE 6 with the reactor vessel head detensioned.

The Note allows entry into MODES 3 and 4 with the lift settings outside the LCO limits. This permits testing and examination of the safety valves at high pressure and temperature near their normal operating range, but only after the valves have had a preliminary cold setting. The cold setting gives assurance that the valves are OPERABLE near their design condition. Only one valve at a time will be removed from service for testing. The [36] hour exception is based on 18 hour outage time for each of the two valves. The 18 hour period is derived from operating experience that hot testing can be performed within this timeframe.

ACTIONS

A.1

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

B.1 and B.2

If the Required Action cannot be met within the required Completion Time or if two or more pressurizer safety valves are inoperable, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 at or below [285]°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. At or below

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

[285]°F, overpressure protection is provided by LTOP. The change from MODE 1, 2, or 3 to MODE 4 reduces the RCS energy (core power and pressure), lowers the potential for large pressurizer insurges, and thereby removes the need for overpressure protection by [two] pressurizer safety valves.

SURVEILLANCE REQUIREMENTS

SR 3.4.10.1

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

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

REFERENCES

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

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.11 Pressurizer Power Operated Relief Valves (PORVs)

BASES

BACKGROUND

The pressurizer is equipped with two types of devices for pressure relief: pressurizer safety valves and PORVs. The PORV is an air 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 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 and its block valve controls are powered from normal power supplies. Their controls are also capable of being powered from emergency supplies. Power supplies for the PORV are separate from those for the block valve. Power supply requirements are defined in NUREG-0737, Paragraph III, G.1 (Ref. 1).

The PORV setpoint is above the high pressure reactor trip setpoint and below the opening setpoint for the pressurizer safety valves as required by Reference 2. The purpose of the relationship of these setpoints is to limit the number of transient pressure increase challenges that might open the PORV, which, if opened, could fail in the open position. The PORV setpoint thus limits the frequency of challenges from transients and limits the possibility of a small break LOCA from a failed open PORV. Placing the setpoint below the pressurizer safety valve opening setpoint reduces the frequency of challenges to the safety valves, which, unlike the PORV, cannot be isolated if they were to fail to open.

The primary purpose of this LCO is to ensure that the PORV, its setpoint, and the block valve are operating correctly so

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BASES

BACKGROUND (continued)

the potential for a small break LOCA through the PORV pathway is minimized, or if a small break LOCA were to occur through a failed open PORV, the block valve could be manually operated to isolate the path.

The PORV may be manually operated to depressurize the RCS as deemed necessary by the operator in response to normal or abnormal transients. The PORV may be used for depressurization when the pressurizer spray is not available, a condition that 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 (SGTR) 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, "Low Temperature Overpressure Protection (LTOP) System," addresses this function.

APPLICABLE SAFETY ANALYSES

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

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

The PORV opening setpoint has been established in accordance with Reference 2. It has been set so expected RCS pressure

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

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.

Pressurizer PORVs satisfy Criterion 3 of the NRC Policy Statement.

LCO

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

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.

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 that cause the PORV to open. Imbalances in the energy output of the core and heat removal by the secondary system can cause the RCS pressure to increase to the PORV opening setpoint. Pressure increase transients can occur any time the steam generators are used for heat removal. The most rapid increases will occur at higher operating power and pressure conditions of MODES 1 and 2.

Pressure increases are less prominent in MODE 3 because the core input energy is reduced, but the RCS pressure is high. Therefore, 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 surges become much less significant. The PORV setpoint is reduced for LTOP in

(continued)

BASES

APPLICABILITY (continued)

MODES 4, 5, and 6 with the reactor vessel head in place. LCO 3.4.12 addresses the PORV requirements in these MODES.

ACTIONS

A.1

The ACTIONS are modified by two Notes. Note 1 clarifies that all pressurizer PORVs are treated as separate entities, each with separate Completion Times (i.e., the Completion Time is on a component basis). Note 2 is an exception to LCO 3.0.4. 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.

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 associated block valve, since removal of power would render the block valve inoperable. 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 requires power be maintained to the valve. This Condition is only intended to permit operation of the plant for a limited period of time not to exceed the next refueling outage (MODE 6) so that maintenance can be performed on the PORVs to eliminate the problem condition. The PORVs should normally be available for automatic mitigation of overpressure events and should be returned to OPERABLE status prior to entering startup (MODE 2).

Quick access to the PORV for pressure control can be made when power remains on the closed block valve. The Completion Time of 1 hour is based on plant operating experience that minor problems can be corrected or closure can be accomplished in this time period.

(continued)

BASES

ACTIONS (continued)

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

If one PORV is inoperable and not capable of being manually cycled, it must either be isolated, by closing the associated block valve and removing the power from the block valve, or restored to OPERABLE status. 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.

C.1 and C.2

If one block valve is inoperable, then it must be restored to OPERABLE status, or the associated PORV placed 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 a Completion Time of 72 hours to restore the inoperable block valve to OPERABLE status. The time allowed to restore the block valve is based upon the Completion Time for restoring an inoperable PORV in Condition B since the PORVs are not capable of mitigating an overpressure event when placed in manual control. If the block valve is restored within the Completion Time of 72 hours, the power will be restored and the PORV restored to OPERABLE status.

D.1 and D.2

If the Required Action cannot be met within the associated Completion Time, the plant must be brought to a MODE in which the requirement does not apply. To achieve this

(continued)

BASES

ACTIONS

D.1 and D.2 (continued)

status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

E.1, E.2, E.3, and E.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 two PORVs inoperable. If no PORVs are restored within the Completion Time, then the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 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. Similarly, the Completion Time of 12 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.

F.1 and F.2

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 to OPERABLE status within 2 hours and the remaining block valve in 72 hours. The Completion Time of 1 hour to either restore the block valves or place the associated PORVs in manual

(continued)

BASES

ACTIONS

F.1 and F.2 (continued)

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.

G.1 and G.2

If the Required Actions and associated Completion Times of Condition E or F are not met, then the plant must be brought to a MODE in which the LCO does not apply. The plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 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 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.

SURVEILLANCE REQUIREMENTS

SR 3.4.11.1

Block valve cycling verifies that it can be closed if necessary. The basis for the Frequency of [92 days] is ASME XI (Ref. 3). If the block valve is closed to isolate a PORV that 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 of [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).

The Note modifies this SR by stating that this SR is not required to be performed with the block valve closed in accordance with the Required Actions of this LCO.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.11.2

SR 3.4.11.2 requires complete cycling of each PORV. PORV cycling demonstrates its function. The Frequency of [18] months is based on a typical refueling cycle and industry accepted practice.

[SR 3.4.11.3

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

[SR 3.4.11.4

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, Paragraph III, G.I, November 1980.
2. Inspection and Enforcement (IE) Bulletin 79-05B, April 21, 1979.
3. ASME, Boiler and Pressure Vessel Code, Section XI.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.12 Low Temperature Overpressure Protection (LTOP) System

BASES

BACKGROUND

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

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

The potential for vessel overpressurization is most acute when the RCS is water solid, occurring only while shutdown; a pressure fluctuation can occur more quickly than an operator can react to relieve the condition. Exceeding the RCS P/T limits by a significant amount could cause brittle cracking of the reactor vessel. LCO 3.4.3 requires administrative control of RCS pressure and temperature during heatup and cooldown to prevent exceeding the P/T limits.

This LCO provides RCS overpressure protection by having a minimum coolant input capability and having adequate pressure relief capacity. Limiting coolant input capability requires all but one high pressure safety injection (HPSI) pump and one charging pump incapable of injection into the RCS and isolating the safety injection tanks (SITs). The pressure relief capacity requires either two OPERABLE redundant power operated relief valves (PORVs) or the RCS depressurized and an 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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BASES

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. If conditions require the use of more than one [HPI or] charging pump for makeup in the event of loss of inventory, then pumps can be made available through manual actions.

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 prevent 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 P/T limits will not be exceeded in any analyzed event.

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

(continued)

BASES

BACKGROUND (continued)

RCS Vent Requirements

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

For an RCS vent to meet the specified flow capacity, it requires removing a pressurizer safety valve, removing a PORV's internals, and disabling its block valve in the open position, or similarly establishing a vent by opening an RCS vent valve. The vent path(s) must be above the level of reactor coolant, 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 any RCS cold leg temperature exceeding [285]°F, the pressurizer safety valves prevent RCS pressure from exceeding the Reference 1 limits. At about [285]°F and below, overpressure prevention falls to the OPERABLE PORVs [or to a depressurized RCS and a sufficient sized RCS vent]. Each of these means has a limited overpressure relief capability.

The actual temperature at which the pressure in the P/T limit curve falls below the pressurizer safety valve setpoint increases as the reactor vessel material toughness decreases due to neutron embrittlement. Each time the P/T limit curves are revised, the LTOP System 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.

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BASES

APPLICABLE SAFETY ANALYSES (continued)

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

Mass Input Type Transients

a. Inadvertent safety injection; or

b. Charging/letdown flow mismatch.

Heat Input Type Transients

a. Inadvertent actuation of pressurizer heaters;

b. Loss of shutdown cooling (SDC); 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. Rendering all but one HPSI pump, and all but one charging pump incapable of injection; and

b. Deactivating the SIT discharge isolation valves in their closed positions.

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. Thus, the LCO allows only one HPSI pump and one charging pump OPERABLE during the LTOP MODES. Since neither the PORV nor the RCS vent can handle the pressure transient produced from accumulator injection, when RCS temperature is low, the LCO also requires the SITs isolation when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed in the PTLR.

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

(continued)

APPLICABLE SAFETY ANALYSES

Heat Input Type Transients (continued)

over a narrower RCS temperature range ([175]°F and below) than that of the LCO ([285]°F and below).

Fracture mechanics analyses established the temperature of LTOP Applicability at [285]°F and below. 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 a maximum of 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 one HPSI pump and one charging pump injecting into the RCS. 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 P/T limits are periodically modified as the reactor vessel material toughness decreases due to embrittlement caused by neutron irradiation. Revised P/T limits are determined using neutron fluence projections and the results of examinations of the reactor vessel material irradiation surveillance specimens. The Bases for LCO 3.4.3, "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.

(continued)

BASES

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 pressure 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 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 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 capable of injecting into the RCS and the SITs isolated when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed in the PTLR.

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

a. Two OPERABLE PORVs; or

b. The depressurized RCS and an 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 has proven its ability to open at that setpoint, and motive power is available to the two valves and their control circuits.

(continued)

BASES

LCO (continued)

An RCS vent is OPERABLE when open with an area $\geq$ [1.3] square inches.

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

APPLICABILITY

This LCO is applicable in MODE 4 when the temperature of any RCS cold leg is $\leq$ [285]°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 [285]°F and below. When the reactor vessel head is off, overpressurization cannot occur.

LCO 3.4.3 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 [285]°F.

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

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

ACTIONS

A.1 and B.1

With two or more HPSI pumps capable of injecting into the RCS, overpressurization is possible.

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

(continued)

BASES

ACTIONS

A.1 and B.1 (continued)

Required Action B.1 is modified by a Note that permits two charging pumps capable of RCS injection for ≤ 15 minutes to allow for pump swaps.

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

An unisolated SIT requires isolation within 1 hour. This is only required when the SIT pressure is greater than or equal to the maximum RCS pressure for the existing cold leg temperature allowed in the PTLR.

If isolation is needed and cannot be accomplished within 1 hour, Required Action D.1 and Required Action D.2 provide two options, either of which must be performed within 12 hours. By increasing the RCS temperature to > [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 ≤ [285]°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 and to provide low temperature overpressure mitigation while withstanding a single failure of an active component.

The Completion Time is based on 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.

(continued)

BASES

ACTIONS (continued)

F.1

The consequences of operational events that will overpressure the RCS are more severe at lower temperature (Ref. 6). Thus, one required 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 a reasonable amount of time 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

If two required PORVs are inoperable, or if a Required Action and the associated Completion Time of Condition A, B, D, E, or F are not met, or if 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.

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 incapable of injecting into the RCS. The [HPI] pump[s] and charging pump[s] are rendered incapable

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.12.1, SR 3.4.12.2, and SR 3.4.12.3 (continued)

of injecting into the RCS through removing the power from the pumps by racking the breakers out under administrative control. An alternate method of LTOP control may be employed using at least two independent means to prevent a pump start such that a single failure or single action will not result in an injection into the RCS. This may be accomplished through the pump control switch being placed in [pull to lock] and at least one valve in the discharge flow path being closed.

The 12 hour interval considers operating practice to regularly assess potential degradation and to verify operation within the safety analysis.

SR 3.4.12.4

SR 3.4.12.4 requires verifying that the RCS vent is open ≥ [1.3] square inches is proven OPERABLE by verifying its open condition either:

a. Once every 12 hours for a valve that is unlocked open; or

b. Once every 31 days for a valve that is locked open.

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 vent 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 to be removed, and the manual actuator is not required

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.12.5 (continued)

locked in the inactive position. Thus, the block valve can be closed in the event the PORV develops excessive leakage or does not close (sticks open) after relieving an overpressure 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 every 31 days to verify and, as necessary, adjust the PORV open setpoints. The CHANNEL FUNCTIONAL TEST 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 31 day Frequency considers experience with equipment reliability.

A Note has been added indicating this SR is required to be performed [12] hours after decreasing RCS cold leg temperature to ≤ [285]°F. 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 required 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 with accuracy to known input.

The [18] month Frequency considers operating experience with equipment reliability and matches the typical refueling outage schedule.

(continued)

BASES (continued)

REFERENCES

1. 10 CFR 50, Appendix G.
2. Generic Letter 88-11.
3. FSAR, Section [15].
4. 10 CFR 50.46.
5. 10 CFR 50, Appendix K.
6. Generic Letter 90-06.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.13 RCS Operational LEAKAGE

BASES

BACKGROUND

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

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

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

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

A limited amount of leakage inside containment is expected from auxiliary systems that cannot be made 100% leaktight. Leakage from these systems should be detected, located, and isolated from the containment atmosphere, if possible, to not interfere with RCS LEAKAGE detection.

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

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event. The safety analysis for an event resulting in steam discharge to the atmosphere assumes a 1 gpm primary to secondary LEAKAGE as the initial condition.

Primary to secondary LEAKAGE is a factor in the dose releases outside containment resulting from a steam line break (SLB) accident. To a lesser extent, other accidents or transients involve secondary steam release to the atmosphere, such as a steam generator tube rupture (SGTR). The leakage contaminates the secondary fluid.

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

The SLB is more limiting for site radiation releases. The safety analysis for the SLB accident assumes 1 gpm primary to secondary LEAKAGE in one generator as an initial condition. The dose consequences resulting from the SLB accident are well within the limits defined in 10 CFR 50 or the staff approved licensing basis (i.e., a small fraction of these limits).

RCS operational LEAKAGE satisfies Criterion 2 of the NRC Policy Statement.

LCO

RCS operational LEAKAGE shall be limited to:

a. Pressure Boundary LEAKAGE

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

(continued)

BASES

LCO (continued)

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.

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.

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

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

(continued)

BASES

LCO (continued)

e. Primary to Secondary LEAKAGE through Any One SG

The [720] gallon per day limit on primary to secondary LEAKAGE through any 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.

ACTIONS

A.1

Unidentified LEAKAGE, identified LEAKAGE, or primary to secondary LEAKAGE in excess of the LCO limits must be reduced to within limits within 4 hours. This Completion Time allows time to verify leakage rates and either identify unidentified LEAKAGE or reduce LEAKAGE to within limits before the reactor must be shut down. This action is necessary to prevent further deterioration of the RCPB.

B.1 and B.2

If any pressure boundary LEAKAGE exists or if unidentified, identified, or primary to secondary LEAKAGE cannot be reduced to within limits within 4 hours, the reactor must be brought to lower pressure conditions to reduce the severity of the LEAKAGE and its potential consequences. The reactor must be brought to MODE 3 within 6 hours and to MODE 5 within 36 hours. This action reduces the LEAKAGE and also reduces the factors that tend to degrade the pressure boundary.

The allowed Completion Times are reasonable, based on operating experience, to reach the required conditions from full power conditions in an orderly manner and without challenging plant systems. In MODE 5, the pressure stresses

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

acting on the RCPB are much lower, and further deterioration is much less likely.

SURVEILLANCE REQUIREMENTS

SR 3.4.13.1

Verifying RCS LEAKAGE to be within the LCO limits ensures the integrity of the RCPB is maintained. Pressure boundary LEAKAGE would at first appear as unidentified LEAKAGE and can only be positively identified by inspection. Unidentified LEAKAGE and identified LEAKAGE are determined by performance of an RCS water inventory balance. Primary to secondary LEAKAGE is also measured by performance of an RCS water inventory balance in conjunction with effluent monitoring within the secondary steam and feedwater systems.

The RCS water inventory balance must be performed with the reactor at steady state operating conditions and near operating pressure. Therefore, this SR is not required to be performed in MODES 3 and 4, until 12 hours of steady state operation near operating pressure have elapsed.

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

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

The 72 hour Frequency is a reasonable interval to trend LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents. A Note under the Frequency column states that this SR is required to be performed during steady state operation.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.13.2

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

REFERENCES

1. 10 CFR 50, Appendix A, GDC 30.

2. Regulatory Guide 1.45, May 1973.

3. FSAR, Section [15].

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.14 RCS Pressure Isolation Valve (PIV) Leakage

BASES

BACKGROUND

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

The PIV leakage limit applies to each individual valve. Leakage through both PIVs in series 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 valves in series is determined by a water inventory balance (SR 3.4.13.1). A known component of the identified LEAKAGE before operation begins is the least of the two individual leakage rates determined for leaking series PIVs during the required surveillance testing; leakage measured through one PIV in a line is not RCS operational LEAKAGE if the other is leaktight.

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

The basis for this LCO is the 1975 NRC "Reactor Safety Study" (Ref. 4) that identified potential intersystem LOCAs as a significant contributor to the risk of core melt. A subsequent study (Ref. 5) evaluated various PIV configurations to determine the probability of intersystem LOCAs.

(continued)

BASES

BACKGROUND (continued)

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.

The PIVs are listed in FSAR section (Ref. 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 reactor coolant pressure boundary (RCPB), 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, leakage testing of the valves, and operational changes to determine the effect on the probability of intersystem LOCAs. This study concluded that periodic leakage testing of the PIVs can substantially reduce the probability of an intersystem LOCA.

RCS PIV leakage satisfies Criterion 2 of the NRC Policy Statement.

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

(continued)

BASES

LCO (continued)	significantly suggests that something is operationally wrong and corrective action must be taken. The LCO PIV leakage limit is 0.5 gpm per nominal inch of valve size, with a maximum limit of 5 gpm. The previous criterion of 1 gpm for all valve sizes imposed an unjustified penalty on the larger valves without providing information on potential valve degradation and resulted in higher personnel radiation exposures. A study concluded a leakage rate limit based on valve size was superior to a single allowable value. Reference 7 permits leakage testing at a lower pressure differential than between the specified maximum RCS pressure and the normal pressure of the connected system during RCS operation (the maximum pressure differential) in those types of valves in which the higher service pressure will tend to diminish the overall leakage channel opening. In such cases, the observed rate may be adjusted to the maximum pressure differential by assuming leakage is directly proportional to the pressure differential to the one half power.
APPLICABILITY	In MODES 1, 2, 3, and 4, this LCO applies because the PIV leakage potential is greatest when the RCS is pressurized. In MODE 4, valves in the SDC flow path are not required to meet the requirements of this LCO when in, or during the transition to or from, the SDC mode of operation. In MODES 5 and 6, leakage limits are not provided because the lower reactor coolant pressure results in a reduced potential for leakage and for a LOCA outside the containment.
ACTIONS	The Actions are modified by two Notes. Note 1 is added to provide clarification that each flow path allows separate entry into a Condition. This is allowed based on the functional independence of the flow path. Note 2 requires an evaluation of affected systems if a PIV is inoperable. The leakage may have affected system operability or isolation of a leaking flow path with an alternate valve may

(continued)

BASES

ACTIONS (continued)

have degraded the ability of the interconnected system to perform its safety function.

A.1 and A.2

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

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

[Required Action A.2 specifies that the double isolation barrier of two valves be restored by closing some other valve qualified for isolation or restoring one leaking PIV. The 72 hour Completion Time after exceeding the limit considers the time required to complete the action and the low probability of a second valve failing during this time period.

or

The 72 hour Completion Time after exceeding the limit allows for the restoration of the leaking PIV to OPERABLE status. This timeframe considers the time required to complete this Action and the low probability of a second valve failing during this period. (Reviewer Note: Two options are provided for Required Action A.2. The second option (72 hour restoration) is appropriate if isolation of a second valve would place the unit in an unanalyzed condition.)]

B.1 and B.2

If leakage cannot be reduced [the system isolated] or other Required Actions accomplished, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

and to MODE 5 within 36 hours. This Action reduces the leakage and also reduces the potential for a LOCA outside the containment. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

C.1

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

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1

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

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

Testing 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

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1 (continued)

7 days. The [18] month Frequency is consistent with 10 CFR 50.55a(g) (Ref. 8), as contained in the Inservice Testing Program, is within frequency allowed by the American Society of Mechanical Engineers (ASME) Code, Section XI (Ref. 7), and is based on the need to perform the Surveillance under conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

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

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

Entry into MODES 3 and 4 is allowed to establish the necessary differential pressures and stable conditions to allow for performance of this Surveillance. The Note that allows this provision is complimentary to the Frequency of prior to entry into MODE 2 whenever the unit has been in MODE 5 for 7 days or more, if leakage testing has not been performed in the previous 9 months. In addition, this Surveillance is not required to be performed on the SDC System when the SDC System is aligned to the RCS in the shutdown cooling mode of operation. PIVs contained in the SDC shutdown cooling flow path must be leakage rate tested after SDC is secured and stable unit conditions and the necessary differential pressures are established.

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

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.14.2 and SR 3.4.14.3 (continued)

interlock setpoint that prevents the valves from being opened is set so the actual RCS pressure must be < [425] psig to open the valves. This setpoint ensures the SDC design pressure will not be exceeded and the SDC relief valves will not lift. The 18 month Frequency is based on the need to perform these Surveillances under conditions that apply during a plant outage. The 18 month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment.

The SRs are modified by Notes allowing the SDC autoclosure function to be disabled when using the SDC System suction relief valves for cold overpressure protection in accordance with SR 3.4.12.7.

REFERENCES

1. 10 CFR 50.2.
2. 10 CFR 50.55a(c).
3. 10 CFR 50, Appendix A, Section V, GDC 55.
4. WASH-1400 (NUREG-75/014), Appendix V, October 1975.
5. NUREG-0677, May 1980.
6. [Document containing list of PIVs.]
7. ASME, Boiler and Pressure Vessel Code, Section XI.
8. 10 CFR 50.55a(g).

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.15 RCS Leakage Detection Instrumentation

BASES

BACKGROUND

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

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

Industry practice has shown that water flow changes of 0.5 gpm 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] and the containment air cooler condensate flow rate monitor [are] instrumented to alarm for increases of 0.5 gpm 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

(continued)

BASES

BACKGROUND (continued)

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

Since the humidity level is influenced by several factors, a quantitative evaluation of an indicated leakage rate by this means may be questionable and should be compared to observed increases in liquid flow into or from the containment sump [and condensate flow from air coolers]. Humidity level monitoring is considered most useful as an indirect alarm or indication to alert the operator to a potential problem. Humidity monitors are not required by this LCO.

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

APPLICABLE SAFETY ANALYSES

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

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

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	RCS leakage detection instrumentation satisfies Criterion 1 of the NRC Policy Statement.
LCO	One method of protecting against large RCS LEAKAGE derives from the ability of instruments to rapidly detect extremely small leaks. This LCO requires instruments of diverse monitoring principles to be OPERABLE to provide a high degree of confidence that extremely small leaks are detected in time to allow actions to place the plant in a safe condition when RCS LEAKAGE indicates possible RCPB degradation. The LCO is satisfied when monitors of diverse measurement means are available. Thus, the containment sump monitor, in combination with a particulate or gaseous radioactivity monitor [and a containment air cooler condensate flow rate monitor], provides an acceptable minimum.
APPLICABILITY	Because of elevated RCS temperature and pressure in MODES 1, 2, 3, and 4, RCS leakage detection instrumentation is required to be OPERABLE. In MODE 5 or 6, the temperature is $\leq$ 200°F and pressure is maintained low or at atmospheric pressure. Since the temperatures and pressures are far lower than those for MODES 1, 2, 3, and 4, the likelihood of leakage and crack propagation is much smaller. Therefore, the requirements of this LCO are not applicable in MODES 5 and 6.
ACTIONS	<u>A.1 and A.2</u> If the containment sump monitor is inoperable, no other form of sampling can provide the equivalent information. However, the containment atmosphere radioactivity monitor will provide indications of changes in leakage. Together with the atmosphere monitor, the periodic surveillance for RCS water inventory balance, SR 3.4.13.1, must be performed

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

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 water inventory balance required by Required Action A.1.

Required Action A.1 and Required Action A.2 are modified by a Note that indicates the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when the containment sump monitor channel is inoperable. This allowance is provided because other instrumentation is available to monitor for RCS LEAKAGE.

B.1.1, B.1.2, B.2.1, and B.2.2

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.

Alternatively, continued operation is allowed if the air cooler condensate flow rate monitoring system is OPERABLE, provided grab samples are taken every 24 hours.

The 24 hour interval provides periodic information that is adequate to detect leakage. The 30 day Completion Time recognizes at least one other form of leakage detection is available.

Required Actions B.1.1, B.1.2, B.2.1, and B.2.2 are modified by a Note that indicates that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when the gaseous and particulate containment atmosphere radioactivity monitor channel is inoperable. This allowance

(continued)

BASES

ACTIONS

B.1.1, B.1.2, B.2.1, and B.2.2 (continued)

is provided because other instrumentation is available to monitor for RCS LEAKAGE.

C.1 and C.2

If the required containment air cooler condensate flow rate monitor is inoperable, alternative action is again required. Either SR 3.4.15.1 must be performed, or water inventory balances, in accordance with SR 3.4.13.1, must be performed to provide alternate periodic information. Provided a CHANNEL CHECK is performed every 8 hours or an inventory balance is performed every 24 hours, reactor 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

If the required containment atmosphere radioactivity monitor and the containment air cooler condensate flow rate monitor are 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 Times ensure that the plant will not be operated in a reduced configuration for a lengthy time period.

E.1 and E.2

If any Required Action of Condition A, B, [C], or [D] cannot be met within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full

(continued)

BASES

ACTIONS

E.1 and E.2 (continued)

power conditions in an orderly manner and without challenging plant systems.

F.1

If all required monitors are 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 3.4.15.1 requires the performance of a CHANNEL CHECK of the required containment atmosphere radioactivity monitors. The check gives reasonable confidence the channel is operating properly. The Frequency of [12] hours is based on instrument reliability and is reasonable for detecting off normal conditions.

SR 3.4.15.2

SR 3.4.15.2 requires the performance of a CHANNEL FUNCTIONAL TEST of the required containment atmosphere radioactivity 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 92 days considers instrument reliability, and operating experience has shown it proper for detecting degradation.

SR 3.4.15.3, SR 3.4.15.4, [and SR 3.4.15.5]

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

(continued)

BASES (continued)

REFERENCES

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

BASES

LCO (continued)

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

APPLICABILITY

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

For operation in MODE 3 with RCS average temperature < 500°F, and in MODES 4 and 5, the release of radioactivity in the event of an SGTR is unlikely since the saturation pressure of the reactor coolant is below the lift pressure settings of the atmospheric dump valves and main steam safety valves.

ACTIONS

A Note to the ACTIONS excludes the MODE change restriction of LCO 3.0.4. This exception allows entry into the applicable MODE(S) while relying on the ACTIONS even though the ACTIONS may eventually require plant shutdown. This exception is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is limiting due to exceeding this limit, and the ability to restore transient specific activity excursions while the plant remains at, or proceeds to power operation.

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

With the DOSE EQUIVALENT I-131 greater than the LCO limit, samples at intervals of 4 hours must be taken to demonstrate the limits of Figure 3.4.16-1 are not exceeded. The Completion Time of 4 hours is required to obtain and analyze a sample.

Sampling must continue for trending. The DOSE EQUIVALENT I-131 must be restored to within limits within 48 hours.

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

The Completion Time of 48 hours is required if the limit violation resulted from normal iodine spiking.

B.1

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

C.1 and C.2

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

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

SURVEILLANCE REQUIREMENTS

SR 3.4.16.1

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

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.16.1 (continued)

Trending the results of this Surveillance allows proper remedial action to be taken before reaching the LCO limit under normal operating conditions. The Surveillance is applicable in MODES 1 and 2, and in MODE 3 with RCS average temperature at least 500°F. The 7 day Frequency considers the unlikelihood of a gross fuel failure during 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 hours and 6 hours after a power change of ≥ 15% RTP within a 1 hour period, is established because the iodine levels peak during this time following fuel failure; samples at other times would provide inaccurate results.

SR 3.4.16.3

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.

This SR has been modified by a Note that indicates sampling is required to be performed within 31 days 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.

(continued)

BASES (continued)

REFERENCES

1. 10 CFR 100.11, 1973.
2. FSAR, Section [15.6.3].

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.17 Special Test Exception (STE) - RCS Loops

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 10 CFR 50, Appendix A, 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% 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.

APPLICABLE SAFETY ANALYSES

Special Test Exception (STE) - RCS loops does not satisfy any Criterion in the NRC Policy Statement, but is included as they support other LCOs that meet a Criterion for inclusion.

(continued)

BASES (continued)

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% RTP and the reactor trip setpoints of the OPERABLE power level channels are set ≤ 20% RTP. These limits ensure no Safety Limits or fuel design limits will be violated.

The exception 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 cool down 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.

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.

(continued)

BASES (continued)

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 is 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 startup or 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 startup or 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. 10 CFR 50, Appendix B, Section XI.
2. 10 CFR 50, Appendix A, GDC 1, 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 through the break. The core is essentially in adiabatic heatup. The balance of the 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 the 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.

(continued)

BASES

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 the valves meet the requirements of the Institute of Electrical and Electronic Engineers (IEEE) Standard 279-1971 (Ref. 1) for "operating bypasses" and that the 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 diesel generators (DGs) 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 the blowdown and reflood phases.

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

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.

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;

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

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 inaccuracy, 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.

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

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BASES

APPLICABLE SAFETY ANALYSES (continued)

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 Policy Statement.

LCO

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

(continued)

BASES

LCO (continued)

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, power removed above [2000] psig, and the limits established in the SR for contained volume, boron concentration, and nitrogen cover pressure must be met.

APPLICABILITY

In MODES 1 and 2, and MODE 3 with RCS pressure ≥ 700 psia, the SIT OPERABILITY requirements are based on an assumption of 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.

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

(continued)

BASES

ACTIONS

A.1 (continued)

injection. Thus, 72 hours is allowed to return the boron concentration to within limits.

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 will be taken to return the inoperable accumulator to OPERABLE status. The Completion Time minimizes the exposure of the plant to a LOCA in these conditions.

C.1 and C.2

If the SIT cannot be restored to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and pressurizer pressure reduced to < 700 psia within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

D.1

If more than one SIT is inoperable, the unit is in a condition outside the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.

(continued)

BASES (continued)

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 partially closed. If an isolation valve is not fully open, the rate of injection to the RCS would be reduced. Although a motor operated valve should not change position 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 that ensure the unlikelihood of a mispositioned isolation valve.

SR 3.5.1.2 and SR 3.5.1.3

SIT borated water volume and nitrogen cover pressure should be verified to be within specified limits every 12 hours in order 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 be appropriate for early detection and correction of off normal trends.

SR 3.5.1.4

Thirty-one days is reasonable for verification to determine that each SIT's boron concentration is within the required limits, because the static design of the SITs limits the ways in which the concentration can be changed. The 31 day Frequency is adequate to identify changes that could occur from mechanisms such as stratification or inleakage. Sampling the affected SIT within 6 hours after a 1% volume increase will identify whether inleakage has caused a reduction in boron concentration to below the required limit. It is not necessary to verify boron concentration if the added water is from the RWT, because the water contained in the RWT is within the SIT boron concentration requirements. This is consistent with the recommendations of NUREG-1366 (Ref. 5).

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.1.5

Verification every 31 days that power is removed from each SIT isolation valve operator when the pressurizer pressure is ≥ 2000 psia 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. Since installation and removal of power to the SIT isolation valve operators is conducted under administrative control, the 31 day Frequency was chosen to provide additional assurance that power is removed.

This SR allows power to be supplied to the motor operated isolation valves when RCS pressure is < 2000 psia, thus allowing operational flexibility by avoiding unnecessary delays to manipulate the breakers during unit startups or shutdowns. Even with power supplied to the valves, inadvertent closure is prevented by the RCS pressure interlock associated with the valves. Should closure of a valve occur in spite of the interlock, the SI signal provided to the valves would open a closed valve in the event of a LOCA.

REFERENCES

1. IEEE Standard 279-1971.
2. FSAR, Section [6.3].
3. 10 CFR 50.46.
4. FSAR, Chapter [15].
5. Draft NUREG-1366, February 1990.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.2 ECCS—Operating

BASES

BACKGROUND

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

a. Loss of coolant accident (LOCA);

b. Control Element Assembly (CEA) ejection accident;

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

d. Steam generator tube rupture (SGTR).

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

There are 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 pump divide into four supply lines. Both HPSI trains feed

(continued)

BASES

BACKGROUND (continued)

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 noncondensible 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 < 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 diesel generators (DGs). 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, "Safety Injection Tanks (SITs)," and LCO 3.5.4, "Refueling Water Tank (RWT)," provide the cooling water necessary to meet GDC 35 (Ref. 1).

(continued)

BASES (continued)

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 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.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

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.

ECCS—Operating satisfies Criterion 3 of the NRC 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 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

(continued)

BASES

LCO (continued)

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.

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, "ECCS—Shutdown."

In MODES 5 and 6, unit 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 (SDC) and Coolant Circulation—High Water Level," and LCO 3.9.5, "Shutdown Cooling (SDC) and Coolant Circulation—Low Water Level."

ACTIONS

A.1

If one or more trains are inoperable and at least 100% of the ECCS flow equivalent to a single OPERABLE ECCS train is available, the inoperable components must be returned to

(continued)

BASES

ACTIONS

A.1 (continued)

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 of time to effect many repairs.

An ECCS train 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 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 ECCS 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 emergency DG 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.

Reference 5 describes situations in which one component, such as a shutdown cooling total flow control 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 train cannot be restored to OPERABLE status within the associated Completion Time, the plant must

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and pressurizer pressure reduced to < 1700 psia within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power in an orderly manner and without challenging unit 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 ensuring that a mispositioned valve is an unlikely possibility.

SR 3.5.2.2

Verifying the correct alignment for manual, power operated, and automatic valves in the ECCS flow paths provides assurance that the proper flow paths will exist for ECCS operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves were verified to be in the correct position prior to locking, sealing, or securing. A valve that receives an actuation signal is allowed to be in a nonaccident position provided the valve automatically repositions 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.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.2.2 (continued)

The 31 day Frequency is appropriate because the valves are operated under procedural control and an improper valve position would only affect a single train. This Frequency has been shown to be acceptable through operating experience.

SR 3.5.2.3

With the exception of systems in operation, the ECCS pumps are normally in a standby, nonoperating mode. As such, flow path piping has the potential to develop voids and pockets of entrained gases. Maintaining the piping from the ECCS pumps to the RCS full of water ensures that the system will perform properly, injecting its full capacity into the RCS upon demand. This will also prevent water hammer, pump cavitation, and pumping of noncondensible gas (e.g., air, nitrogen, or hydrogen) into the reactor vessel following an 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

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

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.2.5

Discharge head at design flow is a normal test of charging 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 the required position on an actual or simulated SIAS and on an 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. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 18 month Frequency is based on the need to perform these Surveillances under the conditions that apply during a plant outage and the potential for unplanned transients if the Surveillances were performed with the reactor at power. The 18 month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment. The actuation logic is tested as part of the 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 an 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 units 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

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. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during an outage, on the need to have access to the location, and on the potential for unplanned transients if the Surveillance were performed with the reactor at power. This Frequency is sufficient to detect abnormal degradation and is confirmed by operating experience.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 35.
2. 10 CFR 50.46.
3. FSAR, Chapter [6].
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, 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, "ECCS—Operating," 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.

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

Only one train of ECCS is required for MODE 4. Protection against single failures is not relied on for this MODE of operation.

ECCS—Shutdown satisfies Criterion 3 of the NRC Policy Statement.

LCO

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

(continued)

BASES

LCO (continued)

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 and to deliver its flow to the RCS hot and cold legs.

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."

APPLICABILITY

In MODES 1, 2, and 3 with RCS pressure ≥ 1700 psia, the OPERABILITY requirements for ECCS are covered by LCO 3.5.2.

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, unit 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 (SDC) and Coolant Circulation—High Water Level," and LCO 3.9.5, "Shutdown Cooling (SDC) and Coolant Circulation—Low Water Level."

ACTIONS

A.1

With no HPSI pump OPERABLE, the unit is not prepared to respond to a loss of coolant accident. The 1 hour Completion Time to restore at least one HPSI train to OPERABLE status ensures that prompt action is taken to

(continued)

BASES

ACTIONS

A.1 (continued)

restore the required cooling capacity or to initiate actions to place the unit in MODE 5, where an ECCS train is not required.

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

SURVEILLANCE REQUIREMENTS

SR 3.5.3.1

The applicable Surveillance descriptions from Bases 3.5.2 apply.

REFERENCES

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 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 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

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

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

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 core 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.

The upper limit of [100]°F and the lower limit of [40]°F on RWT temperature are the limits assumed in the accident

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

analysis. Although RWT temperature affects the outcome 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 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 an adequate level exists in the containment sump to support ESF pump operation in the recirculation mode.

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

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 (SDC) and Coolant Circulation—High Water Level," and LCO 3.9.5, "Shutdown Cooling (SDC) and Coolant Circulation—Low Water Level."

ACTIONS

A.1

With RWT boron concentration or borated water temperature not within limits, it must be returned to within limits within 8 hours. In this condition neither the ECCS nor the Containment Spray System can perform their design functions; therefore, prompt action must be taken to restore the tank to OPERABLE condition. The allowed Completion Time of

(continued)

BASES

ACTIONS

A.1 (continued)

8 hours to restore the RWT to within limits was developed considering the time required to change boron concentration or temperature and that the contents of the tank are still available for injection.

B.1

With RWT borated water volume not within limits, it must be returned to within limits within 1 hour. In this condition, neither the ECCS nor Containment Spray System can perform their design functions; therefore, prompt action must be taken to restore the tank to OPERABLE status or to place the unit in a MODE in which these systems are not required. The allowed Completion Time of 1 hour to restore the RWT to OPERABLE status is based on this condition simultaneously affecting multiple redundant trains.

C.1 and C.2

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

SURVEILLANCE REQUIREMENTS

SR 3.5.4.1

RWT borated water temperature shall be verified every 24 hours to be within the limits assumed in the accident analysis. This Frequency has been shown to be sufficient to identify temperature changes that approach either acceptable limit.

The SR is modified by a Note that eliminates the requirement to perform this Surveillance when ambient air temperatures are within the operating temperature limits of the RWT. With

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.4.1 (continued)

ambient temperatures within this range, the RWT temperature should not exceed the limits.

SR 3.5.4.2

Above minimum RWT water volume level shall be verified every 7 days. This Frequency ensures that a sufficient initial water supply is available for injection and to support continued ESF pump operation on recirculation. Since the RWT volume is normally stable and is provided with a Low Level Alarm, a 7 day Frequency is appropriate and has been shown to be acceptable through operating experience.

SR 3.5.4.3

Boron concentration of the RWT shall be verified every 7 days to be within the required range. This Frequency 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 shown through operating experience to be acceptable.

REFERENCES

1. FSAR, Chapter [6] and Chapter [15].

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.5 Trisodium Phosphate (TSP)

BASES

BACKGROUND

Trisodium phosphate (TSP) dodecahydrate is placed on the floor or in the sump of the containment building to ensure that iodine, which may be dissolved in the recirculated reactor cooling water following a loss of coolant accident (LOCA), remains in solution. TSP also helps inhibit stress corrosion cracking (SCC) of austenitic stainless steel components in containment during the recirculation phase following an accident.

Fuel that is damaged during a LOCA will release iodine in several chemical forms to the reactor coolant and to the containment atmosphere. A portion of the iodine in the containment atmosphere is washed to the sump by containment sprays. The emergency core cooling water is borated for reactivity control. This borated water causes the sump solution to be acidic. In a low pH (acidic) solution, dissolved iodine will be converted to a volatile form. The volatile iodine will evolve out of solution into the containment atmosphere, significantly increasing the levels of airborne iodine. The increased levels of airborne iodine in containment contribute to the radiological releases and increase the consequences from the accident due to containment atmosphere leakage.

After a LOCA, the components of the core cooling and containment spray systems will be exposed to high temperature borated water. Prolonged exposure to the core cooling water combined with stresses imposed on the components can cause SCC. The SCC is a function of stress, oxygen and chloride concentrations, pH, temperature, and alloy composition of the components. High temperatures and low pH, which would be present after a LOCA, tend to promote SCC. This can lead to the failure of necessary safety systems or components.

Adjusting the pH of the recirculation solution to levels above 7.0 prevents a significant fraction of the dissolved iodine from converting to a volatile form. The higher pH thus decreases the level of airborne iodine in containment and reduces the radiological consequences from containment atmosphere leakage following a LOCA. Maintaining the

(continued)

BASES

BACKGROUND (continued)

solution pH above 7.0 also reduces the occurrence of SCC of austenitic stainless steel components in containment. Reducing SCC reduces the probability of failure of components.

Granular TSP dodecahydrate is employed as a passive form of pH control for post LOCA containment spray and core cooling water. Baskets of TSP are placed on the floor or in the sump of the containment building to dissolve from released reactor coolant water and containment sprays after a LOCA. Recirculation of the water for core cooling and containment sprays then provides mixing to achieve a uniform solution pH. The dodecahydrate form of TSP is used because of the high humidity in the containment building during normal operation. Since the TSP is hydrated, it is less likely to absorb large amounts of water from the humid atmosphere and will undergo less physical and chemical change than the anhydrous form of TSP.

APPLICABLE SAFETY ANALYSES

The LOCA radiological consequences analysis takes credit for iodine retention in the sump solution based on the recirculation water pH being ≥ 7.0. The radionuclide releases from the containment atmosphere and the consequences of a LOCA would be increased if the pH of the recirculation water were not adjusted to 7.0 or above.

LCO

The TSP is required to adjust the pH of the recirculation water to > 7.0 after a LOCA. A pH > 7.0 is necessary to prevent significant amounts of iodine released from fuel failures and dissolved in the recirculation water from converting to a volatile form and evolving into the containment atmosphere. Higher levels of airborne iodine in containment may increase the release of radionuclides and the consequences of the accident. A pH > 7.0 is also necessary to prevent SCC of austenitic stainless steel components in containment. SCC increases the probability of failure of components.

The required amount of TSP is based upon the extreme cases of water volume and pH possible in the containment sump after a large break LOCA. The minimum required volume is the volume of TSP that will achieve a sump solution pH of

(continued)

BASES

LCO (continued)

≥ 7.0 when taking into consideration the maximum possible sump water volume and the minimum possible pH. The amount of TSP needed in the containment building is based on the mass of TSP required to achieve the desired pH. However, a required volume is specified, rather than mass, since it is not feasible to weigh the entire amount of TSP in containment. The minimum required volume is based on the manufactured density of TSP dodecahydrate. Since TSP can have a tendency to agglomerate from high humidity in the containment building, the density may increase and the volume decrease during normal plant operation. Due to possible agglomeration and increase in density, estimating the minimum volume of TSP in containment is conservative with respect to achieving a minimum required pH.

APPLICABILITY

In MODES 1, 2, and 3, the RCS is at elevated temperature and pressure, providing an energy potential for a LOCA. The potential for a LOCA results in a need for the ability to control the pH of the recirculated coolant.

In MODES 4, 5, and 6, the potential for a LOCA is reduced or nonexistent, and TSP is not required.

ACTIONS

A.1

If it is discovered that the TSP in the containment building sump is not within limits, action must be taken to restore the TSP to within limits. During plant operation the containment sump is not accessible and corrections may not be possible.

The Completion Time of 72 hours is allowed for restoring the TSP within limits, where possible, because 72 hours is the same time allowed for restoration of other ECCS components.

B.1 and B.2

If the TSP cannot be restored within limits within the Completion Time of Required Action A.1, the plant must be brought to a MODE in which the LCO does not apply. The specified Completion Times for reaching MODES 3 and 4 are

(continued)

ACTIONS

B.1 and B.2 (continued)

those used throughout the Technical Specifications; they were chosen to allow reaching the specified conditions from full power in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.5.5.1

Periodic determination of the volume of TSP in containment must be performed due to the possibility of leaking valves and components in the containment building that could cause dissolution of the TSP during normal operation. A Frequency of 18 months is required to determine visually that a minimum of [291] cubic feet is contained in the TSP baskets. This requirement ensures that there is an adequate volume of TSP to adjust the pH of the post LOCA sump solution to a value ≥ 7.0.

The periodic verification is required every 18 months, since access to the TSP baskets is only feasible during outages, and normal fuel cycles are scheduled for 18 months. Operating experience has shown this Surveillance Frequency acceptable due to the margin in the volume of TSP placed in the containment building.

SR 3.5.5.2

Testing must be performed to ensure the solubility and buffering ability of the TSP after exposure to the containment environment. A representative sample of [ ] grams of TSP from one of the baskets in containment is submerged in 1.0 gal ± 0.05 gal of water at a boron concentration of [ ] ppm and at the standard temperature of 25°C ± 5°C. Without agitation, the solution pH should be raised to ≥ 7 within 4 hours. The representative sample weight is based on the minimum required TSP weight of [ ] kilograms, which at manufactured density corresponds to the minimum volume of [ ] cubic ft, and maximum possible post LOCA sump volume of [ ] gallons, normalized to buffer a 1.0 gal sample. The boron concentration of the test water is representative of the maximum possible boron concentration corresponding to the maximum possible post

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.5.2 (continued)

LOCA sump volume. Agitation of the test solution is prohibited, since an adequate standard for the agitation intensity cannot be specified. The test time of 4 hours is necessary to allow time for the dissolved TSP to naturally diffuse through the sample solution. In the post LOCA containment sump, rapid mixing would occur, significantly decreasing the actual amount of time before the required pH is achieved. This would ensure compliance with the Standard Review Plan requirement of a pH $\geq$ 7.0 by the onset of recirculation after a LOCA.

REFERENCES

None.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1 Containment (Atmospheric)

BASES

BACKGROUND

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

The containment is a reinforced concrete structure with a cylindrical wall, a flat foundation mat, and a shallow dome roof. For containments with ungrouted tendons, the cylinder wall is prestressed with a post tensioning system in the vertical and horizontal directions, and the dome roof is prestressed 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 the leakage of fission product radioactivity from the containment to the environment. SR 3.6.1.1 leakage rate requirements comply with 10 CFR 50, Appendix J (Ref. 1), as modified by approved exemptions.

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

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

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

(continued)

BASES

BACKGROUND (continued)

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

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

c. All equipment hatches are closed; and

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

APPLICABLE SAFETY ANALYSES

The safety design basis for the containment is that the containment must withstand the pressures and temperatures of the limiting DBA without exceeding the design leakage rate.

The DBAs that result in a release of radioactive material within containment are a loss of coolant accident, a main steam line break (MSLB), and a control element assembly ejection accident (Ref. 2). In the analysis of each of these accidents, it is assumed that containment is OPERABLE such that release of fission products to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of [0.10]% of containment air weight per day (Ref. 3). This leakage rate is defined in 10 CFR 50, Appendix J (Ref. 1), as L_a: the maximum allowable containment leakage rate at the calculated maximum peak containment pressure (P_a) of [55.7] psig, which results from the limiting DBA, which is a design basis MSLB (Ref. 2).

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

The containment satisfies Criterion 3 of the NRC Policy Statement.

LCO

Containment OPERABILITY is maintained by limiting leakage to $\leq 1.0\ L_a$, except prior to the first startup after performing a required 10 CFR 50, Appendix J, leakage test. At this

(continued)

BASES

LCO (continued)	time, the combined Type B and C leakage must be $< 0.6\ L_a$, and the overall Type A leakage must be $< 0.75\ L_a$. Compliance with this LCO will ensure a containment configuration, including equipment hatches, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analysis. Individual leakage rates specified for the containment air lock (LCO 3.6.2) [and purge valves with resilient seals (LCO 3.6.3)] are not specifically part of the acceptance criteria of 10 CFR 50, Appendix J. Therefore, leakage rates exceeding these individual limits only result in the containment being inoperable when the leakage results in exceeding the acceptance criteria of Appendix J.
APPLICABILITY	In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material into containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, containment is not required to be OPERABLE in MODE 5 to prevent leakage of radioactive material from containment. The requirements for containment during MODE 6 are addressed in LCO 3.9.3, "Containment Penetrations."
ACTIONS	<u>A.1</u> 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 when containment is inoperable is minimal. <u>B.1 and B.2</u> If containment cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a

(continued)

ACTIONS

B.1 and B.2 (continued)

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

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1

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

SR 3.6.1.2

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

(continued)

BASES (continued)

REFERENCES

1. 10 CFR 50, Appendix J.
2. FSAR, Section [ ].
3. FSAR, Section [ ].
4. Regulatory Guide 1.35, Revision [1].

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1 Containment (Dual)

BASES

BACKGROUND

The containment is a free standing steel pressure vessel surrounded by a reinforced concrete shield building. The containment vessel, including all its penetrations, is a low leakage steel shell designed to contain radioactive material that may be released from the reactor core following a Design Basis Accident (DBA). 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, completely enclosed by a reinforced concrete shield building. A 4 ft wide annular space exists between the walls and domes of the steel containment vessel and the concrete shield building to permit inservice inspection and collection of containment outleakage. Dual containments utilize an outer concrete building for shielding and an inner steel containment for leak tightness.

Containment piping penetration assemblies provide for the passage of process, service, sampling, and instrumentation pipelines into the containment vessel while maintaining containment OPERABILITY. The shield building provides biological shielding and allows controlled release of the annulus atmosphere under accident conditions, as well as environmental missile protection for the containment vessel and the Nuclear Steam Supply System.

The inner steel containment and its penetrations establish the leakage limiting boundary of the containment. Maintaining the containment OPERABLE limits the leakage of fission product radioactivity from the containment to the environment. Loss of containment OPERABILITY could cause site boundary doses, in the event of a DBA, to exceed values given in the licensing basis. SR 3.6.1.1 leakage rate requirements comply with 10 CFR 50, Appendix J (Ref. 1), as modified by approved exemptions.

(continued)

BASES

BACKGROUND (continued)

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

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

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

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

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

c. All equipment hatches are closed; and

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

APPLICABLE SAFETY ANALYSES

The safety design basis for the containment is that the containment must withstand the pressures and temperatures of the limiting DBA without exceeding the design leakage rate.

The DBAs that result in a release of radioactive material within containment are a loss of coolant accident, a main steam line break (MSLB), and a control element assembly ejection accident (Ref. 2). In the analysis of each of these accidents, it is assumed that containment is OPERABLE such that release of fission products to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of [0.50]% of containment air weight per day (Ref. 3). This leakage rate is defined in 10 CFR 50, Appendix J (Ref. 1), as L_a: the maximum allowable containment leakage rate at the calculated maximum peak containment pressure (P_a) of [42.3] psig, which results from the limiting DBA, which is a 75% RTP MSLB (Ref. 2).

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

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

The containment satisfies Criterion 3 of the NRC Policy Statement.

LCO

Containment OPERABILITY is maintained by limiting leakage to $\leq 1.0\ L_a$, except prior to the first startup after performing a required 10 CFR 50, Appendix J, leakage test. At this time, the combined Type B and C leakage must be $< 0.6\ L_a$, and the overall Type A leakage must be $< 0.75\ L_a$. Compliance with this LCO will ensure a containment configuration, including equipment hatches, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analysis.

Individual leakage rates specified for the containment air lock (LCO 3.6.2) [, purge valves with resilient seals, and secondary bypass leakage (LCO 3.6.3)] are not specifically part of the acceptance criteria of 10 CFR 50, Appendix J. Therefore, leakage rates exceeding these individual limits only result in the containment being inoperable when the leakage results in exceeding the acceptance criteria of Appendix J.

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

(continued)

BASES

ACTIONS

A.1 (continued)

problem commensurate with the importance of maintaining containment OPERABLE during MODES 1, 2, 3, and 4. This time period also ensures that the probability of an accident (requiring containment OPERABILITY) occurring during periods when containment is inoperable is minimal.

B.1 and B.2

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

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1

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

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1 (continued)

containment leakage rate does not exceed the leakage rate assumed in the safety analysis.

[SR 3.6.1.2

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

REFERENCES

1. 10 CFR 50, Appendix J.
2. FSAR, Section [ ].
3. FSAR, Section [ ].
4. Regulatory Guide 1.35, Revision [1].

B 3.6 CONTAINMENT SYSTEMS

B 3.6.2 Containment Air Locks (Atmospheric and Dual)

BASES

BACKGROUND

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

Each air lock is nominally a right circular cylinder, 10 ft in diameter, with a door at each end. The doors are interlocked to prevent simultaneous opening. During periods when containment is not required to be OPERABLE, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary. Each air lock door has been designed and 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 leak tightness is essential for maintaining the containment leakage rate within limit in the event of a DBA. Not maintaining air lock integrity or leak tightness may result in a leakage rate in excess of that assumed in the unit safety analysis.

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BASES (continued)

APPLICABLE SAFETY ANALYSES

For atmospheric containment, the DBAs that result in a release of radioactive material within containment are a loss of coolant accident (LOCA), a main steam line break (MSLB) and a control element assembly (CEA) ejection accident (Ref. 2). In the analysis of each of these accidents, it is assumed that containment is OPERABLE such that release of fission products to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of [0.10]% of containment air weight per day (Ref. 3). This leakage rate is defined in 10 CFR 50, Appendix J (Ref. 1), as L_a: the maximum allowable containment leakage rate at the calculated maximum peak containment pressure (P_a) of [55.7] psig, which results from the limiting DBA, which is a design basis MSLB (Ref. 2). This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air lock.

For dual containment, the DBAs that result in a release of radioactive material within containment are a LOCA, an MSLB, and a CEA ejection accident (Ref. 2). In the analysis of each of these accidents, it is assumed that containment is OPERABLE such that release of fission products to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of [0.50]% of containment air weight per day (Ref. 3). This leakage rate is defined in 10 CFR 50, Appendix J (Ref. 1), as L_a: the maximum allowable containment leakage rate at the calculated maximum peak containment pressure (P_a) of [42.3] psig, which results from the limiting DBA, which is a 75% RTP MSLB (Ref. 2). This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air lock.

The containment air locks satisfy Criterion 3 of the NRC 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 the containment leakage rate resulting from a DBA. Thus, each air lock's structural integrity and leak tightness are essential to the successful mitigation of such an event.

(continued)

BASES

LCO (continued)	Each air lock is required to be OPERABLE. For the air lock to be considered OPERABLE, the air lock interlock mechanism must be OPERABLE, the air lock must be in compliance with the Type B air lock leakage test, and both air lock doors must be OPERABLE. The interlock allows only one air lock door of an air lock to be opened at one time. This provision ensures that a gross breach of containment does not exist when containment is required to be OPERABLE. Closure of a single door in each air lock is sufficient to provide a leak tight barrier following postulated events. Nevertheless, both doors are kept closed when the air lock is not being used for normal entry into and exit from containment.
APPLICABILITY	In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the containment air locks are not required in MODE 5 to prevent leakage of radioactive material from containment. The requirements for the containment air locks during MODE 6 are addressed in LCO 3.9.3, "Containment Penetrations."
ACTIONS	The ACTIONS are modified by a Note that allows entry and exit to perform repairs on the affected air lock component. If the outer door is inoperable, then it may be easily accessed for most repairs. It is preferred that the air lock be accessed from inside primary containment by entering through the other OPERABLE air lock. However, if this is not practicable, or if repairs on either door must be performed from the barrel side of the door then it is permissible to enter the air lock through the OPERABLE door, which means there is a short time during which the containment boundary is not intact (during access through the OPERABLE door). The ability to open the OPERABLE door, even if it means the containment boundary is temporarily not intact, is acceptable 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

(continued)

BASES

ACTIONS (continued)

immediately closed. If ALARA conditions permit, entry and exit should be via an OPERABLE air lock.

A second Note has been added to provide clarification that, for this LCO, separate Condition entry is allowed for each air lock. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable air lock. Complying with the Required Actions may allow for continued operation, and a subsequent inoperable air lock is governed by subsequent Condition entry and application of associated Required Actions. A third Note has been included that requires entry into the applicable Conditions and Required Actions of LCO 3.6.1, "Containment," when leakage results in exceeding the overall containment leakage limit.

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

With one air lock door inoperable in one or more containment air locks, the OPERABLE door must be verified closed (Required Action A.1) in each affected containment air lock. This ensures that a leak tight containment barrier is maintained by the use of an OPERABLE air lock door. This action must be completed within 1 hour. This specified time period is consistent with the ACTIONS of LCO 3.6.1, which requires containment be restored to OPERABLE status within 1 hour.

In addition, the affected air lock penetration must be isolated by locking closed an OPERABLE air lock door within the 24 hour Completion Time. The 24 hour Completion Time is considered reasonable for locking the OPERABLE air lock door, considering the OPERABLE door of the affected air lock is being maintained closed.

Required Action A.3 verifies that an air lock with an inoperable door has been isolated by the use of a locked and closed OPERABLE air lock door. This ensures that an acceptable containment leakage boundary is maintained. The Completion Time of once per 31 days is based on engineering judgment and is considered adequate in view of the low likelihood of a locked door being mispositioned and other administrative controls. Required Action A.3 is modified by a Note that applies to air lock doors located in high radiation areas and allows these doors to be verified locked

(continued)

BASES

ACTIONS

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

closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of the door, once it has been verified to be in the proper position, is small.

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

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

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

The Required Actions have been modified by two Notes. Note 1 ensures that only the Required Actions and associated Completion Times of Condition C are required if both doors

(continued)

BASES

ACTIONS

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

in the same air lock are inoperable. With both doors in the same air lock inoperable, an OPERABLE door is not available to be closed. Required Actions C.1 and C.2 are the appropriate remedial actions. Note 2 allows entry into and exit from containment under the control of a dedicated individual stationed at the air lock to ensure that only one door is opened at a time (i.e., the individual performs the function of the interlock).

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

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

With one or more air locks inoperable for reasons other than those described in Condition A or B, Required Action C.1 requires action to be initiated immediately to evaluate previous combined leakage rates using current air lock test results. An evaluation is acceptable since it is overly conservative to immediately declare the containment inoperable if both doors in an air lock have failed a seal test or if the overall air lock leakage is not within limits. In many instances (e.g., only one seal per door has failed), containment remains OPERABLE, yet only 1 hour (per LCO 3.6.1) would be provided to restore the air lock door to OPERABLE status prior to requiring a plant shutdown. In addition, even with both doors failing the seal test, the overall containment leakage rate can still be within limits.

Required Action C.2 requires that one door in the affected containment air lock must be verified to be closed. This action must be completed within the 1 hour Completion Time. This specified time period is consistent with the ACTIONS of LCO 3.6.1, which requires that containment be restored to OPERABLE status within 1 hour.

(continued)

BASES

ACTIONS

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

Additionally, the affected air lock(s) must be restored to OPERABLE status within the 24 hour Completion Time. The specified time period is considered reasonable for restoring an inoperable air lock to OPERABLE status, assuming that at least one door is maintained closed in each affected air lock.

D.1 and D.2

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

SURVEILLANCE REQUIREMENTS

SR 3.6.2.1

Maintaining containment air locks OPERABLE requires compliance with the leakage rate test requirements of 10 CFR 50, Appendix J (Ref. 1), as modified by approved exemptions. This SR reflects the leakage rate testing requirements with regard to air lock leakage (Type B leakage tests). The acceptance criteria were established during initial air lock and containment OPERABILITY testing. The periodic testing requirements verify that the air lock leakage does not exceed the allowed fraction of the overall containment leakage rate. The Frequency is required by Appendix J, as modified by approved exemptions. Thus, SR 3.0.2 (which allows Frequency extensions) does not apply.

The SR has been modified by two Notes. Note 1 states that an inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. This is considered reasonable since either air lock door is capable of providing a fission product barrier in the event of a DBA. Note 2 has been added to this SR requiring the results to be evaluated against the acceptance criteria of

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.2.1 (continued)

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

SR 3.6.2.2

The air lock interlock is designed to prevent simultaneous opening of both doors in a single air lock. Since both the inner and outer doors of an air lock are designed to withstand the maximum expected post accident containment pressure, closure of either door will support containment OPERABILITY. Thus, the door interlock feature supports containment OPERABILITY while the air lock is being used for personnel transit into and out of 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 upon entering containment but is not required more frequently than every 184 days. The 184 day Frequency is based on engineering judgment and is considered adequate in view of other indications of door and interlock mechanism status available to operations personnel.

REFERENCES

1. 10 CFR 50, Appendix J.
2. FSAR, Section [ ].
3. FSAR, Section [ ].

B 3.6 CONTAINMENT SYSTEMS

B 3.6.3 Containment Isolation Valves (Atmospheric and Dual)

BASES

BACKGROUND

The containment isolation valves form part of the containment pressure boundary and provide a means for fluid penetrations not serving accident consequence limiting systems to be provided with two isolation barriers that are closed on an automatic isolation signal. These isolation devices are either passive or active (automatic). Manual valves, de-activated automatic valves secured in their closed position (including check valves with flow through the valve secured), blind flanges, and closed systems are considered passive devices. Check valves, or other automatic valves designed to close without operator action following an accident, are considered active devices. Two barriers in series are provided for each penetration so that no single credible failure or malfunction of an active component can result in a loss of isolation or leakage that exceeds limits assumed in the safety analysis. One of these barriers may be a closed system.

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 systems in order to prevent leakage of radioactive material. Upon actuation of safety injection, automatic containment isolation 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).

The OPERABILITY requirements for containment isolation valves help ensure that containment is isolated within the time limits assumed in the safety analysis. Therefore, the OPERABILITY requirements provide assurance that the containment function assumed in the accident analysis will be maintained.

(continued)

BASES

BACKGROUND (continued)

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: normal purge and exhaust valves and minipurge and exhaust valves. The normal and minipurge supply and exhaust lines are each supplied with inside and outside containment isolation valves but share common supply and exhaust penetration lines.

The normal purge valves are designed for purging the containment atmosphere to the unit stack while introducing filtered makeup from the outside to provide adequate ventilation for personnel comfort when the unit 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. The air operated valves fail closed upon a loss of air. Because of their large size, the normal purge valves in some units are not qualified for automatic closure from their open position under DBA conditions. Therefore, the normal purge valves are normally maintained closed in MODES 1, 2, 3, and 4 to ensure the containment boundary is maintained.

Open normal purge valves, or a failure of the minipurge valves to close, following an accident that releases contamination to the containment atmosphere would cause a significant increase in the containment leakage rate.

APPLICABLE SAFETY ANALYSES

The containment isolation valve LCO was derived from the assumptions related to minimizing the loss of reactor coolant inventory and establishing the containment boundary during major accidents. As part of the containment boundary, containment isolation valve OPERABILITY supports leak tightness of the containment. Therefore, the safety analysis of any event requiring isolation of containment is applicable to this LCO.

The DBAs that result in a release of radioactive material within containment are a loss of coolant accident (LOCA), a

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

main steam line break, and 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 paths to the environment through containment isolation valves (including containment purge valves) are minimized. The safety analysis assumes that the normal purge valves are closed at event initiation.

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

The single failure criterion required to be imposed in the conduct of unit safety analyses was considered in the original design of the containment purge valves. Two valves in 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 compromising the containment boundary as long as the system is operated in accordance with the subject LCO. The minipurge 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 satisfy Criterion 3 of the NRC Policy Statement.

(continued)

BASES (continued)

LCO

Containment isolation valves form a part of the containment boundary. The containment isolation valve safety function is related to minimizing the loss of reactor coolant inventory and establishing the containment boundary during a DBA.

The automatic power operated isolation valves are required to have isolation times within limits and to actuate on an automatic isolation signal. The purge valves must be maintained sealed closed [or have blocks installed to prevent full opening]. [Blocked purge valves also actuate on an automatic signal.] The valves covered by this LCO are listed with their associated stroke times in the FSAR (Ref. 1).

The normally closed isolation valves are considered OPERABLE when manual valves are closed, automatic valves are de-activated and secured in their closed position, blind flanges are in place, and closed systems are intact. These passive isolation valves or devices are those listed in Reference 2.

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

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

APPLICABILITY

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

(continued)

BASES (continued)

ACTIONS

The ACTIONS are modified by a Note allowing penetration flow paths, except for [42] inch purge valve penetration flow paths, to be unisolated intermittently under administrative controls. These administrative controls consist of stationing a dedicated operator at the valve controls, who is in continuous communication with the control room. In this way, the penetration can be rapidly isolated when a need for containment isolation is indicated. Due to the size of the containment purge line penetration and the fact that those penetrations exhaust directly from the containment atmosphere to the environment, these valves may not be opened under administrative controls.

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

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

A fourth Note has been added that requires entry into the applicable Conditions and Required Actions of LCO 3.6.1 when leakage results in exceeding the overall containment leakage limit.

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

In the event one containment isolation valve in one or more penetration flow paths is inoperable [except for purge valve leakage and shield building bypass leakage not within limit], the affected penetration flow path must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic containment isolation valve, a closed manual valve, a blind

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

flange, and a check valve with flow through the valve secured. For penetrations isolated in accordance with Required Action A.1, the device used to isolate the penetration should be the closest available one to containment. Required Action A.1 must be completed within the 4 hour Completion Time. The 4 hour Completion Time is reasonable, considering the time required to isolate the penetration and the relative importance of supporting containment OPERABILITY during MODES 1, 2, 3, and 4.

For affected penetration flow paths that cannot be restored to OPERABLE status within the 4 hour Completion Time and that have been isolated in accordance with Required Action A.1, the affected penetration flow paths must be verified to be isolated on a periodic basis. This is necessary to ensure that containment penetrations required to be isolated following an accident and no longer capable of being automatically isolated will be in the isolation position should an event occur. This Required Action does not require any testing or device manipulation. Rather, it involves verification, through a system walkdown, that those isolation devices outside containment and capable of being mispositioned are in the correct position. The Completion Time of "once per 31 days for isolation devices outside containment" is appropriate considering the fact that the devices are operated under administrative controls and the probability of their misalignment is low. For the isolation devices inside containment, the time period specified as "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is based on engineering judgment and is considered reasonable in view of the inaccessibility of the isolation devices and other administrative controls that will ensure that isolation device misalignment is an unlikely possibility.

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

Required Action A.2 is modified by a Note that applies to isolation devices located in high radiation areas and allows these devices to be verified closed by use of administrative

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of these devices, once they have been verified to be in the proper position, is small.

B.1

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

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

C.1 and C.2

With one or more penetration flow paths with one containment isolation valve inoperable, the inoperable valve must be restored to OPERABLE status or the affected penetration flow

(continued)

ACTIONS

C.1 and C.2 (continued)

path must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, a closed manual valve, and a blind flange. A check valve may not be used to isolate the affected penetration. Required Action C.1 must be completed within the [4] hour Completion Time. The specified time period is reasonable, considering the relative stability of the closed system (hence, reliability) to act as a penetration isolation boundary and the relative importance of supporting containment OPERABILITY during MODES 1, 2, 3, and 4. In the event the affected penetration is isolated in accordance with Required Action C.1, the affected penetration flow path must be verified to be isolated on a periodic basis. This is necessary to assure leak tightness of containment and that containment penetrations requiring isolation following an accident are isolated. The Completion Time of once per 31 days for verifying that each affected penetration flow path is isolated is appropriate considering the valves are operated under administrative controls and the probability of their misalignment is low.

Condition C is modified by a Note indicating that this Condition is only applicable to those penetration flow paths with only one containment isolation valve and a closed system. This Note is necessary since this Condition is written to specifically address those penetration flow paths in a closed system.

Required Action C.2 is modified by a Note that applies to valves and blind flanges located in high radiation areas and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of these valves, once they have been verified to be in the proper position, is small.

[D.1

With the secondary containment bypass leakage rate not within limit, the assumptions of the safety analysis are not]

(continued)

BASES

ACTIONS

D.1 (continued)

met. Therefore, the leakage must be restored to within limit within 4 hours. Restoration can be accomplished by isolating the penetration(s) that caused the limit to be exceeded by use of one closed and de-activated automatic valve, closed manual valve, or blind flange. When a penetration is isolated, the leakage rate for the isolated penetration is assumed to be the actual pathway leakage through the isolation device. If two isolation devices are used to isolate the penetration, the leakage rate is assumed to be the lesser actual pathway leakage of the two devices. The 4 hour Completion Time is reasonable considering the time required to restore the leakage by isolating the penetration(s) and the relative importance of secondary containment bypass leakage to the overall containment function.

E.1, E.2, and E.3

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

In accordance with Required Action E.2, this penetration flow path must be verified to be isolated on a periodic basis. The periodic verification is necessary to ensure that containment penetrations required to be isolated following an accident, which are no longer capable of being automatically isolated, will be in the isolation position should an event occur. This Required Action does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those

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BASES

ACTIONS

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

isolation devices outside containment capable of being mispositioned are in the correct position. For the isolation devices inside containment, the time period specified as "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is based on engineering judgment and is considered reasonable in view of the inaccessibility of the isolation devices and other administrative controls that will ensure that isolation device misalignment is an unlikely possibility.

For the containment purge valve with resilient seal that is isolated in accordance with Required Action E.1, SR 3.6.3.6 must be performed at least once every [92] days. This assures that degradation of the resilient seal is detected and confirms that the leakage rate of the containment purge valve does not increase during the time the penetration is isolated. The normal Frequency for SR 3.6.3.6, 184 days, is based on an NRC initiative, Generic Issue B-20 (Ref. 3). 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 and has been shown to be acceptable based on operating experience.

F.1 and F.2

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

SURVEILLANCE REQUIREMENTS

SR 3.6.3.1

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

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.3.1 (continued)

opening of a containment purge valve. Detailed analysis of the purge valves failed to conclusively demonstrate their ability to close during a LOCA in time to limit offsite doses. Therefore, these valves are required to be in the sealed closed position during MODES 1, 2, 3, and 4. A containment purge valve that is sealed closed must have motive power to the valve operator removed. This can be accomplished by de-energizing the source of electric power or by removing the air supply to the valve operator. In this application, the term "sealed" has no connotation of leak tightness. The Frequency is a result of an NRC initiative, Generic Issue B-24 (Ref. 4), related to containment purge valve use during unit operations. This SR is not required to be met while in Condition E of this LCO. This is reasonable since the penetration flow path would be isolated.

SR 3.6.3.2

This SR ensures that the minipurge valves are closed as required or, if open, open for an allowable reason. If a purge valve is open in violation of this SR, the valve is considered inoperable. If the inoperable valve is not otherwise known to have excessive leakage when closed, it is not considered to have leakage outside of limits. The SR is not required to be met when the purge valves are open for pressure control, ALARA or air quality considerations for personnel entry, or for Surveillances that require the valves to be open. The minipurge valves are capable of closing in the environment following a LOCA. Therefore, these valves are allowed to be open for limited periods of time. The 31 day Frequency is consistent with other containment isolation valve requirements discussed in SR 3.6.3.3.

SR 3.6.3.3

This SR requires verification that each containment isolation manual valve and blind flange located outside containment and required to be closed during accident conditions is closed. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside the

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.3.3 (continued)

containment boundary is within design limits. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those containment isolation valves outside containment and capable of being mispositioned are in the correct position. Since verification of valve position for containment isolation valves outside containment is relatively easy, the 31 day Frequency is based on engineering judgment and was chosen to provide added assurance of the correct positions. Containment isolation valves that are open under administrative controls are not required to meet the SR during the time the valves are open.

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

SR 3.6.3.4

This SR requires verification that each containment isolation manual valve and blind flange located inside containment and required to be closed during accident conditions is closed. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside the containment boundary is within design limits. For containment isolation valves inside containment, the Frequency of "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is appropriate, since these containment isolation valves are operated under administrative controls and the probability of their misalignment is low. Containment isolation valves that are open under administrative controls are not required to meet the SR during the time that they are open.

The Note allows valves and blind flanges located in high radiation areas to be verified closed by use of administrative means. Allowing verification by

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.3.4 (continued)

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

SR 3.6.3.5

Verifying that the isolation time of each power operated and automatic containment isolation valve is within limits is required to demon.trate 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 Testing Program or 92 days.]

SR 3.6.3.6

For containment purge valves with resilient seals, additional leakage rate testing beyond the test requirements of 10 CFR 50, Appendix J (Ref. 5), is required to ensure OPERABILITY. Operating experience has demonstrated that this type of seal has the potential to degrade in a shorter time period than do other seal types. Based on this observation and the importance of maintaining this penetration leak tight (due to the direct path between containment and the environment), a Frequency of 184 days was established as part of the NRC resolution of Generic Issue B-20, "Containment Leakage Due to Seal Deterioration" (Ref. 3).

Additionally, this SR must be performed within 92 days after opening the valve. The 92 day Frequency was chosen recognizing that cycling the valve could introduce additional seal degradation (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.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.3.7

Automatic containment isolation valves close on a containment isolation signal to prevent leakage of radioactive material from containment following a DBA. This SR ensures each automatic containment isolation valve will actuate to its isolation position on a containment isolation actuation signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The [18] month Frequency was developed considering it is prudent that this SR be performed only during a unit 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.

SR 3.6.3.8

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

Verifying that each [42] inch containment purge valve is blocked to restrict opening to ≤ [50]% is required to ensure that the valves can close under DBA conditions within the times assumed in the analyses of References 2 and 3. If a LOCA occurs, the purge valves must close to maintain containment leakage within the valves assumed in the accident analysis. At other times when purge valves are required to be capable of closing (e.g., during movement of irradiated fuel assemblies), pressurization concerns are not present, thus the purge valves can be fully open. The [18] month Frequency is appropriate because the blocking devices are typically removed only during a refueling outage.

SR 3.6.3.9

This SR ensures that the combined leakage rate of all secondary containment bypass leakage paths is less than or

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.3.9 (continued)

equal to the specified leakage rate. This provides assurance that the assumptions in the safety analysis are met. The leakage rate of each bypass leakage path is assumed to be the maximum pathway leakage (leakage through the worse of the two isolation valves) unless the penetration is isolated by use of one closed and de-activated automatic valve, closed manual valve, or blind flange. In this case, the leakage rate of the isolated bypass leakage path is assumed to be the actual pathway leakage through the isolation device. If both isolation valves in the penetration are closed, the actual leakage rate is the lesser leakage rate of the two valves. This method of quantifying maximum pathway leakage is only to be used for this SR (i.e., Appendix J maximum pathway leakage limits are to be quantified in accordance with Appendix J). The Frequency is required by 10 CFR 50, Appendix J, as modified by approved exemptions (and therefore, the Frequency extensions of SR 3.0.2 may not be applied), since the testing is an Appendix J, Type C test. This SR simply imposes additional acceptance criteria.

[Bypass leakage is considered part of L_a. [Reviewer's Note: Unless specifically exempted].]

REFERENCES

1. FSAR, Section [ ].
2. FSAR, Section [ ].
3. Generic Issue B-20.
4. Generic Issue B-24.
5. 10 CFR 50, Appendix J.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4A Containment Pressure (Atmospheric)

BASES

BACKGROUND

The containment pressure is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a loss of coolant accident (LOCA) or 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. The containment pressure limits are derived from the input conditions used in the containment functional analyses and the containment structure external pressure analysis. Should operation occur outside these limits coincident with a Design Basis Accident (DBA), post accident containment pressures could exceed calculated values.

APPLICABLE SAFETY ANALYSES

Containment internal pressure is an initial condition used in the DBA analyses to establish the maximum peak containment internal pressure. The limiting DBAs considered for determining the maximum containment internal pressure (P_a) are the LOCA and MSLB. An MSLB at 102% RTP results in the highest calculated internal containment pressure of 55.7 psig, which is below the internal design pressure of 60 psig. 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). 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 an MSLB of [55.7] psig. The LCO limit 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

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

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 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 pressure differential following the inadvertent actuation of the Containment Spray System.

APPLICABILITY

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

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, maintaining containment pressure within the limits of the LCO is not required in MODE 5 or 6.

(continued)

BASES (continued)

ACTIONS

A.1

When containment pressure is not within the limits of the LCO, containment pressure must be restored to within these limits within 1 hour. The Required Action is necessary to return operation to within the bounds of the containment analysis. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1, "Containment," which requires that containment be restored to OPERABLE status within 1 hour.

B.1 and B.2

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

SURVEILLANCE REQUIREMENTS

SR 3.6.4A.1

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

REFERENCES

None.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4B Containment Pressure (Dual)

BASES

BACKGROUND

The containment pressure is limited, during normal operation, to preserve the initial conditions assumed in the accident analyses for a loss of coolant accident (LOCA) or 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. The containment pressure limits are derived from the input conditions used in the containment functional analyses and the containment structure external pressure analysis. Should operation occur outside these limits coincident with a Design Basis Accident (DBA), post accident containment pressures could exceed calculated values.

APPLICABLE SAFETY ANALYSES

Containment internal pressure is an initial condition used in the DBA analyses to establish the maximum peak containment internal pressure. The limiting DBAs considered for determining the maximum containment internal pressure (P_a) are the LOCA and MSLB. An MSLB at 75% RTP results in the highest calculated internal containment pressure of 42.3 psig, which is below the internal design pressure of 44 psig. 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). 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] 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

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

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.

Containment pressure satisfies Criterion 2 of the NRC 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.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. Since maintaining containment pressure within limits is essential to ensure initial conditions assumed in the accident analysis are maintained, the LCO is applicable in MODES 1, 2, 3, and 4. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES.

ACTIONS

A.1

When containment pressure is not within the limits of the LCO, containment pressure must be restored to within these limits within 1 hour. The Required Action is necessary to return operation to within the bounds of the containment analysis. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1, "Containment," which requires that containment be restored to OPERABLE status within 1 hour.

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

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

SURVEILLANCE REQUIREMENTS

SR 3.6.4B.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 trending of containment pressure variations during the applicable MODES. Furthermore, the 12 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal containment pressure condition.

REFERENCES

None.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.5 Containment Air Temperature (Atmospheric and 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). 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).

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 unit 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 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. 1). Operation with containment temperature in excess of the LCO limit violates an initial condition assumed in the accident analysis.

APPLICABLE SAFETY ANALYSES

Containment average air temperature is an initial condition used in the DBA analyses that establishes the containment environmental qualification operating envelope for both pressure and temperature. The limit for containment average air temperature ensures that operation is maintained within the assumptions used in the DBA analysis 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

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

initial pre-accident temperature inside containment was assumed to be [120]°F (Ref. 2).

For atmospheric containment, 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.

For dual containment, the initial containment condition of [120]°F resulted in a maximum vapor temperature in containment of [413.5]°F. The temperature of the containment steel pressure vessel also reaches approximately [413.5]°F. The containment average temperature limit of [120]°F ensures that, in the event of an accident, the maximum design temperature for containment of [269.3]°F during LOCA conditions and [413.5]°F during MSLB conditions is not exceeded. The consequences 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 Policy Statement.

LCO

During a DBA, with an initial containment average air temperature less than or equal to the LCO temperature limit, the resultant peak accident temperature is maintained below the containment design temperature. As a result, the ability of containment to perform its function is ensured.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, maintaining containment average air temperature within the limit is not required in MODE 5 or 6.

(continued)

BASES (continued)

ACTIONS

A.1

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

B.1 and B.2

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

SURVEILLANCE REQUIREMENTS

SR 3.6.5.1

Verifying that containment average air temperature is within the LCO limit ensures that containment operation remains within the limit assumed for the containment analyses. In order to determine the containment average air temperature, an arithmetic average is calculated using measurements taken at locations within the containment selected to provide a representative sample of the overall containment atmosphere. The 24 hour Frequency of this SR is considered acceptable based on 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.

(continued)

BASES (continued)

REFERENCES

1. FSAR, Section [ ].

2. FSAR, Section [ ].

B 3.6 CONTAINMENT SYSTEMS

B 3.6.6A Containment Spray and Cooling Systems (Atmospheric and Dual) (Credit taken for iodine removal by the Containment Spray System)

BASES

BACKGROUND

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

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

Containment Spray System

The Containment Spray System consists of two separate trains of equal capacity, each capable of meeting the design bases. Each train includes a containment spray pump, spray headers, nozzles, valves, and piping. Each train is powered from a separate 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).

(continued)

BASES

BACKGROUND

Containment Spray System (continued)

The Containment Spray System provides a spray of cold borated water mixed with sodium hydroxide from the spray additive tank into the upper regions of containment to reduce containment pressure and temperature and to reduce the concentration of fission products in the containment atmosphere 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 and 100% of the iodine removal design bases.

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 is available 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 that aligns valves from the containment spray pump suction to the containment sump. 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

(continued)

BASES

BACKGROUND

Containment Spray System (continued)

operator in accordance with the emergency operating procedures.

Containment Cooling System

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.

APPLICABLE SAFETY ANALYSES

The Containment Spray System and Containment Cooling System limit the temperature and pressure that could be experienced following a DBA. The limiting DBAs considered relative to containment temperature and pressure 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 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 one train of the 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 [413]°F

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

(experienced during an MSLB). Both results are within the design. (See the Bases for Specifications 3.6.4A and 3.6.4B, "Containment Pressure," and 3.6.5, "Containment Air Temperature," for a detailed discussion.) The analyses and evaluations assume a power level of [102]% 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 reduces the containment pressure to [-2.8] psig due to the sudden cooling effect in the interior of the air tight containment. Additional discussion is provided in the Bases for Specifications 3.6.4A, 3.6.4B, and 3.6.12, "Vacuum Relief Valves."

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 achieve full flow through the containment spray nozzles. The Containment Spray System total response time of [60] seconds includes diesel generator startup (for loss of offsite power), block loading of equipment, containment spray pump startup, and spray line filling (Ref. 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.

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

The Containment Spray System and the Containment Cooling System satisfy Criterion 3 of the NRC Policy Statement.

(continued)

BASES (continued)

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 concentrations below those assumed in the safety analysis. To ensure that these requirements are met, two containment spray trains and two containment cooling units must be OPERABLE. Therefore, in the event of an accident, the minimum requirements are met, assuming that the worst case single active failure occurs.

Each Containment Spray System typically includes a spray pump, spray headers, nozzles, valves, piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the RWT upon an ESF actuation signal and automatically transferring suction to the containment sump.

Each Containment Cooling System typically includes demisters, cooling coils, dampers, fans, instruments, and controls to ensure an OPERABLE flow path.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment and an increase in containment pressure and temperature, requiring the operation of the containment spray trains and containment cooling trains.

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

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BASES

ACTIONS

A.1 (continued)

removal capability afforded by the Containment Spray System, reasonable time for repairs, and the low probability of a DBA occurring during this period.

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

B.1 and B.2

If the inoperable containment spray train cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 84 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems. The extended interval to reach MODE 5 allows additional time for the restoration of the containment spray train and is reasonable when considering that the driving force for a release of radioactive material from the Reactor Coolant System is reduced in MODE 3.

C.1

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

(continued)

BASES

ACTIONS

C.1 (continued)

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

D.1

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

E.1 and E.2

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

F.1

With two containment spray trains or any combination of three or more Containment Spray System and Containment Cooling System trains inoperable, the unit is in a condition

(continued)

BASES

ACTIONS

F.1 (continued)

outside the accident analysis. Therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.6.6A.1

Verifying the correct alignment for manual, power operated, and automatic valves in the containment spray flow path provides assurance that the proper flow paths will exist for Containment Spray System operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position since these were verified to be in the correct position prior to being secured. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This SR does not require any testing or valve manipulation. Rather, it involves verifying, through a system walkdown, that those valves outside containment and capable of potentially being mispositioned are in the correct position.

SR 3.6.6A.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 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 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). Also considered in selecting this Frequency were the known reliability of the Cooling Water System, the two train

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.6A.3 (continued)

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

Verifying 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 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 Testing Program.

SR 3.6.6A.6 and SR 3.6.6A.7

These SRs verify that each automatic containment spray valve actuates to its correct position and that each containment spray pump starts upon receipt of an actual or simulated actuation signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The [18] month Frequency is based on the need to perform these Surveillances under the conditions that apply during a plant

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.6A.6 and SR 3.6.6A.7 (continued)

outage and the potential for an unplanned transient if the Surveillances were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillances when performed at the [18] month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

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 verifies that each containment cooling train actuates upon receipt of an actual or simulated actuation signal. The [18] month Frequency is based on engineering judgment and has been shown to be acceptable through operating experience. See SR 3.6.6A.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 at] 10 year intervals is considered adequate to detect obstruction of the spray nozzles.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 38, GDC 39, GDC 40, GDC 41, GDC 42, and GDC 43.
2. FSAR, Section [ ].
3. FSAR, Section [ ].

(continued)

BASES

REFERENCES (continued)	4.	FSAR, Section [ ].
	5.	FSAR, Section [ ].
	6.	ASME, Boiler and Pressure Vessel Code, Section XI.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.6B Containment Spray and Cooling Systems (Atmospheric and Dual) (Credit not taken for iodine removal by the Containment Spray System)

BASES

BACKGROUND

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

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

<u>Containment Spray System</u>

The Containment Spray System consists of two separate trains of equal capacity, each capable of meeting the design bases. Each train includes a containment spray pump, spray headers, nozzles, valves, and piping. Each train is powered from a separate 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).

(continued)

BASES

BACKGROUND

Containment Spray System (continued)

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.

The Containment Sp ay 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 is available 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 that aligns valves from the containment spray pump suction to the containment sump. 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

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 and pressurizer compartments.

(continued)

BASES

BACKGROUND

Containment Cooling System (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.

APPLICABLE SAFETY ANALYSES

The Containment Spray System and Containment Cooling System limit the temperature and pressure that could be experienced following a DBA. The limiting DBAs considered relative to containment temperature and pressure 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 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 one train of 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 the Bases for Specifications 3.6.4A and 3.6.4B, "Containment Pressure," and 3.6.5, "Containment Air Temperature," for a detailed discussion.) The analyses and evaluations assume a power level of [102]% 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 conservative peak calculated containment pressure and temperature responses.

The effect of an inadvertent containment spray actuation has been analyzed. An inadvertent spray actuation reduces the containment pressure to [-2.8] psig due to the sudden

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

cooling effect in the interior of the air tight containment. Additional discussion is provided in the Bases for Specifications 3.6.4A, 3.6.4B, and 3.6.12, "Vacuum Relief Valves."

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 achieve full flow through the containment spray nozzles. The Containment Spray System total response time of [60] seconds includes diesel generator startup (for loss of offsite power), block loading of equipment, containment spray pump startup, and spray line filling (Ref. 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].

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

The Containment Spray System and the Containment Cooling System satisfy Criterion 3 of the NRC 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). To ensure that these requirements are met, two containment spray trains and two containment cooling units must be OPERABLE. Therefore, in the event of an accident, the minimum requirements are met, assuming the worst case single active failure occurs.

Each Containment Spray System typically includes a spray pump, spray headers, nozzles, valves, piping, instruments, and controls to ensure an OPERABLE flow path capable of

(continued)

BASES

LCO (continued)

taking suction from the RWT upon an ESF actuation signal and automatically transferring suction to the containment sump.

Each Containment Cooling System typically includes demisters, cooling coils, dampers, fans, instruments, and controls to ensure an OPERABLE flow path.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment and an increase in containment pressure and temperature requiring the operation of the containment spray trains and containment cooling trains.

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

The 14 day portion of the Completion Time for Required Action A.1 is based upon engineering judgment. It takes into account the low probability of coincident entry into two Conditions in this Specification coupled with the low probability of an accident occurring during this time. Refer to Section 1.3, "Completion Times," for a more detailed discussion of the purpose of the "from discovery of failure to meet the LCO" portion of the Completion Time.

(continued)

BASES

ACTIONS (continued)

B.1

With one required containment cooling train 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.

The 14 day portion of the Completion Time for Required Action B.1 is based upon engineering judgment. It takes into account the low probability of coincident entry into two Conditions in this Specification coupled with the low probability of an accident occurring during this time. Refer to Section 1.3 for a more detailed discussion of the purpose of the "from discovery of failure to meet the LCO" portion of the Completion Time.

C.1

With two 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 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 required containment spray train 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 at least 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.

(continued)

BASES

ACTIONS (continued)

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 and F.2

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

G.1

With any combination of three or more Containment Spray System and Containment Cooling System trains inoperable, the unit is in a condition outside the accident analysis. Therefore, LCO 3.0.3 must be entered immediately.

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. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.6B.1 (continued)

those valves outside 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). 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 3.6.6B.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.]

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.6B.5

Verifying 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 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 Testing Program.

SR 3.6.6B.6 and SR 3.6.6B.7

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

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 verifies each containment cooling train actuates upon receipt of an actual or simulated actuation signal. The [18] month Frequency is based on engineering judgment and has been shown to be acceptable through operating

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.6B.8 (continued)

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 at] 10 year intervals is considered adequate to detect obstruction of the spray nozzles.

REFERENCES

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

B 3.6 CONTAINMENT SYSTEMS

B 3.6.7 Spray Additive System (Atmospheric and 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).

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 additive tank 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 spray chemical addition tank, the spray chemical addition pumps stop and the injection and isolation valves close (Ref. 1).

The Spray Additive System aids in reducing the iodine fission production inventory in the containment atmosphere.

APPLICABLE SAFETY ANALYSES

The Spray Additive System is essential to the effective removal of airborne iodine within containment following a Design Basis Accident (DBA).

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Following the assumed release of radioactive materials into containment, the containment is assumed to leak at its design value following the accident.

The DBA response time assumed for the Spray Additive System is the same as for the Containment Spray System and is discussed in the Bases for Specification 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 Spray Additive System satisfies Criterion 3 of the NRC Policy Statement.

LCO

The Spray Additive System is necessary to reduce the release of radioactive material to the environment in the event of a DBA. To be considered OPERABLE, the volume and concentration of the spray additive solution must be sufficient to 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.

(continued)

BASES (continued)

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment requiring the operation of the Spray Additive System. The Spray Additive System assists in reducing the iodine fission product inventory prior to release to the environment.

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Thus, the Spray Additive System is not required to be OPERABLE in MODES 5 and 6.

ACTIONS

A.1

With the Spray Additive System inoperable, the system must be restored to OPERABLE status within 72 hours. 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 path capabilities and the low probability of the worst case DBA occurring during this period.

B.1 and B.2

If the Spray Additive System cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 84 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems. 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 Reactor Coolant System.

(continued)

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. 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 contents are normally acidic, the volume of the spray additive tank must provide a sufficient volume of spray additive to adjust pH for all water injected. This SR is performed to verify the availability of sufficient hydrazine (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 unit operations). Tank level is also indicated and alarmed in the control room, such that there is a high confidence that a substantial change in level would be detected.

SR 3.6.7.3

This SR provides verification of the N_2H_4 concentration in the spray additive tank and is sufficient to ensure that the spray solution being injected into containment is at the correct pH level. The concentration of N_2H_4 in the spray additive tank must be determined by chemical analysis. The 184 day Frequency is sufficient to ensure that the concentration level of N_2H_4 in the spray additive tank remains within the established limits. This is based on the low likelihood of an uncontrolled change in concentration (the tank is normally isolated) and the probability that any substantial variance in tank volume will be detected.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.7.4

The chemical addition pump must be verified to provide the flow rate assumed in the accident analysis to the Containment Spray System. The Spray Additive System is not operated during normal 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 spray additive tank. The differential pressure obtained by the pump on recirculation is analogous to the full spray additive flow provided to the Containment Spray System on an actual CSAS. The Frequency of this SR is in accordance with the Inservice Testing Program and is sufficient to identify component degradation that may affect flow rate.

SR 3.6.7.5

This SR verifies that each automatic valve in the Spray Additive System flow path actuates to its correct position on a CSAS. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the [18] month Frequency. Therefore, the Frequency 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 verified 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 the passive nature of the spray additive flow controls, the

(continued)

BASES

SURVEILLANCE REQUIREMENTS	SR 3.6.7.6 (continued) 5 year Frequency is sufficient to identify component degradation that may affect flow rate.
REFERENCES	1. FSAR, Section [ ].

B 3.6 CONTAINMENT SYSTEMS

B 3.6.8 Hydrogen Recombiners (Atmospheric and Dual)(If permanently installed)

BASES

BACKGROUND

The function of the hydrogen recombiners is to eliminate 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. The hydrogen recombiners are manually initiated since flammability limits would not be reached until several days after a Design Basis Accident (DBA).

Two 100% capacity independent hydrogen recombiners are provided. Each consists of controls located in the control room, a power supply, and a recombiner located 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 recombiner. 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 Features bus and is provided with a separate power panel and control panel.

APPLICABLE SAFETY ANALYSES

The hydrogen recombiners provide for controlling the bulk hydrogen concentration in containment to less than the lower flammable concentration of 4.1 v/o following a DBA. This control would prevent a containmentwide hydrogen burn, thus ensuring the pressure and temperature assumed in the analysis are not exceeded and minimizing damage to safety

(continued)

BASES

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 Systems solutions.

To evaluate the potential for hydrogen accumulation in containment following a LOCA, the hydrogen generation as a function of time following the initiation of the accident is calculated. Conservative assumptions recommended in Reference 3 are used to maximize the amount of hydrogen calculated.

The hydrogen recombiners satisfy Criterion 3 of the NRC Policy Statement.

LCO

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

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

APPLICABILITY

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

(continued)

BASES

APPLICABILITY (continued)

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

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

ACTIONS

A.1

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

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

B.1 and B.2

[Reviewer's Note: This Condition is only allowed for units with an alternate hydrogen control system acceptable to the technical staff.]

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

With two hydrogen recombiners inoperable, the ability to perform the hydrogen control function via alternate capabilities must be verified by administrative means within 1 hour. The alternate hydrogen control capabilities are provided by [the containment Hydrogen Purge System/hydrogen recombiner/Hydrogen Ignitor System/Hydrogen Mixing System/Containment Air Dilution System/Containment Inerting System]. The 1 hour Completion Time allows a reasonable period of time to verify that a loss of hydrogen control function does not exist. [Reviewer's Note: The following is to be used if a non-Technical Specification alternate hydrogen control function is used to justify this Condition: In addition, the alternate hydrogen control system capability must be verified every 12 hours thereafter to ensure its continued availability.] [Both] the [initial] verification [and all subsequent verifications] may be performed as an administrative check, by examining logs or other information to determine the availability of the alternate hydrogen control system. It does not mean to perform the Surveillances needed to demonstrate OPERABILITY of the alternate hydrogen control system. If the ability to perform the hydrogen control function is maintained, continued operation is permitted with two hydrogen recombiners inoperable for up to 7 days. Seven days is a reasonable time to allow two hydrogen recombiners to be inoperable because the hydrogen control function is maintained and because of the low probability of the occurrence of a LOCA that would generate hydrogen in amounts capable of exceeding the flammability limit.

C.1

If the inoperable hydrogen recombiner(s) cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.8.1

Performance of a system functional test for each hydrogen recombiner ensures that the recombiners are operational and can 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. Operating experience has shown that these components usually pass the Surveillance when performed at the [18] month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.8.2

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

SR 3.6.8.3

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

REFERENCES

1. 10 CFR 50.44.
2. 10 CFR 50, Appendix A, GDC 41.
3. Regulatory Guide 1.7, Revision [1].

B 3.6 CONTAINMENT SYSTEMS

B 3.6.9 Hydrogen Mixing System (HMS) (Atmospheric and Dual)

BASES

BACKGROUND

The HMS reduces the potential for breach of containment due to a hydrogen oxygen reaction 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 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 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 accumulations of hydrogen from exceeding the flammability limit of 4.1 volume percent (v/o).

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The HMS mixes the containment atmosphere to provide a uniform hydrogen concentration.

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

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

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

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

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

To evaluate the potential for hydrogen accumulation in containment following a LOCA, the hydrogen generation as a function of time following the initiation of the accident is calculated. Conservative assumptions recommended by Reference 3 are used to maximize the amount of hydrogen calculated.

The HMS satisfies Criterion 3 of the NRC Policy Statement.

LCO

Two HMS trains must be OPERABLE, with power to each from an independent, safety related power supply. Each train typically consists of two fans with their own motors and controls and is automatically initiated by a CCAS.

Operation with at least one HMS train provides the mixing necessary to ensure uniform hydrogen concentration throughout containment.

APPLICABILITY

In MODES 1 and 2, 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.

(continued)

BASES

APPLICABILITY (continued)

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

In MODES 5 and 6, the probability and consequences of a LOCA or main steam line break are low due to the pressure and temperature limitations of these MODES. Therefore, the HMS is not required in these MODES.

ACTIONS

A.1

With one HMS 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 other HMS train, the small probability of a LOCA or SLB occurring (that would generate an amount of hydrogen that exceeds the flammability limit), the amount of time available after a LOCA or SLB (should one occur) for operator action to prevent hydrogen accumulation from exceeding the flammability limit, and the availability of the hydrogen recombiners, Containment Spray System, Hydrogen Purge System, and hydrogen monitors.

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

B.1 and B.2

[Reviewer's Note: This Condition is only allowed for units with an alternate hydrogen control system acceptable to the technical staff.]

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

With two HMS inoperable, the ability to perform the hydrogen control function via alternate capabilities must be verified by administrative means within 1 hour. The alternate hydrogen control capabilities are provided by [the containment Hydrogen Purge System/hydrogen recombiner/Hydrogen Ignitor System/HMS/Containment Air Dilution System/Containment Inerting System]. The 1 hour Completion Time allows a reasonable period of time to verify that a loss of hydrogen control function does not exist. [Reviewer's Note: The following is to be used if a non-Technical Specification alternate hydrogen control function is used to justify this Condition. In addition, the alternate hydrogen control system capability must be verified every 12 hours thereafter to ensure its continued availability.] [Both] the [initial] verification [and all subsequent verifications] may be performed as an administrative check, by examining logs or other information to determine the availability of the alternate hydrogen control system. It does not mean to perform the Surveillances needed to demonstrate OPERABILITY of the alternate hydrogen control system. If the ability to perform the hydrogen control function is maintained, continued operation is permitted with two HMS trains inoperable for up to 7 days. Seven days is a reasonable time to allow two HMS trains to be inoperable because the hydrogen control function is maintained and because of the low probability of the occurrence of a LOCA that would generate hydrogen in the amounts capable of exceeding the flammability limit.

C.1

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

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.9.1

Operating each HMS train for ≥ 15 minutes ensures that the train is OPERABLE and that all associated controls are functioning properly. It also ensures that blockage, fan and/or motor failure, or excessive vibration can be detected for corrective action. The 92 day Frequency is consistent with Inservice Testing Program Surveillance Frequencies, operating experience, the known reliability of the fan motors and controls, and the two train redundancy available.

SR 3.6.9.2

Verifying that each HMS train flow rate on slow speed is ≥ [37,000] cfm ensures that each train is capable of maintaining localized hydrogen concentrations below the flammability limit. The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the [18] month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.9.3

This SR ensures that the HMS responds properly to a CCAS. The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the [18] month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. 10 CFR 50.44.
2. 10 CFR 50, Appendix A, GDC 41.
3. Regulatory Guide 1.7, Revision [1].

B 3.6 CONTAINMENT SYSTEMS

B 3.6.10 Iodine Cleanup System (ICS) (Atmospheric and Dual)

BASES

BACKGROUND

The ICS is provided per GDC 41, "Containment Atmosphere Cleanup," GDC 42, "Inspection of Containment Atmosphere Cleanup Systems," and GDC 43, "Testing of Containment Atmosphere Cleanup Systems" (Ref. 1), to reduce the concentration of fission products 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 the environment.

The ICS consists of two 100% capacity separate, independent, and redundant trains. Each train includes a heater, [cooling coils,] a prefilter, a 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 adsorber section are credited in the analysis. The system initiates filtered recirculation of the containment atmosphere following receipt of a containment isolation actuation signal. The system design is described in Reference 2.

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 (Ref. 3).

The moisture separator is included for moisture (free water) removal from the gas stream. Heaters are used to heat the gas stream, which lowers the relative humidity. Continuous operation of each train for at least 10 hours per month with the heaters on reduces moisture buildup on the HEPA filters and adsorbers. Both the moisture separator and heater are important to the effectiveness of the charcoal adsorbers.

(continued)

BASES

BACKGROUND (continued)	Two ICS trains are provided to meet the requirement for separation, independence, and redundancy. Each ICS train is powered from a separate Engineered Safety Features bus and is provided with a separate power panel and control panel. [Service water is required to supply cooling water to the cooling coils.]
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 to limit potential leakage to the environment. Additionally, it is assumed that the amount of radioactive iodine release is limited by reducing the iodine concentration 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 ICS satisfies Criterion 3 of the NRC Policy Statement.
LCO	Two separate, 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.
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, and CEA ejection accident. Because these accidents are considered credible accidents in MODES 1, 2, 3, and 4, the ICS must be operable in these

(continued)

BASES

APPLICABILITY (continued)

MODES to ensure the reduction in iodine concentration assumed in the accident analysis.

In MODES 5 and 6, the probability and consequences of a LOCA are low due to the pressure and temperature limitations of these MODES. The ICS is not required in these MODES to remove iodine from the containment atmosphere.

ACTIONS

A.1

With one ICS train inoperable, the inoperable train must be restored to OPERABLE status within 7 days. The components in this degraded condition are capable of providing 100% of the iodine removal needs after a DBA. The 7 day Completion Time is based on consideration of such factors as:

a. The availability of the OPERABLE redundant ICS train;

b. The fact that, even with no ICS train in operation, almost the same amount of iodine would be removed from the containment atmosphere through absorption by the Containment Spray System; and

c. The fact that the Completion Time is adequate to make most repairs.

B.1 and B.2

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

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.10.1

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 continuous hours eliminates moisture on the adsorbers and HEPA filters. Experience from filter testing at operating units 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 3.6.10.2

This SR verifies that the required ICS filter testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The ICS filter tests are in accordance with Regulatory Guide 1.52 (Ref. 3). The VFTP includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the VFTP.

SR 3.6.10.3

The automatic startup test verifies that both trains of equipment start upon receipt of an actual or simulated test signal. The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the [18] month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint. Furthermore, the Frequency was developed considering that

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.10.3 (continued)

the system equipment OPERABILITY is demonstrated on a 31 day Frequency by SR 3.6.10.1.

SR 3.6.10.4

The ICS 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 Surveillance when performed at the [18] month Frequency.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 41, GDC 42, and GDC 43.
2. FSAR, Section [ ].
3. Regulatory Guide 1.52, Revision [1].
4. FSAR, Section [ ].

B 3.6 CONTAINMENT SYSTEMS

B 3.6.11 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 Building 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 the Bases for Specification 3.6.13, "Shield Building Exhaust Air Cleanup System (SBEACS)." 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 material from the primary containment atmosphere is restricted to those leakage paths and associated leakage rates assumed in the accident analysis.

The shield building satisfies Criterion 3 of the NRC Policy Statement.

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.

(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 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, shield building OPERABILITY is not required in MODE 5 or 6.

ACTIONS

A.1

In the event shield building OPERABILITY is not maintained, shield building OPERABILITY must be restored within 24 hours.

Twenty-four hours is a reasonable Completion Time considering the limited leakage design of the containment and the low probability of a DBA occurring during this time period.

B.1 and B.2

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

SURVEILLANCE REQUIREMENTS

SR 3.6.11.1

Verifying that shield building annulus pressure is within limit ensures that operation remains within the limit

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.11.1 (continued)

assumed in the containment analysis. The 12 hour Frequency of this SR was developed considering operating experience related to shield building annulus pressure variations and pressure instrument drift during the applicable MODES.

SR 3.6.11.2

Maintaining shield building OPERABILITY requires maintaining each door in the access opening closed except when the access opening is being used for normal transient entry and exit; then, at least one door must remain closed. The 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.

SR 3.6.11.3

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.

SR 3.6.11.4

The SBEACS is required to produce the required negative pressure of ≥ [0.25] inch water gauge during test operation within 1 minute after a start signal. The negative pressure ensures 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 material leaks from the shield building prior to developing the negative pressure.

The [18] month Frequency to verify the required negative pressure in the shield building is consistent with Regulatory Guide 1.52 (Ref. 1) guidance for functional

(continued)

BASES

SURVEILLANCE REQUIREMENTS	SR 3.6.11.4 (continued) testing of the ability of the SBEACS to "pull down" the required negative pressure every [18] months.
REFERENCES	1. Regulatory Guide 1.52, Revision [1].

B 3.6 CONTAINMENT SYSTEMS

B 3.6.12 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 or human errors are necessary to have inadvertent actuation.

The containment pressure vessel contains two 100% vacuum relief lines installed in parallel that protect the containment from excessive external loading. The vacuum relief lines are 24 inch penetrations that connect the shield building annulus to the containment. Each vacuum relief line is isolated by 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 relief line will not exceed the containment pressure vessel design external pressure differential of [0.65] psig with any prevailing atmospheric pressure.

APPLICABLE SAFETY ANALYSES

Design of the vacuum relief lines involves calculating the effect of an inadvertent containment spray actuation that can reduce the atmospheric temperature (and hence pressure) inside containment (Ref. 1). 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

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

when their negative pressure setpoint is reached. It is also assumed that one vacuum relief line 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 a containment high pressure event. For this reason, the system is designed to take the full containment positive design pressure and the containment design basis accident (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 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 lines are required to be OPERABLE to ensure that at least one is available, assuming one or both valves in the other line fail to open.

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 lines 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

(continued)

BASES

APPLICABILITY (continued)	Containment Cooling System are not required to be OPERABLE in MODES 5 and 6. Therefore, maintaining OPERABLE vacuum relief lines is not required in MODE 5 or 6.

ACTIONS

A.1

With one of the required vacuum relief lines inoperable, the inoperable line must be restored to OPERABLE status within 72 hours. The specified time period is consistent with other LCOs for the loss of one train of a system required to mitigate the consequences of a LOCA or other DBA.

B.1 and B.2

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

SURVEILLANCE REQUIREMENTS

SR 3.6.12.1

This SR references the Inservice Testing Program, which establishes the requirement that inservice testing of the ASME Code Class 1, 2, and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda (Ref. 2). Therefore, SR Frequency is governed by the Inservice Testing Program.

REFERENCES

1. FSAR, Section [6.2].

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

B 3.6 CONTAINMENT SYSTEMS

B 3.6.13 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 material leaking from the primary containment of a dual containment into the shield building (secondary containment) following a Design Basis Accident are filtered and adsorbed prior to exhausting to the environment.

The 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, a prefilter, a 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 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 2.

(continued)

BASES

BACKGROUND (continued)

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 to reduce the relative humidity of the airstream 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 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.

APPLICABLE SAFETY ANALYSES

The SBEACS design basis is established by the consequences of the limiting DBA, which is a LOCA. The accident analysis (Ref. 3) 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.

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 [$\geq$ 0.25] inch water gauge in the shield building is [1 minute]. 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 Policy Statement.

(continued)

BASES (continued)

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.

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 decrease 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.

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 this period.

B.1 and B.2

If the SBEACS train cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

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

SURVEILLANCE REQUIREMENTS

SR 3.6.13.1

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 continuous hours eliminates moisture on the adsorbers and HEPA filters. Experience from filter testing at operating units 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 3.6.13.2

This SR verifies that the required SBEACS filter testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The Emergency Ventilation System filter tests are in accordance with Regulatory Guide 1.52 (Ref. 4). The VFTP includes testing of HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the VFTP.

SR 3.6.13.3

The automatic startup ensures that each SBEACS train responds properly. The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.13.3 (continued)

unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the [18] month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint. Furthermore, the SR interval was developed considering that the SBEACS equipment OPERABILITY is demonstrated at a 31 day Frequency by SR 3.6.13.1.

SR 3.6.13.4

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 Surveillance when performed at the [18] month Frequency.

SR 3.6.13.5

The SBEACS train flow rate is verified ≥ [ ] cfm to ensure that the flow rate is adequate to "pull down" the shield building pressure as required. This test also will verify the proper functioning of the fans, dampers, filters, absorbers, etc., when this SR is performed in conjunction with SR 3.6.11.4.

The [18] month on a STAGGERED TEST BASIS Frequency is consistent with the Regulatory Guide 1.52 (Ref. 4) guidance.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 41.
2. FSAR, Section [ ].
3. FSAR, Section [ ].
4. Regulatory Guide 1.52, Revision [1].

B 3.7 PLANT SYSTEMS

B 3.7.1 Main Steam Safety Valves (MSSVs)

BASES

BACKGROUND

The primary purpose of the MSSVs is to provide overpressure protection for the secondary system. The MSSVs also provide protection against overpressurizing the reactor coolant pressure boundary (RCPB) by providing a heat sink for 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 the FSAR, Section [5.2] (Ref. 1). The MSSV rated capacity passes the full steam flow at 102% RTP (100% + 2% for instrument error) with the valves full open. This meets the requirements of the ASME Code, Section III (Ref. 2). The MSSV design includes staggered setpoints, according to Table 3.7.1-1, in the accompanying LCO, so that only the 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.

APPLICABLE SAFETY ANALYSES

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

The events that challenge the MSSV relieving capacity, and thus RCS pressure, are those characterized as decreased heat removal events, and are presented in the FSAR, Section [15.2] (Ref. 3). Of these, the full power 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 to the steam generators, RCS pressure reaches ≤ 2630 psig. This peak pressure is < 110% of the design pressure of 2500 psig, but high enough to actuate the pressurizer safety valves. The maximum relieving rate

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

during the LOCV event is 2.5 E6 lb/hour, which is less than the rated capacity of two MSSVs.

The limiting accident for peak RCS pressure is the full power feedwater line break (FWLB), 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 ≤ 2730 psig, with the pressurizer safety valves providing relief capacity. The maximum relieving rate of the MSSVs during the FWLB event is ≤ 2.5 E6 lb/hour, which is less than the rated capacity of two MSSVs.

Using conservative analysis assumptions, a small range of FWLB sizes less than a full double ended guillotine break produce an RCS pressure of 2765 psig for a period of 20 seconds; exceeding 110% (2750 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.

The MSSVs satisfy Criterion 3 of the NRC Policy Statement.

LCO

This LCO requires all MSSVs to be OPERABLE in compliance with Reference 2, 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 Reference 2 requirements), and adjustment to the Reactor Protection System trip setpoints. These limitations are according to those shown in Table 3.7.1-1, Required Action A.2, and Required Action A.3 in the accompanying LCO. An MSSV is considered inoperable if it fails to open upon demand.

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

(continued)

BASES

LCO (continued)

surveillance testing in accordance with the Inservice Testing Program.

The lift settings, according to Table 3.7.1-2 in the accompanying LCO, correspond to ambient conditions of the valve at nominal operating temperature and pressure.

This LCO provides assurance that the MSSVs will perform their designed safety function to mitigate the consequences of accidents that could result in a challenge to the RCPB.

APPLICABILITY

In MODE 1, a minimum of two MSSVs per steam generator are required to be OPERABLE, according to Table 3.7.1-1 in the accompanying LCO, which is limiting and bounds all lower MODES. In MODES 2 and 3, both the ASME Code and the accident analysis require only one MSSV per steam generator to provide overpressure protection.

In MODES 4 and 5, there are no credible transients requiring the MSSVs.

The steam generators are not normally used for heat removal in MODES 5 and 6, and thus cannot be overpressurized; there is no requirement for the MSSVs to be OPERABLE in these MODES.

ACTIONS

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

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

An alternative to restoring the inoperable MSSV(s) to OPERABLE status is to reduce power so that the available MSSV relieving capacity meets Code requirements for the power level. Operation may continue provided the allowable THERMAL POWER is equal to the product of: 1) the ratio of the number of MSSVs available per steam generator to the total number of MSSVs per steam generator, and 2) the ratio of the available relieving capacity to total steam flow, multiplied by 100%.

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

$$\text{Allowable THERMAL POWER} = \frac{(8 - N)}{8} \times 109.2$$

With one or more MSSVs inoperable, the ceiling on the variable overpower trip is reduced to an amount over the allowable THERMAL POWER equal to the band given for this trip, according to Table 3.7.1-1 in the accompanying LCO.

SP = Allowable THERMAL POWER + 9.8

where:

SP = Reduced reactor trip setpoint in percent RTP. This is a ratio of the available relieving capacity over the total steam flow at rated power.

8 = Total number of MSSVs per steam generator.

N = Number of inoperable MSSVs on the steam generator with the greatest number of inoperable valves.

109.2 = Ratio of MSSV relieving capacity at 110% steam generator design pressure to calculated steam flow rate at 100% RTP + 2% instrument uncertainty expressed as a percentage (see text above).

9.8 = Band between the maximum THERMAL POWER and the variable overpower trip setpoint ceiling (Table 3.7.1-1).

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 is consistent with A.1. An additional 8 hours is allowed to reduce the setpoints in recognition of the difficulty of resetting all channels of this trip function within a period of 8 hours. The Completion Time of 12 hours for Required Action A.3 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.

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

If the MSSVs cannot be restored to OPERABLE status in the associated Completion Time, or if one or more steam generators have less than two MSSVs OPERABLE, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within [12] hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1

This SR verifies the OPERABILITY of the MSSVs by the verification of each MSSV lift setpoints in accordance with the Inservice Testing Program. The ASME Code, Section XI (Ref. 4), requires that safety and relief valve tests be performed in accordance with ANSI/ASME OM-1-1987 (Ref. 5). According to Reference 5, the following tests are required for MSSVs:

a. Visual examination;

b. Seat tightness determination;

c. Setpoint pressure determination (lift setting);

d. Compliance with owner's seat tightness criteria; and

e. Verification of the balancing device integrity 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 ASME Code specifies the activities and frequencies necessary to satisfy the requirements. Table 3.7.1-2 allows a ± [3]% setpoint tolerance for OPERABILITY; however, the valves are reset to ± 1% during the Surveillance to allow for drift.

This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. This is to

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1 (continued)

allow testing of the MSSVs at hot conditions. The MSSVs may be either bench tested or tested in situ at hot conditions using an assist device to simulate lift pressure. If the MSSVs are not tested at hot conditions, the lift setting pressure shall be corrected to ambient conditions of the valve at operating temperature and pressure.

REFERENCES

1. FSAR, Section [5.2].
2. ASME, Boiler and Pressure Vessel Code, Section III, Article NC-7000, Class 2 Components.
3. FSAR, Section [15.2].
4. ASME, Boiler and Pressure Vessel Code, Section XI, Article IWV-3500.
5. ANSI/ASME OM-1-1987.

B 3.7 PLANT SYSTEMS

B 3.7.2 Main Steam Isolation Valves (MSIVs)

BASES

BACKGROUND	The MSIVs isolate steam flow from the secondary side of the steam generators following a high energy line break (HELB). MSIV closure terminates flow from the unaffected (intact) steam generator. One MSIV is located in each main steam line outside, but close to, containment. The MSIVs are downstream from the main steam safety valves (MSSVs), atmospheric dump valves, and auxiliary feedwater pump turbine 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 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 (MFIVs) to close. The MSIVs may also be actuated manually. A description of the MSIVs is found in the FSAR, Section [10.3] (Ref. 1).
APPLICABLE SAFETY ANALYSES	The design basis of the MSIVs is established by the containment analysis for the large steam line break (SLB) inside containment, as discussed in the FSAR, Section [6.2] (Ref. 2). It is also influenced by the accident analysis of the SLB events presented in the FSAR, Section [15.1.5] (Ref. 3). The design precludes the blowdown of more than one steam generator, assuming a single active component failure (e.g., the failure of one MSIV to close on demand). The limiting case for the containment analysis is the hot zero power SLB inside containment with a loss of offsite power following turbine trip, and failure of the MSIV on the affected steam generator to close. At zero power, the steam generator inventory and temperature are at their maximum,

(continued)

APPLICABLE SAFETY ANALYSES (continued)

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 an MFIV to close, and failure of an emergency diesel generator to start.

The accident analysis compares several different SLB events against different acceptance criteria. The large SLB outside containment upstream of the MSIV is limiting for offsite dose, although a break in this short section of main steam header has a very low probability. The large SLB inside containment at 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 (HPSI) pumps, is delayed. Significant single failures considered include: failure of a MSIV to close, failure of an emergency diesel generator, and failure of a HPSI pump.

The MSIVs serve only a safety function and remain open during power operation. These valves operate under the following situations:

a. An HELB 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 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

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

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 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 Policy Statement.

LCO

This LCO requires that the MSIV in each of the [two] steam lines be OPERABLE. The MSIVs are considered OPERABLE when the isolation times are within limits, and they close on an isolation actuation signal.

This LCO provides assurance that the MSIVs will perform their design safety function to mitigate the consequences of accidents that could result in offsite exposures comparable to the 10 CFR 100 (Ref. 4) limits or the NRC staff approved licensing basis.

(continued)

BASES (continued)

APPLICABILITY

The MSIVs must be OPERABLE in MODE 1 and in MODES 2 and 3 except when all MSIVs are closed and [deactivated] when there is significant mass and energy in the RCS and steam generators. When the MSIVs are closed, they are already performing their safety function.

In MODE 4, the steam generator energy is low; therefore, the MSIVs are not required to be OPERABLE.

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

ACTIONS

A.1

With one MSIV inoperable in MODE 1, time is allowed to restore the component to OPERABLE status. Some repairs can be made to the MSIV with the unit hot. The [8] hour Completion Time is reasonable, considering the probability of an accident occurring during the time period that would require closure of the MSIVs.

The [8] hour Completion Time is greater than that normally allowed for containment isolation valves because the MSIVs are valves that isolate a closed system penetrating containment. These valves differ from other containment isolation valves in that the closed system provides an additional means for containment isolation.

B.1

If the MSIV cannot be restored to OPERABLE status within [8] hours, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in MODE 2 within 6 hours and Condition C would be entered. The Completion Time is reasonable, based on operating experience, to reach MODE 2, and close the MSIVs in an orderly manner and without challenging unit systems.

(continued)

BASES

ACTIONS (continued)

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

Condition C is modified by a Note indicating that separate Condition entry is allowed for each MSIV.

Since the MSIVs are required to be OPERABLE in MODES 2 and 3, the inoperable MSIVs may either be restored to OPERABLE status or closed. When closed, the MSIVs are already in the position required by the assumptions in the safety analysis.

The [8] hour Completion Time is consistent with that allowed in Condition A.

Inoperable MSIVs that cannot be restored to OPERABLE status within the specified Completion Time, but are closed, must be verified on a periodic basis to be closed. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, MSIV status indications available in the control room, and other administrative controls, to ensure these valves are in the closed position.

D.1 and D.2

If the MSIVs cannot be restored to OPERABLE status, or closed, within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within [12] hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from MODE 2 conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.2.1

This SR verifies that the closure time of each MSIV is ≤ [4.6] seconds on an actual or simulated actuation signal. The MSIV closure time is assumed in the accident and containment analyses. This SR is normally performed upon returning the unit to operation following a refueling

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.2.1 (continued)

outage. The MSIVs should not be tested at power since even a part stroke exercise increases the risk of a valve closure with the unit generating power. As the MSIVs are not tested at power, they are exempt from the ASME Code, Section XI (Ref. 5), requirements during operation in MODES 1 and 2.

The Frequency for this SR is in accordance with the [Inservice Testing Program or [18] months]. This [18] month 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 at the [18] month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

This test is conducted in MODE 3, with the unit at operating temperature and pressure, as discussed in the Reference 5 exercising requirements. This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. This allows a delay of testing until MODE 3, in order to establish conditions consistent with those under which the acceptance criterion was generated.

REFERENCES

1. FSAR, Section [10.3].
2. FSAR, Section [6.2].
3. FSAR, Section [15.1.5].
4. 10 CFR 100.11.
5. ASME, Boiler and Pressure Vessel Code, Section XI, Inservice Inspection, Article IWV-3400.

B 3.7 PLANT SYSTEMS

B 3.7.3 Main Feedwater Isolation Valves (MFIVs) [and [MFIV] 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 (HELB). Closure of the MFIVs and the bypass valves terminates flow to both steam generators, terminating the event for feedwater line breaks (FWLBs) occurring upstream of the MFIVs. The consequences of events occurring in the main steam lines or in the MFW lines downstream of the MFIVs will be mitigated by their closure. Closure of the MFIVs and bypass valves effectively terminates the addition of feedwater to an affected steam generator, limiting the mass and energy release for steam line breaks (SLBs) or FWLBs inside containment, and reducing the cooldown effects for SLBs.

The MFIVs and bypass valves isolate the nonsafety 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 provide 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 FWLB.

The MFIVs and its 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 bypass valves may also be actuated manually. In addition to the MFIVs and the bypass valves, a check valve inside containment is available to isolate the feedwater line penetrating containment, and to ensure that the consequences of events do not exceed the capacity of the containment heat removal systems.

(continued)

BASES

BACKGROUND (continued)	A description of the MFIVs is found in the FSAR, Section [10.4.7] (Ref. 1).
APPLICABLE SAFETY ANALYSES	The design basis of the MFIVs is established by the analysis for the large SLB. It is also influenced by the accident analysis for the large FWLB. Closure of the MFIVs and their bypass valves may also be relied on to terminate a steam break for core response analysis and an excess feedwater flow event upon receipt of a MSIS on high steam generator level. Failure of an MFIV and the bypass valve to close following an SLB, FWLB, or excess feedwater flow 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 FWLB event. The MFIVs satisfy Criterion 3 of the NRC Policy Statement.
LCO	This LCO ensures that the MFIVs and the bypass valves will isolate MFW flow to the steam generators. Following an FWLB or SLB, these valves will also isolate the nonsafety related portions from the safety related portions of the system. This LCO requires that [two] MFIVs [and [MFIV] bypass valves] in each feedwater line be OPERABLE. The MFIVs and the bypass valves are considered OPERABLE when the isolation times are within limits, and are closed on an isolation actuation signal. Failure to meet the LCO requirements can result in additional mass and energy being released to containment following an SLB or FWLB inside containment. If 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.
APPLICABILITY	The MFIVs and the bypass valves must be OPERABLE whenever there is significant mass and energy in the Reactor Coolant

(continued)

BASES

APPLICABILITY (continued)

System and steam generators. This ensures that, in the event of an HELB, a single failure cannot result in the blowdown of more than one steam generator.

In MODES 1, 2, and 3, the MFIV [or [MFIV] bypass valves] are required to be OPERABLE, except when they are closed and deactivated or isolated by a closed manual valve, 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 and deactivated or isolated by a closed manual valve, they are already performing their safety function.

In MODES 4, 5, and 6, steam generator energy is low. Therefore, the MFIVs and the bypass valves are normally closed since MFW is not required.

ACTIONS

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

A.1 and A.2

With one MFIV or the bypass valve inoperable, action must be taken to close or isolate the inoperable valves within [8 or 72] hours. When these valves are closed or isolated, they are performing their required safety function (e.g., to isolate the line).

For units with only one MFIV per feedwater line: The [8] hour Completion Time is reasonable to close the MFIV or its bypass valve, which includes performing a controlled unit shutdown to MODE 2.

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

B.1

If more than one MFIV or [MFIV] bypass valve in the same flow path cannot be restored to OPERABLE status, then there

(continued)

BASES

ACTIONS

B.1 (continued)

may be no redundant system to operate automatically and perform the required safety function. Although the containment can be isolated with the failure of two valves in parallel in the same flow path, the double failure can be an indication of a common mode failure in the valves of this flow path, and as such is treated the same as a loss of the isolation capability of this flow path. Under these conditions, valves in each flow path must be restored to OPERABLE status, closed, or the 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 reasonable to close the MFIV or its bypass valve, or otherwise isolate the affected flow path.

Inoperable MFIVs and [MFIV] bypass valves that cannot be restored to OPERABLE status within the Completion Time, but are closed or isolated, must be verified on a periodic basis that they are closed or isolated. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of valve status indications available in the control room, and other administrative controls to ensure that these valves are closed or isolated.

C.1 and C.2

If the MFIVs and their bypass valves cannot be restored to OPERABLE status, closed, or isolated in the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within [12] hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1

This SR ensures the verification of each MFIV [and [MFIV] bypass valve] is ≤ [7] seconds on an actual or simulated

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1 (continued)

actuation signal. The MFIV closure time is assumed in the accident and containment analyses. This Surveillance is normally performed upon returning the unit to operation following a refueling outage. The MFIVs should not be tested at power since even a part stroke exercise increases the risk of a valve closure with the unit generating power. As these valves are not tested at power, they are exempt from the ASME Code, Section XI (Ref. 2) requirements during operation in MODES 1 and 2.

The Frequency is in accordance with the [Inservice Testing Program or [18] months]. The [18] month Frequency for valve closure time is based on the refueling cycle. Operating experience has shown that these components usually pass the SR when performed at the [18] month Frequency.

REFERENCES

1. FSAR, Section [10.4.7].
2. ASME, Boiler and Pressure Vessel Code, Section XI, Inservice Inspection, Article IWV-3400.

B 3.7 PLANT SYSTEMS

B 3.7.4 Atmospheric Dump Valves (ADVs)

BASES

BACKGROUND

The ADVs provide a safety grade method for cooling the unit to Shutdown Cooling (SDC) System entry conditions, should the preferred heat sink via the Steam Bypass System to the condenser not be available, as discussed in the FSAR, Section [10.3] (Ref. 1). This is done in conjunction with the 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 usually provided with a pressurized gas supply of bottled nitrogen that, 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 the SDC System 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.

APPLICABLE SAFETY ANALYSES

The design basis of the ADVs is established by the capability to cool the unit to SDC System entry conditions. A cooldown rate of 75°F per hour is obtainable by one or

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

both steam generators. This design is adequate to cool the unit to SDC System 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 unit to SDC System 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 steam generator pressure and temperature at the MSSV 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 upstream of the main steam isolation valves, a feedwater line break, and an SGTR event (although the ADVs on the affected steam generator may still be available following a SGTR event).

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 opens, or fails to close during use.

The ADVs satisfy Criterion 3 of the NRC Policy Statement.

LCO

[Two] ADV lines are required to be OPERABLE on each steam generator to ensure that at least one ADV is OPERABLE to conduct a unit 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 unit to SDC System entry conditions following an event

(continued)

BASES

LCO (continued)	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.
APPLICABILITY	In MODES 1, 2, and 3, [and in MODE 4, when steam generator is being relied upon for heat removal,] the ADVs are required to be OPERABLE. In MODES 5 and 6, an SGTR is not a credible event.

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that LCO 3.0.4 does not apply.

With one required ADV line inoperable, action must be taken to restore the OPERABLE status within 7 days. The 7 day Completion Time takes into account the redundant capability afforded by the remaining OPERABLE ADV lines, and a nonsafety grade backup in the Steam Bypass System and MSSVs.

B.1

With [two] or more [required] ADV lines inoperable, action must be taken to restore [one] 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 unit at power. The 24 hour Completion Time is reasonable to repair inoperable ADV lines, based on 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 unit must be placed in a MODE in which the LCO does not apply. To

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4, without reliance upon the steam generator for heat removal, within [12] hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.4.1

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 unit cooldown may satisfy this requirement. Operating experience has shown that these components usually pass the SR when performed at the [18] month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

[SR 3.7.4.2

The function of the ADV 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 unit cooldown may satisfy this requirement. Operating experience has shown that these components usually pass the SR when performed at the [18] month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.]

REFERENCES

1. FSAR, Section [10.3].

B 3.7 PLANT SYSTEMS

B 3.7.5 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 3.7.6, "Condensate Storage Tank (CST)") 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, "Main Steam Safety Valves (MSSVs)") or atmospheric dump valves (ADVs) (LCO 3.7.4, "Atmospheric Dump Valves (ADVs)"). 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.

One pump at full flow is sufficient to remove decay heat and cool the unit to Shutdown Cooling (SDC) System entry conditions.

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

(continued)

BASES

BACKGROUND (continued)

valves actuated to the appropriate steam generator by the Emergency Feedwater Actuation System (EFAS).

The AFW System supplies feedwater to the steam generators during normal unit 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 unit 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, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation." 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 the FSAR, Section [10.4.9] (Ref. 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 MSSV set pressure plus 3%.

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

a. Feedwater Line Break (FWLB); and

b. Loss of normal feedwater.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

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

The AFW System design is such that it can perform its function following an FWLB between the MFW 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 Policy Statement.

LCO

This LCO requires that [three] AFW trains be OPERABLE to ensure that the AFW System will perform the 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 two 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 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 not isolated by the closure of the MSIVs.

The AFW System is considered to be 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 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 also be OPERABLE.

(continued)

BASES

LCO (continued)

The LCO is modified by a Note indicating that only one AFW train, which includes a motor driven pump, is required to 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.

APPLICABILITY

In MODES 1, 2, and 3, the AFW System is required to be OPERABLE and to function in the event that the MFW 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, the AFW System may be used for heat removal via the steam generator.

In MODES 5 and 6, the steam generators are not normally used for decay heat removal, and the AFW System is not required.

ACTIONS

A.1

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

a. The redundant OPERABLE steam supply to the turbine driven 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.

The second Completion Time for Required Action A.1 establishes a limit on the maximum time allowed for any combination of Conditions to be inoperable during any continuous failure to meet this LCO.

(continued)

BASES

ACTIONS

A.1 (continued)

The 10 day Completion Time provides a limitation time allowed in this specified Condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which Conditions A and B are entered concurrently. The AND connector between 7 days and 10 days dictates that both Completion Times apply simultaneously, and the more restrictive must be met.

B.1

With one of the required AFW trains (pump or flow path) inoperable, action must be taken to restore OPERABLE status within 72 hours. This Condition includes the loss of two steam supply lines to the turbine driven AFW pump. The 72 hour Completion Time is reasonable, based on the redundant capabilities afforded by the AFW System, the time needed 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. The second Completion Time for Required Action B.1 establishes a limit on the maximum time allowed for any combination of Conditions to be inoperable during any continuous failure to meet this LCO.

The 10 day Completion Time provides a limitation time allowed in this specified Condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which Conditions A and B are entered concurrently. The AND connector between 72 hours and 10 days dictates that both Completion Times apply simultaneously, and the more restrictive must be met.

C.1 and C.2

When either Required Action A.1 or B.1 cannot be completed within the required Completion Time, [or if two AFW trains are inoperable in MODES 1, 2, and 3], the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within [18] hours.

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

In MODE 4, with [two AFW trains inoperable in MODES 1, 2, and 3], operation is allowed to continue because only one motor driven AFW pump is required in accordance with the Note that modifies the LCO. Although it is not required, the unit may continue to cool down and start the SDC.

D.1

Required Action D.1 is modified by a Note indicating that all required MODE changes or power reductions are suspended until one AFW train is restored to OPERABLE status.

With all [three] AFW trains inoperable in MODES 1, 2, and 3, the unit is in a seriously degraded condition with no safety related means for conducting a cooldown, and only limited means for conducting a cooldown with nonsafety grade equipment. In such a condition, the unit should not be perturbed by any action, including a power change, that might result in a trip. The seriousness of this condition requires that action be started immediately to restore one AFW train to OPERABLE status. LCO 3.0.3 is not applicable, as it could force the unit into a less safe condition.

E.1

Required Action E.1 is modified by a Note indicating that all required MODE changes or power reductions are suspended until one AFW train is restored to OPERABLE status.

With one AFW train inoperable, action must be taken to immediately restore the inoperable train to OPERABLE status or to immediately verify, by administrative means, the OPERABILITY of a second train. LCO 3.0.3 is not applicable, as it could force the unit into a less safe condition.

(continued)

BASES

ACTIONS

E.1 (continued)

In MODE 4, either the reactor coolant pumps or the SDC loops can be used to provide forced circulation as discussed in LCO 3.4.6, "RCS Loops—MODE 4."

SURVEILLANCE REQUIREMENTS

SR 3.7.5.1

Verifying the correct alignment for manual, power operated, and automatic valves in the AFW water and steam supply 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 are verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This Surveillance does not require any testing or valve manipulations; rather, it involves verification that those valves capable of potentially being mispositioned are in the correct position.

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

SR 3.7.5.2

Verifying that each AFW pump's developed head at the flow test point is greater than or equal to this required developed head ensures that AFW pump performance has not degraded during the cycle. Flow and differential head are normal tests of pump performance required by Section XI of the ASME Code (Ref.2). Because it is undesirable to introduce cold AFW into the steam generators while they are operating, this testing is performed on recirculation flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice tests confirm component OPERABILITY, tread performance, and detect incipient failures by indicating abnormal performance. Performance of inservice testing, discussed in the ASME Code, Section XI (Ref. 2), at 3 month intervals satisfies this requirement. The [31] day Frequency on a STAGGERED

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.5.2 (continued)

TEST BASIS results in testing each pump once every 3 months, as required by Reference 2.

This SR is modified by a Note indicating that the SR should be deferred until suitable test conditions are established. This deferral is required because there is an insufficient steam pressure to perform the test.

SR 3.7.5.3

This SR ensures that AFW can be delivered to the appropriate steam generator, in the event of any accident or transient that generates an EFAS signal, by demonstrating that each automatic valve in the flow path actuates to its correct position on an actual or simulated actuation signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The 18 month Frequency is acceptable, based on the design reliability and operating experience of the equipment.

This SR is modified by a Note indicating that the SR should be deferred until suitable test conditions have been established. This deferral is required because there is an insufficient steam pressure to perform the test.

Also, this SR is modified by a Note that states the SR is not required in MODE 4. In MODE 4, the required AFW train is already aligned and operating.

SR 3.7.5.4

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 [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.5.4 (continued)

potential for an unplanned transient if the Surveillance were performed with the reactor at power. The 18 month Frequency is acceptable, based on the design reliability and operating experience of the equipment.

This SR is modified by [a] [two] Note[s]. [Note 1 indicates that the SR be deferred until suitable test conditions are established. This deferral is required because there is insufficient steam pressure to perform the test.] [The] Note [2] states that the SR is not required in MODE 4. [In MODE 4, the required pump is already operating and the autostart function is not required.] [In MODE 4, the heat removal requirements would be less providing more time for operator action to manually start the required AFW pump.]

[Reviewer's Note: Some plants may not routinely use the AFW for heat removal in MODE 4. The second justification is provided for plants that use a startup feedwater pump rather than AFW for startup and shutdown.]

SR 3.7.5.5

This SR ensures that the AFW System is properly aligned by verifying 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 verified before sufficient core heat is generated that would require the operation of the AFW System during a subsequent shutdown. The Frequency is reasonable, based on engineering judgment, and other administrative controls to ensure that flow paths remain OPERABLE. To further ensure AFW System alignment, the OPERABILITY of the flow paths is verified following extended outages to determine that no misalignment of valves has occurred. 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 units that use AFW for normal startup and shutdown.)

(continued)

BASES (continued)

REFERENCES

1. FSAR, Section [10.4.9].
2. ASME, Boiler and Pressure Vessel Code, Section XI, Inservice Inspection, Article IWV-3400.

B 3.7 PLANT SYSTEMS

B 3.7.6 Condensate Storage Tank (CST)

BASES

BACKGROUND

The CST provides a safety grade source of water to the steam generators for removing decay and sensible heat from the Reactor Coolant System (RCS). The CST provides a passive flow of water, by gravity, to the Auxiliary Feedwater (AFW) System (LCO 3.7.4, "Auxiliary Feedwater (AFW) System"). The steam produced is released to the atmosphere by the main steam safety valves (MSSVs) 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 nonsafety 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.

A description of the CST is found in the FSAR, Section [9.2.6] (Ref. 1).

APPLICABLE SAFETY ANALYSES

The CST provides cooling water to remove decay heat and to cool down the unit following all events in the accident analysis, discussed in the FSAR, Chapters [6] and [15] (Refs. 2 and 3, respectively). 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 (SDC) entry conditions at the design cooldown rate.

The limiting event for the condensate volume is the large feedwater line break with a coincident loss of offsite

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

power. Single failures that also affect this event include the following:

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 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 nonlimiting 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 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 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% RTP, and then cool down the RCS to SDC 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 turbine, or before isolating AFW to a broken line.

The CST level required is a usable volume of ≤ [350,000] gallons, which is based on holding the unit in MODE 3 for [4] hours, followed by a cooldown to SDC entry conditions at 75°F per hour. This basis is established by the NRC Standard Review Plan Branch Technical Position, Reactor

(continued)

BASES

LCO (continued)

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.

APPLICABILITY

In MODES 1, 2, and 3, [and in MODE 4, when steam generator is being relied upon for heat removal,] the CST is required to be OPERABLE.

In MODES 5 and 6, the CST is not required because the AFW System is not required.

ACTIONS

A.1 and A.2

If the CST level is not within the limit, the OPERABILITY of the backup water supply must be verified by administrative means within 4 hours.

OPERABILITY of the backup feedwater supply must 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 level must be returned to OPERABLE status within 7 days, as the backup supply may be performing this function in addition to its normal functions. The 4 hour Completion Time is reasonable, based on operating experience, to verify the OPERABILITY of the backup water supply. The 7 day Completion Time is reasonable, based on an OPERABLE backup water supply being available, and the low probability of an event requiring the use of the water from the CST occurring during this period.

B.1 and B.2

If the CST cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4, without reliance on steam generator for heat removal, within [18] hours. The allowed Completion

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

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

SURVEILLANCE REQUIREMENTS

SR 3.7.6.1

This SR verifies that the CST contains the required volume of cooling water. (This level ≥ [350,000] gallons.) The 12 hour Frequency is based on operating experience, and the need for operator awareness of unit evolutions that may affect the CST inventory between checks. The 12 hour Frequency is considered adequate in view of other indications in the control room, including alarms, to alert the operator to abnormal CST level deviations.

REFERENCES

1. FSAR, Section [9.2.6].
2. FSAR, Chapter [6].
3. FSAR, Chapter [15].
4. NRC Standard Review Plan Branch Technical Position (BTP) RSB 5-1.

B 3.7 PLANT SYSTEMS

B 3.7.7 Component Cooling Water (CCW) System

BASES

BACKGROUND

The CCW System provides a heat sink for the removal of process and operating heat from safety related components during a Design Basis Accident (DBA) or transient. During normal operation, the CCW System also provides this function for various nonessential components, as well as the spent fuel pool. The CCW System serves as a barrier to the release of radioactive byproducts between potentially radioactive systems and the Service Water System, and thus to the environment.

The 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.

Additional information on the design and operation of the system, along with a list of the components served, is presented in the FSAR, Section [9.2.2], Reference 1. The principal safety related function of the CCW System is the removal of decay heat from the reactor via the Shutdown Cooling (SDC) System 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

(continued)

BASES

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 unit from SDC 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 SDC trains operating. One CCW 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 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 DBA, one CCW train 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 CCW trains must be OPERABLE. At least one CCW train will operate assuming the worst single active failure occurs coincident with the loss of offsite power.

A CCW train is considered OPERABLE when the following:

a. The associated pump and surge tank are OPERABLE; and

b. The associated piping, valves, heat exchanger and instrumentation and controls required to perform the safety related function are OPERABLE.

The isolation of CCW from other components or systems not required for safety may render those components or systems inoperable, but does not affect the OPERABILITY of the CCW System.

(continued)

BASES (continued)

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 SDC heat exchanger.

In MODES 5 and 6, the OPERABILITY requirements of the CCW System are determined by the systems it supports.

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating the requirement of entry into the applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops—MODE 4," for SDC made inoperable by CCW. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components.

With one CCW train inoperable, action must be taken to restore OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE CCW train is adequate to perform the heat removal function. The 72 hour Completion Time is based on the redundant capabilities afforded by the OPERABLE train, and the low probability of a DBA occurring during this period.

B.1 and B.2

If the CCW train cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

(continued)

BASES (continued)

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 are verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This Surveillance does not require any testing or valve manipulation; rather, it involves verification that those valves capable of potentially being mispositioned are in their correct position.

This SR is modified by a Note indicating that the isolation of the CCW components or systems may render those components inoperable but does not affect the OPERABILITY of the CCW System.

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

SR 3.7.7.2

This SR verifies proper automatic operation of the CCW valves on an actual or simulated actuation signal. The CCW System is a normally operating system that cannot be fully actuated as part of routine testing during normal operation. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the [18] month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.7.3

This SR verifies proper automatic operation of the CCW pumps on an actual or simulated actuation signal. The CCW System is a normally operating system that cannot be fully actuated as part of routine testing during normal operation. The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the [18] month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

REFERENCES

1. FSAR, Section [9.2.2].

B 3.7 PLANT SYSTEMS

B 3.7.8 Service Water System (SWS)

BASES

BACKGROUND

The SWS provides a heat sink for the removal of process and operating heat from safety related components during a Design Basis Accident (DBA) or transient. During normal operation or a normal shutdown, the SWS also provides this function for various safety related and nonsafety related components. The safety related function is covered by this LCO.

The 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 and all essential valves are aligned to their post accident positions. The SWS also provides emergency makeup to the spent fuel pool and CCW System [and is the backup water supply to the Auxiliary Feedwater System].

Additional information about the design and operation of the SWS, along with a list of the components served, is presented in the FSAR, Section [9.2.1] (Ref. 1). The principal safety related function of the SWS is the removal of decay heat from the reactor via the [CCW System].

APPLICABLE SAFETY ANALYSES

The design basis of the SWS is for one SWS train, in conjunction with the CCW System and a 100% capacity containment cooling system (containment spray, containment coolers, or a combination), removing core decay heat 20 minutes following a design basis LOCA, as discussed in the FSAR, Section [6.2] (Ref. 2). This prevents the containment sump fluid from increasing in temperature during the recirculation phase following a LOCA and provides for a gradual reduction in the temperature of this fluid as it is supplied to the Reactor Coolant System by the safety injection pumps. The SWS is designed to perform its

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

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 unit from shutdown cooling (SDC), as discussed in the FSAR, Section [5.4.7] (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 SDC 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 Policy Statement.

LCO

Two SWS trains are required to be OPERABLE to provide the required redundancy to ensure that the system functions to remove post accident heat loads, assuming the worst single active failure occurs coincident with the loss of offsite power.

An SWS train is considered OPERABLE when:

a. The associated pump is OPERABLE; and

b. The associated piping, valves, instrumentation, heat exchanger, and instrumentation and controls required to perform the safety related function are OPERABLE.

APPLICABILITY

In MODES 1, 2, 3, and 4, the SWS System is a normally operating system, which is required to support the OPERABILITY of the equipment serviced by the SWS and required to be OPERABLE in these MODES.

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

(continued)

BASES (continued)

ACTIONS

A.1

With one SSW train inoperable, action must be taken to restore OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE SWS train is adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure in the SWS train could result in loss of SWS function. Required Action A.1 is modified by two Notes. The first Note indicates that the applicable Conditions of LCO 3.8.1, "AC Sources—Operating," should be entered if the inoperable SWS train results in an inoperable emergency diesel generator. The second Note indicates that the applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops—MODE 4," should be entered if an inoperable SWS train results in an inoperable SDC. The 72 hour Completion Time is based on the redundant capabilities afforded by the OPERABLE train, and the low probability of a DBA occurring during this time period.

B.1 and B.2

If the SWS train cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, in MODE 4 within [12] hours, and in MODE 5 within 36 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.8.1

Verifying the correct alignment for manual, power operated, and automatic valves in the SWS flow path ensures that the proper flow paths exist for SWS operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since they are verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.8.1 (continued)

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. This SR is modified by a Note indicating that the isolation of the SWS components or systems may render those components inoperable but does not affect the OPERABILITY of the SWS.

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

SR 3.7.8.2

This SR verifies proper automatic operation of the SWS valves on an actual or simulated actuation signal. The SWS is a normally operating system that cannot be fully actuated as part of the normal testing. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the [18] month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

SR 3.7.8.3

The SR verifies proper automatic operation of the SWS pumps on an actual or simulated actuation signal. The SWS is a normally operating system that cannot be fully actuated as part of the normal testing during normal operation. The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the [18] month

(continued)

BASES

SURVEILLANCE REQUIREMENTS	SR 3.7.8.3 (continued) Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.
REFERENCES	1. FSAR, Section [9.2.1]. 2. FSAR, Section [6.2]. 3. FSAR, Section [5.4.7].

B 3.7 PLANT SYSTEMS

B 3.7.9 Ultimate Heat Sink (UHS)

BASES

BACKGROUND

The UHS provides a heat sink for process and operating heat from safety related components during a Design Basis Accident (DBA) or transient, as well as during normal operation. This is done utilizing the Service Water System.

The UHS has been defined as that complex of water sources, including necessary retaining structures (e.g., a pond with its dam, or a river with its dam), and the canals or conduits connecting the sources with, but not including, the cooling water system intake structures as, discussed in the FSAR, Section [9.2.5] (Ref. 1). If cooling towers or portions thereof are required to accomplish the UHS safety functions, they should meet the same requirements as the sink. The two principal functions of the UHS are the dissipation of residual heat after reactor shutdown, and dissipation of residual heat after an accident.

A variety of complexes is used to meet the requirements for a UHS. A lake or an ocean may qualify as a single source. If the complex includes a water source contained by a structure, it is likely that a second source will be required.

The basic performance requirements are that a 30 day supply of water be available, and that the design basis temperatures of safety related equipment not be exceeded. Basins of cooling towers generally include less than a 30 day supply of water, typically 7 days or less. A 30 day supply would be dependent on another source(s) and a makeup system(s) for replenishing the source in the cooling tower basin. For smaller basin sources, which may be as small as a 1 day supply, the systems for replenishing the basin and the backup source(s) become of sufficient importance that the makeup system itself may be required to meet the same design criteria as an Engineered Safety Feature (e.g., single failure considerations, and multiple makeup water sources may be required).

It follows that the many variations in the UHS configurations will result in many unit to unit variations in OPERABILITY determinations and SRs. The ACTIONS and SRs

(continued)

BASES

BACKGROUND (continued)	are illustrative of a cooling tower UHS without a makeup requirement. Additional information on the design and operation of the system along with a list of components served can be found in Reference 1.
APPLICABLE SAFETY ANALYSES	The UHS is the sink for heat removed from the reactor core following all accidents and anticipated operational occurrences in which the unit is cooled down and placed on shutdown cooling. For those units using it as the normal heat sink for condenser cooling via the Circulating Water System, unit 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 unit switches from injection to recirculation, and the containment cooling systems are required to remove the core decay heat. The operating limits are based on conservative heat transfer analyses for the worst case LOCA. Reference 1 provides the details of the assumptions used in the analysis. The assumptions include: worst expected meteorological conditions, conservative uncertainties when calculating decay heat, and the worst case failure (e.g., single failure of a manmade structure). The UHS is designed in accordance with Regulatory Guide 1.27 (Ref. 2), which requires a 30 day supply of cooling water in the UHS. The UHS satisfies Criterion 3 of the NRC Policy Statement.
LCO	The UHS is required to be OPERABLE. 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 unit operation.

(continued)

BASES (continued)

APPLICABILITY

In MODES 1, 2, 3, and 4, the UHS is a normally operating system that is required to support the OPERABILITY of the equipment serviced by the UHS and required to be OPERABLE in these MODES.

In MODES 5 and 6, the OPERABILITY requirements of the UHS are determined by the systems it supports.

ACTIONS

A.1

If one or more cooling towers have one fan inoperable (i.e., up to one fan per cooling tower inoperable), action must be taken to restored the inoperable cooling tower fan(s) to OPERABLE status within 7 days.

The 7 day Completion Time is reasonable, based on the low probability of an accident occurring during the 7 days that one cooling tower fan is inoperable, the number of available systems, and the time required to complete the action.

B.1 and B.2

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

SURVEILLANCE REQUIREMENTS

SR 3.7.9.1

This SR verifies adequate long term (30 days) cooling can be maintained. The level specified also ensures sufficient NPSH is available for operating the SWS pumps. The 24 hour Frequency is based on operating experience related to the

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.9.1 (continued)

[trending of the parameter variations during the applicable MODES. This SR verifies that the UHS water level is ≥ [562] ft [mean sea level].]

[SR 3.7.9.2

This SR verifies that the SWS is available to 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. This SR verifies that the UHS water temperature is ≤ [92]°F.]

[SR 3.7.9.3

Operating each cooling tower fan for ≥ [15] minutes verifies that all fans are OPERABLE and that all associated controls are functioning properly. It also ensures that fan or motor failure, or excessive vibration can be detected for corrective action. The 31 day Frequency is based on operating experience, 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.]

REFERENCES

1. FSAR, Section [9.2.5].
2. Regulatory Guide 1.27.

B 3.7 PLANT SYSTEMS

B 3.7.10 Essential Chilled Water (ECW) System

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 Design Basis Accident (DBA) or transient.

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 Engineered 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 nonsafety 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 the FSAR, Section [9.2.9] (Ref. 1).

APPLICABLE SAFETY ANALYSES

The design basis of the ECW System is to remove the post accident heat load from ESF spaces following a DBA 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. During recirculation, hot fluid from the containment sump is supplied to the high pressure safety

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

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 satisfies Criterion 3 of the NRC Policy Statement.

LCO

[Two] ECW trains are required to be OPERABLE to provide the required redundancy to ensure that the system functions to remove post accident heat loads, assuming the worst single failure.

An ECW train is considered OPERABLE when:

a. The associated pump and surge tank are OPERABLE; and

b. The associated piping, valves, heat exchanger, refrigeration unit, and instrumentation and controls required to perform the safety related function 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.

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.

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, action must be taken to restore OPERABLE status within 7 days. In this condition, one OPERABLE ECW train is adequate to perform the cooling function. The 7 day Completion Time is reasonable, based on the low probability of an event occurring during this time, the 100% capacity OPERABLE ECW train, and the redundant availability of the normal HVAC System.

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

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

SURVEILLANCE REQUIREMENTS

SR 3.7.10.1

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 are verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This Surveillance does not require any testing or valve manipulation; rather, it involves verification that those valves capable of potentially being mispositioned are in the correct position.

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

SR 3.7.10.2

This SR verifies proper automatic operation of the ECW System components that the ECW pumps will start in the event of any accident or transient that generates an SIAS. This SR also ensures that each automatic valve in the flow paths actuates to its correct position on an actual or simulated SIAS. 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 test every 92 days, except for the subgroup relays that actuate the system that cannot be tested during normal unit

(continued)

BASES

SURVEILLANCE REQUIREMENTS	SR 3.7.10.2 (continued) operation. The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The [18] month Frequency is based on operating experience and design reliability of the equipment.
REFERENCES	1. FSAR, Section [9.2.9].

B 3.7 PLANT SYSTEMS

B 3.7.11 Control Room Emergency Air Cleanup System (CREACS)

BASES

BACKGROUND

The CREACS provides a protected environment from which operators can control the unit 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 and 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 unit 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 filter trains of the system. The prefilters and 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.

(continued)

BASES

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 the FSAR, Section [9.4] (Ref. 1).

Redundant supply and recirculation trains provide the required filtration should an excessive pressure drop develop across the other filter train. Normally open isolation dampers are arranged in series pairs so that the failure of one damper to shut will not result in a breach of isolation. The 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 or its equivalent to any part of the body.

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.

The CREACS provides airborne radiological protection for the control room operators, as demonstrated by the control room

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

accident dose analyses for the most limiting design basis loss of coolant accident fission product release presented in the FSAR, Chapter [15] (Ref. 2).

The 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 Policy Statement.

LCO

Two independent and redundant trains of the CREACS are required to be OPERABLE 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 CREACS train is considered OPERABLE when the associated:

a. Fan is OPERABLE;

b. HEPA filters and charcoal adsorber are not excessively restricting flow, and are capable of performing their filtration functions; and

c. Heater, demister, ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained.

In addition, the control room boundary must be maintained, including the integrity of the walls, floors, ceilings, ductwork, and access doors.

(continued)

BASES (continued)

APPLICABILITY

In MODES 1, 2, 3, and 4, the CREACS must be OPERABLE to limit operator exposure during and following a DBA.

In MODES [5 and 6], the CREACS is required to cope with the release from a rupture of an outside waste gas tank.

During movement of irradiated fuel assemblies [and CORE ALTERATIONS], the CREACS must be OPERABLE to cope with the release from a fuel handling accident.

ACTIONS

A.1

With one CREACS train inoperable, action must be taken to restore OPERABLE status within 7 days. In this Condition, the remaining OPERABLE CREACS subsystem is adequate to perform control room radiation protection function. However, the overall reliability is reduced because a single failure in the OPERABLE CREACS train could result in loss of CREACS 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.

B.1 and B.2

If the inoperable CREACS cannot be restored to OPERABLE status within the required Completion Time in MODE 1, 2, 3, or 4, the unit must be placed in a MODE that minimizes the accident risk. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

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

Required Action C.1 is modified by a Note indicating to place the system in the emergency radiation protection mode if the automatic transfer to emergency mode is inoperable.

(continued)

BASES

ACTIONS

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

In MODE 5 or 6, or during movement of irradiated fuel assemblies [, or during CORE ALTERATIONS], if Required Action A.1 cannot be completed within the required Completion Time, the OPERABLE CREACS train must be immediately placed in the emergency mode of operation. This action ensures that the remaining train is OPERABLE, that no failures preventing automatic actuation will occur, and that any active failure will be readily detected.

An alternative to Required Action C.1 is to immediately suspend activities that could result in a release of radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes the accident risk. This does not preclude the movement of fuel assemblies to a safe position.

D.1 and D.2

When [in MODES 5 and 6, or] during movement of irradiated fuel assemblies [, or during CORE ALTERATIONS], with two CREACS trains inoperable, action must be taken immediately to suspend activities that could result in a release of radioactivity that might enter the control room. This places the unit in a condition that minimizes the accident risk. This does not preclude the movement of fuel to a safe position.

E.1

If both CREACS trains are inoperable in MODE 1, 2, 3, or 4, the CREACS may not be capable of performing the intended function and the unit is in a condition outside the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.7.11.1

Standby systems should be checked periodically to ensure that they function properly. Since the environment and normal operating conditions on this system are not severe,

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.7.11.1 (continued)

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 with heaters must be operated for ≥ 10 continuous hours with the heaters energized. Systems without heaters need only be operated for ≥ 15 minutes to demonstrate the function of the system.] The 31 day Frequency is based on the known reliability of the equipment, and the two train redundancy available.

SR 3.7.11.2

This SR verifies that the required CREACS testing is performed in accordance with the [Ventilation Filter Testing Program (VFTP)]. The CREACS filter tests are in accordance with Regulatory Guide 1.52 (Ref. 3). The [VFTP] includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the [VFTP].

SR 3.7.11.3

This SR verifies each CREACS train starts and operates on an actual or simulated actuation signal. The Frequency of [18] months is consistent with that specified in Reference 3.

SR 3.7.11.4

This SR verifies the integrity of the control room enclosure and the assumed inleakage 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

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.11.4 (continued)

operation, the CREACS is designed to pressurize the control room ≥ [0.125] inches water gauge positive pressure with respect to adjacent areas in order to prevent unfiltered inleakage. The CREACS is designed to maintain this positive pressure with one train at an emergency ventilation flow rate of [3000] cfm. The Frequency of [18] months on a STAGGERED TEST BASIS is consistent with the guidance provided in NUREG-0800, Section 6.4 (Ref. 4).

REFERENCES

1. FSAR, Section [9.4].
2. FSAR, Chapter [15].
3. Regulatory Guide 1.52 (Rev. 2).
4. NUREG-0800, Section 6.4, Rev. 2, July 1981.

B 3.7 PLANT SYSTEMS

B 3.7.12 Control Room Emergency Air Temperature Control System (CREATCS)

BASES

BACKGROUND

The CREATCS provides temperature control for the control room following isolation of the control room.

The CREATCS consists of two independent, 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 CREATCS is a subsystem providing air temperature control for the control room.

The CREATCS is an emergency system, parts of which may also operate during normal unit operations. A single train will provide the required temperature control to maintain the control room between [70]°F and [85]°F. The CREATCS operation to maintain the control room temperature is discussed in the FSAR, Section [6.4] (Ref. 1).

APPLICABLE SAFETY ANALYSES

The design basis of the CREATCS is to maintain temperature of the control room environment throughout 30 days of continuous occupancy.

The CREATCS components are arranged in redundant safety related trains. During emergency operation, the CREATCS maintains the temperature between [70]°F and [85]°F. A single active failure of a component of the CREATCS, 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 CREATCS is designed in accordance with Seismic Category I requirements. The CREATCS is capable of removing sensible and latent heat loads from the control room, considering equipment heat loads and personnel occupancy requirements, to ensure equipment OPERABILITY.

The CREATCS satisfies Criterion 3 of the NRC Policy Statement.

(continued)

BASES (continued)

LCO

Two independent and redundant trains of the CREATCS are required to be OPERABLE to ensure that at least one is available, assuming a single failure disables the other train. Total system failure could result in the equipment operating temperature exceeding limits in the event of an accident.

The CREATCS is considered OPERABLE when the individual components that are necessary to maintain the control room temperature are OPERABLE in both trains. These components include the cooling coils and associated temperature control instrumentation. In addition, the CREATCS must be OPERABLE to the extent that air circulation can be maintained.

APPLICABILITY

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

In MODES 5 and 6, CREATCS may not be required for those facilities which do not require automatic control room isolation.

ACTIONS

A.1

With one CREATCS train inoperable, action must be taken to restore OPERABLE status within 30 days. In this Condition, the remaining OPERABLE CREATCS train is adequate to maintain the control room temperature within limits. The 30 day Completion Time is reasonable, based on the low probability of an event occurring requiring control room isolation, consideration that the remaining train can provide the required capabilities, and the alternate safety or nonsafety related cooling means that are available.

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 unit must be placed in a MODE that minimizes the accident risk. To

(continued)

ACTIONS

B.1 and B.2 (continued)

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

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

In MODE 5 or 6, or during movement of irradiated fuel assemblies [, or during CORE ALTERATIONS], when Required Action A.1 cannot be completed within the required Completion Time, the OPERABLE CREATCS train must be placed in operation immediately. This action ensures that the remaining train is OPERABLE, that no failures preventing automatic actuation will occur, and that any active failure will be readily detected.

An alternative to Required Action C.1 is to immediately suspend activities that could result in a release of radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes the accident risk. This does not preclude the movement of fuel assemblies to a safe position.

D.1 and D.2

In [MODE 5 or 6, or] during movement of irradiated fuel assemblies [, or during CORE ALTERATIONS], with two CREATCS trains inoperable, action must be taken immediately to suspend activities that could result in a release of radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes the accident risk. This does not preclude the movement of fuel to a safe position.

E.1

If both CREATCS trains are inoperable in MODE 1, 2, 3, or 4, the CREATCS may not be capable of performing the intended

(continued)

BASES

ACTIONS	E.1 (continued) function and the unit is in a condition outside the accident analysis. Therefore, LCO 3.0.3 must be entered immediately.
SURVEILLANCE REQUIREMENTS	SR 3.7.12.1 This SR verifies that the heat removal capability of the system is sufficient to meet design requirements. This SR consists of a combination of testing and calculations. An [18] month Frequency is appropriate, since significant degradation of the CREATCS is slow and is not expected over this time period.
REFERENCES	1. FSAR, Section [6.4].

B 3.7 PLANT SYSTEMS

B 3.7.13 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 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 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 unit 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 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.

(continued)

BASES

BACKGROUND (continued)

The ECCS PREACS is discussed in the FSAR, Sections [6.5.1], [9.4.5], and [15.6.5] (Refs. 1, 2, and 3, respectively), 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 at an acceptable level consistent with iodine removal efficiencies, as discussed in the 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 unit to go into the recirculation mode of long term cooling and to clean up releases of smaller leaks, such as from valve stem packing.

The two types of system failures that are considered in the accident analysis are 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 Policy Statement.

LCO

Two independent and redundant ECCS PREACS trains are required to be OPERABLE 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 required limits in the event of a Design Basis Accident (DBA).

(continued)

BASES

LCO (continued)

ECCS PREACS is considered OPERABLE when the individual components necessary to maintain the ECCS Pump Room filtration are OPERABLE in both trains.

An ECCS PREACS train is considered OPERABLE when its associated:

a. Fan is OPERABLE;

b. HEPA filter and charcoal adsorber are not excessively restricting flow and are capable of performing their filtration functions; and

c. Heater, demister, ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained.

APPLICABILITY

In MODES 1, 2, 3, and 4, the 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 inoperable, action must be taken to restore OPERABLE status within 7 days. During this time, the remaining OPERABLE train is adequate to perform the ECCS PREACS function.

The 7 day Completion Time is appropriate because the risk contribution is less than that for the ECCS (72 hour Completion Time) and this system is not a direct support system for the ECCS. The 7 day Completion Time is reasonable, based on the low probability of a DBA occurring during this time period, and the consideration that the remaining train can provide the required capability.

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

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

SURVEILLANCE REQUIREMENTS

SR 3.7.13.1

Standby systems should be checked periodically to ensure that they function properly. Since the environment and normal operating conditions on this system are not severe, testing each train once a month provides an adequate check on this system. Monthly heater operations dry out any moisture that may have accumulated in the charcoal from humidity in the ambient air. [Systems with heaters must be operated for ≥ 10 continuous hours with the heaters energized. Systems without heaters need only be operated for ≥ 15 minutes to demonstrate the function of the system.] The 31 day Frequency is based on the known reliability of equipment, and the two train redundancy available.

SR 3.7.13.2

This SR verifies that the required ECCS PREACS testing is performed in accordance with the [Ventilation Filter Testing Program (VFTP)]. The ECCS PREACS filter tests are in accordance with Regulatory Guide 1.52 (Ref. 4). The [VFTP] includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the [VFTP].

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.13.3

This SR verifies 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 3.7.13.4

This SR verifies 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 post accident 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 the NUREG-0800, Section 6.5.1 (Ref. 6).

This test is conducted with the tests for filter penetration; thus, an [18] month Frequency, on a STAGGERED TEST BASIS is consistent with other filtration SRs.

[SR 3.7.13.5

Operating the ECCS PREACS filter bypass damper is necessary to ensure that the system functions properly. The OPERABILITY of the bypass damper is verified if it can be closed. An [18] month Frequency is consistent with that specified in Reference 4.]

REFERENCES

1. FSAR, Section [6.5.1].
2. FSAR, Section [9.4.5].
3. FSAR, Section [15.6.5].
4. Regulatory Guide 1.52 (Rev. 2).

(continued)

BASES

REFERENCES (continued)	5.	10 CFR 100.11.
	6.	NUREG-0800, Section 6.5.1, Rev. 2, July 1981.

B 3.7 PLANT SYSTEMS

B 3.7.14 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. 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 unit 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 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 the FSAR, Sections [6.5.1], [9.4.5], and [15.7.4] (Refs. 1, 2, and 3, respectively), because it may be used for normal, as well as post accident, atmospheric cleanup functions.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The FBACS is designed to mitigate the consequences of a fuel handling accident in which [all] rods in the fuel assembly are assumed to be damaged. The analysis of the fuel handling accident is given in Reference 3. The Design Basis Accident 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. These assumptions and the analysis follow the guidance provided in Regulatory Guide 1.25 (Ref. 4).

The FBACS satisfies Criterion 3 of the NRC Policy Statement.

LCO

Two independent and redundant trains of the FBACS are required to be OPERABLE to ensure that at least one is available, assuming a single failure that disables the other train coincident with a loss of offsite power. Total system failure could result in the atmospheric release from the fuel building exceeding the 10 CFR 100 limits (Ref. 5) 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. An FBACS train is considered OPERABLE when its associated:

a. Fan is OPERABLE;

b. HEPA filter and charcoal adsorber are not excessively restricting flow, and are capable of performing their filtration functions; and

c. Heater, demister, ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained.

APPLICABILITY

In MODES 1, 2, 3, and 4, the FBACS is required to be OPERABLE to provide fission product removal associated with ECCS leaks due to a LOCA (refer to LCO 3.7.13, "Emergency

(continued)

BASES

APPLICABILITY (continued)

Core Cooling System (ECCS) Pump Room Exhaust Air Cleanup System (PREACS)") for units that use this system as part of their ECCS PREACS.

During movement of irradiated fuel assemblies in the fuel building, the FBACS is required to be OPERABLE to mitigate the consequences of a fuel handling accident.

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

ACTIONS

A.1

If one FBACS train is inoperable, action must be taken to restore OPERABLE status within 7 days. During this time period, the remaining OPERABLE train is adequate to perform the FBACS function. The 7 day Completion Time is reasonable, based on the risk from an event occurring requiring the inoperable FBACS train, and ability of the remaining FBACS train to provide the required protection.

B.1 and B.2

In MODE 1, 2, 3, or 4, when Required Action A.1 cannot be completed within the Completion Time, or when both FBACS trains are inoperable, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

C.1 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 must be started immediately or fuel movement suspended. This action ensures that the remaining train is OPERABLE, that no undetected failures preventing system operation will occur, and that any active failure will be readily detected.

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

If the system is not placed in operation, this action requires suspension of fuel movement, which precludes a fuel handling accident. This does not preclude the movement of fuel to a safe position.

D.1

When two trains of the FBACS are inoperable during movement of irradiated fuel assemblies in the fuel building, action must be taken to place the unit in a condition in which the LCO does not apply. This LCO involves immediately suspending movement of irradiated fuel assemblies in the fuel building. This does not preclude the movement of fuel to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.7.14.1

Standby systems should be checked periodically to ensure that they function properly. As the environment and normal operating conditions on this system are not severe, testing each train once every month provides an adequate check on this system. Monthly heater operation dries out any moisture accumulated in the charcoal from humidity in the ambient air. [Systems with heaters must be operated for ≥ 10 continuous hours with the heaters energized. Systems without heaters need only be operated for ≥ 15 minutes to demonstrate the function of the system.] The 31 day Frequency is based on the known reliability of the equipment and the two train redundancy available.

SR 3.7.14.2

This SR verifies the performance of FBACS filter testing in accordance with the [Ventilation Filter Testing Program (VFTP)]. The FBACS filter tests are in accordance with the Regulatory Guide 1.52 (Ref. 6). The [VFTP] includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.14.2 (continued)

operations). Specific test frequencies and additional information are discussed in detail in the [VFTP].

SR 3.7.14.3

This SR verifies that each FBACS train starts and operates on an actual or simulated actuation signal. The [18] month Frequency is consistent with that specified in Reference 6.

SR 3.7.14.4

This SR verifies 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 post accident 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 ≤ [3000] cfm to the fuel building. The Frequency of [18] months is consistent with the guidance provided in NUREG-0800, Section 6.5.1 (Ref. 7).

This test is conducted with the tests for filter penetration; thus, an [18] month Frequency, on a STAGGERED TEST BASIS is consistent with other filtration SRs.

SR 3.7.14.5

Operating the FBACS filter bypass damper is necessary to ensure that the system functions properly. The OPERABILITY of the FBACS filter bypass damper is verified if it can be closed. The 18 month Frequency is consistent with that specified in Reference 6.

(continued)

BASES (continued)

REFERENCES

1. FSAR, Section [6.5.1].
2. FSAR, Section [9.4.5].
3. FSAR, Section [15.7.4].
4. Regulatory Guide 1.25.
5. 10 CFR 100.
6. Regulatory Guide 1.52.
7. NUREG-0800, Section 6.5.1, July 1981.

B 3.7 PLANT SYSTEMS

B 3.7.15 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 of the main HEPA filter bank. The downstream HEPA filter, although not credited in the accident analysis, collects charcoal fines and serves as a backup should the upstream HEPA filter develop a leak. The system initiates filtered ventilation following receipt of a safety injection actuation signal or containment isolation actuation signal.

The PREACS is a standby system, parts of which may also operate during normal unit 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 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 the FSAR, Sections [6.5.1], [9.4.5], and [15.6.5] (Refs. 1, 2, and 3, respectively), 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,

(continued)

BASES

BACKGROUND (continued)

consistent with iodine removal efficiencies, as discussed in the Regulatory Guide 1.52 (Ref. 4).

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.

There are two types of system failures considered in the accident analysis: a complete loss of function and 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 Policy Statement.

LCO

Two independent and redundant trains of the PREACS are required to be OPERABLE to ensure that at least one train is available, assuming there is a single failure disabling the other train coincident with a loss of offsite power.

The PREACS is considered OPERABLE when the individual components necessary to control radioactive releases are OPERABLE in both trains. A PREACS 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 the filtration functions; and

c. Heater, demister, ductwork, valves, and dampers are OPERABLE, and circulation can be maintained.

(continued)

BASES (continued)

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, action must be taken to restore OPERABLE status within 7 days. During this time 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 hour Completion Time), and because this system is not a direct support system for the ECCS. The 7 day Completion Time is reasonable, is based on the low probability of a DBA occurring during this time period, and the consideration that the remaining train can provide the required capability.

B.1 and B.2

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

SURVEILLANCE REQUIREMENTS

SR 3.7.15.1

Standby systems should be checked periodically to ensure that they function properly. As the environment and normal operating conditions on this system are not severe, testing each train once every month provides an adequate check on this system.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.15.1 (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 with heaters must be operated for ≥ 10 continuous hours with the heaters energized. Systems without heaters need only be operated for ≥ 15 minutes to demonstrate the function of the system.] The 31 day Frequency is based on the known reliability of the equipment and the two train redundancy available.

SR 3.7.15.2

This SR verifies the performance of PREACS filter testing in accordance with the [Ventilation Filter Testing Program (VFTP)]. The PREACS filter tests are in accordance with Reference 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 3.7.15.3

This SR verifies that each PREACS train starts and operates on an actual or simulated actuation signal. The [18] month Frequency is consistent with that specified in Reference 4.]

[SR 3.7.15.4

This SR verifies 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 post accident 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 NUREG-0800, Section 6.5.1 (Ref. 6).]

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.15.4 (continued)

[The minimum system flow rate maintains a slight negative pressure in the penetration room area and provides sufficient air velocity to transport particulate contaminants, assuming only one filter train is operating.

The number of filter elements is selected to limit the flow rate through any individual element to about [1000] cfm. This may vary based on filter housing geometry. The maximum limit ensures that 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.

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 adsorber elements at the design flow rate. An increase in pressure drop or 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; thus, an [18] month Frequency on a STAGGERED TEST BASIS consistent with other filtration SRs.]

[SR 3.7.15.5

Operating the PREACS filter bypass damper is necessary to ensure that the system functions properly. The OPERABILITY of the PREACS filter bypass damper is verified if it can be closed. An [18] month Frequency is consistent with that specified in Reference 4.]

REFERENCES

1. FSAR, Section [6.5.1].

2. FSAR, Section [9.4.5].

(continued)

BASES

REFERENCES (continued)

3. FSAR, Section [15.6.5].
4. Regulatory Guide 1.52.
5. 10 CFR 100.11.
6. NUREG-0800, Section 6.5.1.

B 3.7 PLANT SYSTEMS

B 3.7.16 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 filled to their maximum capacity. The water also provides shielding during the movement of spent fuel. A general description of the fuel storage pool design is given in the FSAR, Section [9.1.2], Reference 1, and the Spent Fuel Pool Cooling and Cleanup System is given in the FSAR, Section [9.1.3] (Ref. 2). The assumptions of the fuel handling accident are given in the FSAR, Section [15.7.4] (Ref. 3).
APPLICABLE SAFETY ANALYSES	The minimum water level in the fuel storage pool meets the assumptions of the fuel handling accident described in Regulatory Guide 1.25 (Ref. 4). The resultant 2 hour thyroid dose to a person at the exclusion area boundary is a small fraction of the 10 CFR 100 (Ref. 5) limits. According to Reference 4, there is 23 ft of water between the top of the damaged fuel bundle and the fuel pool surface for a fuel handling accident. With a 23 ft water level, the assumptions of Reference 4 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 < 23 ft of water above the top of the bundle and the surface, by the width of the bundle. To offset this small nonconservatism, the analysis assumes that all fuel rods fail, although analysis shows that only the first few rods fail from a hypothetical maximum drop. The fuel storage pool water level satisfies Criterion 3 of the NRC Policy Statement.

(continued)

BASES (continued)

LCO

The specified water level preserves the assumptions of the fuel handling accident analysis (Ref. 3). As such, it is the minimum required for fuel storage and movement within the fuel storage pool.

APPLICABILITY

This LCO applies during movement of irradiated fuel assemblies in the fuel storage pool since the potential for a release of fission products exists.

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply.

When the initial conditions for an accident cannot be met, steps should be taken to preclude the accident from occurring. When the fuel storage pool water level is lower than the required level, the movement of irradiated fuel assemblies in the fuel storage pool is immediately suspended. This effectively precludes a spent fuel handling accident from occurring. This does not preclude moving a fuel assembly to a safe position.

If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODES 1, 2, 3, and 4, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.7.16.1

This SR verifies sufficient fuel storage pool water is available in the event of a fuel handling accident. The water level in the fuel storage pool must be checked periodically. The 7 day Frequency is appropriate because the volume in the pool is normally stable. Water level changes are controlled by unit procedures and are acceptable, based on operating experience.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.16.1 (continued)

During refueling operations, the level in the fuel storage pool is at equilibrium with that of the refueling canal, and the level in the refueling canal is checked daily in accordance with LCO 3.7.17, "Fuel Storage Pool Boron Concentration."

REFERENCES

1. FSAR, Section [9.1.2].
2. FSAR, Section [9.1.3].
3. FSAR, Section [15.7.4].
4. Regulatory Guide 1.25
5. 10 CFR 100.11.

B 3.7 PLANT SYSTEMS

B 3.7.17 Fuel Storage Pool Boron Concentration

BASES

BACKGROUND	As described in LCO 3.7.18, "Spent Fuel Assembly Storage," fuel assemblies are stored in the spent fuel racks [in a "checkerboard" pattern] in accordance with criteria based on [initial enrichment and discharge burnup]. Although the water in the spent fuel pool is normally borated to ≥ [1800] ppm, the criteria that limit the storage of a fuel assembly to specific rack locations is conservatively developed without taking credit for boron.
APPLICABLE SAFETY ANALYSES	A fuel assembly could be inadvertently loaded into a spent fuel rack location not allowed by LCO 3.7.18 (e.g., an unirradiated fuel assembly or an insufficiently depleted fuel assembly). This accident is analyzed assuming the extreme case of completely loading the fuel pool racks with unirradiated assemblies of maximum enrichment. Another type of postulated accident is associated with a fuel assembly that is dropped onto the fully loaded fuel pool storage rack. Either incident could have a positive reactivity effect, decreasing the margin to criticality. However, the negative reactivity effect of the soluble boron compensates for the increased reactivity caused by either one of the two postulated accident scenarios. The concentration of dissolved boron in the fuel pool satisfies Criterion 2 of the NRC Policy Statement.
LCO	The specified concentration of dissolved boron in the fuel pool preserves the assumptions used in the analyses of the potential accident scenarios described above. This concentration of dissolved boron is the minimum required concentration for fuel assembly storage and movement within the fuel pool.
APPLICABILITY	This LCO applies whenever fuel assemblies are stored in the spent fuel pool until a complete spent fuel pool

(continued)

BASES

APPLICABILITY (continued)

verification has been performed following the last movement of fuel assemblies in the spent fuel pool. This LCO does not apply following the verification since the verification would confirm that there are no misloaded fuel assemblies. With no further fuel assembly movements in progress, there is no potential for a misloaded fuel assembly or a dropped fuel assembly.

ACTIONS

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

The Required Actions are modified by a Note indicating that LCO 3.0.3 does not apply.

When the concentration of boron in the spent fuel pool is less than required, immediate action must be taken to preclude an accident from happening or to mitigate the consequences of an accident in progress. This is most efficiently achieved by immediately suspending the movement of fuel assemblies. This does not preclude the movement of fuel assemblies to a safe position. In addition, action must be immediately initiated to restore boron concentration to within limit. Alternately, an immediate verification, by administrative means, of the fuel storage pool fuel locations, to ensure proper locations of the fuel since the last movement of fuel assemblies in the fuel storage pool, can be performed.

If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, inability to suspend movement of fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.7.17.1

This SR verifies that the concentration of boron in the spent fuel pool is within the required limit. As long as this SR is met, the analyzed incidents are fully addressed. The 7 day Frequency is appropriate because no major replenishment of pool water is expected to take place over a short period of time.

(continued)

BASES (continued)

REFERENCES None.

B 3.7 PLANT SYSTEMS

B 3.7.18 Spent Fuel Assembly Storage

BASES

BACKGROUND

The spent fuel storage facility is designed to store either new (nonirradiated) nuclear fuel assemblies, or burned (irradiated) fuel assemblies in a vertical configuration underwater. The storage pool is sized to store [735] irradiated fuel assemblies, which includes storage for [15] failed fuel containers. The spent fuel storage cells are installed in parallel rows with center to center spacing of [12 31/32] inches in one direction, and [13 3/16] inches in the other orthogonal direction. This spacing and "flux trap" construction, whereby the fuel assemblies are inserted into neutron absorbing stainless steel cans, is sufficient to maintain a k_{eff} of $\leq$ 0.95 for spent fuel of original enrichment of up to [3.3]%. However, as higher initial enrichment fuel assemblies are stored in the spent fuel pool, they must be stored in a checkerboard pattern taking into account fuel burnup to maintain a k_{eff} of 0.95 or less.

APPLICABLE SAFETY ANALYSES

The spent fuel storage facility is designed for noncriticality by use of adequate spacing, and "flux trap" construction whereby the fuel assemblies are inserted into neutron absorbing stainless steel cans.

The spent fuel assembly storage satisfies Criterion 2 of the NRC Policy Statement.

LCO

The restrictions on the placement of fuel assemblies within the spent fuel pool, according to [Figure 3.7.18-1], in the accompanying LCO, ensures that the k_{eff} of the spent fuel pool will always remain < 0.95 assuming the pool to be flooded with unborated water. The restrictions are consistent with the criticality safety analysis performed for the spent fuel pool according to [Figure 3.7.18-1], in the accompanying LCO. Fuel assemblies not meeting the criteria of [Figure 3.7.18-1] shall be stored in accordance with Specification 4.3.1.1.

(continued)

BASES (continued)

APPLICABILITY	This LCO applies whenever any fuel assembly is stored in [Region 2] of the spent fuel pool.
ACTIONS	A.1 Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply. When the configuration of fuel assemblies stored in [Region 2] the spent fuel pool is not in accordance with Figure [3.7.18-1], immediate action must be taken to make the necessary fuel assembly movement(s) to bring the configuration into compliance with Figure [3.7.18-1]. If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, in either case, inability to move fuel assemblies is not sufficient reason to require a reactor shutdown.
SURVEILLANCE REQUIREMENTS	SR 3.7.18.1 This SR verifies by administrative means that the initial enrichment and burnup of the fuel assembly is in accordance with Figure [3.7.18-1] in the accompanying LCO. For fuel assemblies in the unacceptable range of [Figure 3.7.18-1], performance of this SR will ensure compliance with Specification 4.3.1.1.
REFERENCES	None.

B 3.7 PLANT SYSTEMS

B 3.7.19 Secondary Specific Activity

BASES

BACKGROUND

Activity in the secondary coolant results from steam generator tube outleakage 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 environment because of normal operation, anticipated operational occurrences, and accidents.

This limit is lower than the activity value that might be expected from a 1 gpm tube leak (LCO 3.4.13, "RCS Operational LEAKAGE") of primary coolant at the limit of 1.0 μCi/gm (LCO 3.4.16, "RCS Specific Activity"). The steam line failure is assumed to result in the release of the noble gas and iodine activity contained in the steam generator inventory, the feedwater, and reactor coolant LEAKAGE. Most of the iodine isotopes have short half lives (i.e., < 20 hours). I-131, with a half life of 8.04 days, concentrates faster than it decays, but does not reach equilibrium because of blowdown and other losses.

With the specified activity level, the resultant 2 hour thyroid dose to a person at the exclusion area boundary (EAB) would be about [.13] rem should the main steam safety valves (MSSVs) open for the 2 hours following a trip from full power.

Operating a unit at the allowable limits could result in a 2 hour EAB exposure of a small fraction of the 10 CFR 100 (Ref. 1) limits.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The accident analysis of the main steam line break (MSLB), as discussed in the FSAR, Chapter [15] (Ref. 2), assumes the initial secondary coolant specific activity to have a radioactive isotope concentration of [0.10] μCi/gm DOSE EQUIVALENT I-131. This assumption is used in the analysis for determining the radiological consequences of the postulated accident. The accident analysis, based on this and other assumptions, shows that the radiological consequences of an MSLB do not exceed a small fraction of the unit EAB limits (Ref. 1) 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 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 have 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.

Secondary specific activity limits satisfy Criterion 2 of the NRC Policy Statement.

LCO

As indicated in the Applicable Safety Analyses, the specific activity limit in the secondary coolant system of ≤ [0.10] μCi/gm DOSE EQUIVALENT I-131 to limit the radiological consequences of a Design Basis Accident (DBA) to a small fraction of the required limit (Ref. 1).

Monitoring the specific activity of the secondary coolant ensures that when secondary specific activity limits are exceeded, appropriate actions are taken in a timely manner to place the unit in an operational MODE that would minimize the radiological consequences of a DBA.

(continued)

BASES (continued)

APPLICABILITY

In MODES 1, 2, 3, and 4, the limits on secondary specific activity apply due to the potential for secondary steam releases to the atmosphere.

In MODES 5 and 6, the steam generators are not being used for heat removal. Both the RCS and steam generators are depressurized, and primary to secondary LEAKAGE is minimal. Therefore, monitoring of secondary specific activity is not required.

ACTIONS

A.1 and A.2

DOSE EQUIVALENT I-131 exceeding the allowable value in the secondary coolant, is an indication of a problem in the RCS, and contributes to increased post accident doses. If secondary specific activity cannot be restored to within limits in the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.19.1

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 analysis assumptions as to the source terms in post accident releases. It also serves to identify and trend any unusual isotopic concentrations that might indicate changes in reactor coolant activity or LEAKAGE. The [31] day Frequency is based on the detection of increasing trends of the level of DOSE EQUIVALENT I-131, and allows for appropriate action to be taken to maintain levels below the LCO limit.

(continued)

BASES (continued)

REFERENCES

1. 10 CFR 100.11.
2. FSAR, Chapter [15].

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources—Operating

BASES

BACKGROUND

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

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

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

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

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

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

(continued)

BASES

BACKGROUND (continued)

automatically on a safety injection (SI) signal (i.e., low pressurizer pressure or high containment pressure signals) or on an [ESF bus degraded voltage or undervoltage signal]. After the DG has started, it will automatically tie to its respective bus after offsite power is tripped as a consequence of ESF bus undervoltage or degraded voltage, independent of or coincident with an SI signal. The DGs will also start and operate in the standby mode without tying to the ESF bus on an SI signal alone. Following the trip of offsite power, [a sequencer/an undervoltage signal] strips nonpermanent loads from the ESF bus. When the DG is tied to the ESF bus, loads are then sequentially connected to its respective ESF bus by the automatic load sequencer. The sequencing logic controls the permissive and starting signals to motor breakers to prevent overloading the DG by automatic load application.

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

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

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

APPLICABLE SAFETY ANALYSES

The initial conditions of DBA and transient analyses in the FSAR, Chapter [6] (Ref. 4) and Chapter [15] (Ref. 5), assume ESF systems are OPERABLE. The AC electrical power sources are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, Reactor Coolant System (RCS), and containment design limits are not exceeded. These limits are discussed in more detail in the

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Bases for Section 3.2, Power Distribution Limits; Section 3.4, Reactor Coolant System (RCS); and Section 3.6, Containment Systems.

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

a. An assumed loss of all offsite power or all onsite AC power; and

b. A worst case single failure.

The AC sources satisfy Criterion 3 of NRC Policy Statement.

LCO

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

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

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

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

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

(continued)

BASES

LCO (continued)

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

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

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

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

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

APPLICABILITY

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

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

(continued)

BASES

APPLICABILITY (continued)

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

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

ACTIONS

A.1

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

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

A.2

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

(continued)

BASES

ACTIONS

A.2 (continued)

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

a. The train has no offsite power supplying its loads; and

b. A required feature on the other train is inoperable.

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

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

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

A.3

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

(continued)

BASES

ACTIONS

A.3 (continued)

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 unit safety systems. In this Condition, however, the remaining OPERABLE offsite circuit and DGs are adequate to supply electrical power to the onsite Class 1E Distribution System.

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

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

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

(continued)

BASES

ACTIONS (continued)

B.1

To ensure a highly reliable power source remains with an inoperable DG, it is necessary to verify the availability of the offsite circuits on a more frequent basis. Since the Required Action only specifies "perform," a failure of SR 3.8.1.1 acceptance criteria does not result in a Required Action being not met. However, if a circuit fails to pass SR 3.8.1.1, it is inoperable. Upon offsite circuit inoperability, additional Conditions and Required Actions must then be entered.

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

B.2

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

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

(continued)

BASES

ACTIONS

B.2 (continued)

a. An inoperable DG exists; and

b. A required feature on the other train is inoperable.

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

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

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

B.3.1 and B.3.2

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

(continued)

BASES

ACTIONS

B.3.1 and B.3.2 (continued)

SR 3.8.1.2 suffices to provide assurance of continued OPERABILITY of that DG.

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

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

B.4

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

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

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

(continued)

BASES

ACTIONS

B.4 (continued)

and B are entered concurrently. The "AND" connector between the 72 hour and 6 day Completion Times means that both Completion Times apply simultaneously, and the more restrictive Completion Time must be met.

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

C.1 and C.2

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

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

a. All required offsite circuits are inoperable; and

b. A required feature is inoperable.

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

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

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

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

a. The configuration of the redundant AC electrical power system that remains available is not susceptible to a single bus or switching failure; and

b. The time required to detect and restore an unavailable offsite power source is generally much less than that required to detect and restore an unavailable onsite AC source.

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

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

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

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

D.1 and D.2

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

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

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

E.1

With Train A and Train B DGs inoperable, there are no remaining standby AC sources. Thus, with an assumed loss of offsite electrical power, insufficient standby AC sources

(continued)

BASES

ACTIONS

E.1 (continued)

are available to power the minimum required ESF functions. Since the offsite electrical power system is the only source of AC power for this level of degradation, the risk associated with continued operation for a short time could be less than that associated with an immediate controlled shutdown (the immediate shutdown could cause grid instability, which could result in a total loss of AC power). Since any inadvertent generator trip could also result in a total loss of offsite AC power, however, the time allowed for continued operation is severely restricted. The intent here is to avoid the risk associated with an immediate controlled shutdown and to minimize the risk associated with this level of degradation.

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

F.1

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

This Condition is preceded by a Note that allows the Condition to be deleted if the unit design is such that any sequencer failure mode will only affect the ability of the associated DG to power its respective safety loads under any conditions. Implicit in this Note is the concept that the Condition must be retained if any sequencer failure mode results in the inability to start all or part of the safety loads when required, regardless of power availability, or results in overloading the offsite power circuit to a safety bus during an event, thereby causing its failure. Also

(continued)

BASES

ACTIONS

F.1 (continued)

implicit in the Note, is that the Condition is not applicable to any train that does not have a sequencer.

G.1 and G.2

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

H.1

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

SURVEILLANCE REQUIREMENTS

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

Where the SRs discussed herein specify voltage and frequency tolerances, the following is applicable. The minimum steady state output voltage of [3740] V is 90% of the nominal 4160 V output voltage. This value, which is specified in

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

ANSI C84.1-1982 (Ref. 11), allows for voltage drop to the terminals of 4000 V motors whose minimum operating voltage is specified as 90% or 3600 V. It also allows for voltage drops to motors and other equipment down through the 120 V level where minimum operating voltage is also usually specified as 80% of name plate rating. The specified maximum steady state output voltage of [4756] V is equal to the maximum operating voltage specified for 4000 V motors. It ensures that for a lightly loaded distribution system, the voltage at the terminals of 4000 V motors is no more than the maximum rated operating voltages. The specified minimum and maximum frequencies of the DG are 58.8 Hz and 61.2 Hz, respectively. These values are equal to ± 2% of the 60 Hz nominal frequency and are derived from the recommendations given in Regulatory Guide 1.9 (Ref. 3).

SR 3.8.1.1

This SR 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 that distribution buses and loads are connected to their preferred power source, and that appropriate independence of offsite circuits is maintained. The 7 day Frequency is adequate since breaker position is not likely to change without the operator being aware of it and because its status is displayed in the control room.

SR 3.8.1.2 and SR 3.8.1.7

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

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

For the purposes of SR 3.8.1.2 and SR 3.8.1.7 testing, the DGs are started from standby conditions. Standby conditions

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.2 and SR 3.8.1.7 (continued)

for a DG mean the diesel engine coolant and oil are being continuously circulated and temperature is being maintained consistent with manufacturer recommendations.

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

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

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

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

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

SR 3.8.1.3

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

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.3 (continued)

while minimizing the time that the DG is connected to the offsite source.

Although no power factor requirements are established by this SR, the DG is normally operated at a power factor between [0.8 lagging] and [1.0]. The 0.8 value is the design rating of the machine, while [1.0] is an operational limitation [to ensure circulating currents are minimized]. The normal 31 day Frequency for this Surveillance (Table 3.8.1-1) is consistent with Regulatory Guide 1.9 (Ref. 3).

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

SR 3.8.1.4

This SR provides verification that the level of fuel oil in the day tank [and engine mounted tank] is at or above the level at which fuel oil is automatically added. The level is expressed as an equivalent volume in gallons, and is selected to ensure adequate fuel oil for a minimum of 1 hour of DG operation at full load plus 10%.

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

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.5

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

SR 3.8.1.6

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

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

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.6 (continued)

[Frequency is appropriate. Since proper operation of fuel transfer systems is an inherent part of DG OPERABILITY, the Frequency of this SR should be modified to reflect individual designs.]

SR 3.8.1.7

See SR 3.8.1.2.

[SR 3.8.1.8

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

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

SR 3.8.1.9

Each DG is provided with an engine overspeed trip to prevent damage to the engine. Recovery from the transient caused by the loss of a large load could cause diesel engine overspeed, which, if excessive, might result in a trip of the engine. This Surveillance demonstrates the DG load response characteristics and capability to reject the largest single load without exceeding predetermined voltage

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.9 (continued)

and frequency and while maintaining a specified margin to the overspeed trip. [For this unit, the single load for each DG and its horsepower rating is as follows:] This Surveillance may be accomplished by:

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

b. Tripping its associated single largest post-accident load with the DG solely supplying the bus.

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

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

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

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.9 (continued)

of the actual design basis inductive loading that the DG would experience.

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

a. Performance of the SR will not render any safety system or component inoperable;

b. Performance of the SR will not cause perturbations to any of the electrical distribution systems that could result in a challenge to steady state operation or to plant safety systems; and

c. Performance of the SR, or failure of the SR, will not cause, or result in, an AOO with attendant challenge to plant safety systems.

SR 3.8.1.10

This Surveillance demonstrates the DG capability to reject a full load without overspeed tripping or exceeding the predetermined voltage limits. The DG full load rejection may occur because of a system fault or inadvertent breaker tripping. This Surveillance ensures proper engine generator load response under the simulated test conditions. This test simulates the loss of the total connected load that the DG experiences following a full load rejection and verifies that the DG will not trip upon loss of the load. These acceptance criteria provide DG damage protection. While the DG is not expected to experience this transient during an event and continues to be available, this response ensures that the DG is not degraded for future application, including reconnection to the bus if the trip initiator can be corrected or isolated.

In order to ensure that the DG is tested under load conditions that are as close to design basis conditions as possible, testing must be performed using a power factor ≤ [0.9]. This power factor is chosen to be representative

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.10 (continued)

of the actual design basis inductive loading that the DG would experience.

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

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

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

a. Performance of the SR will not render any safety system or component inoperable;

b. Performance of the SR will not cause perturbations to any of the electrical distribution systems that could result in a challenge to steady state operation or to plant safety systems; and

c. Performance of the SR, or failure of the SR, will not cause, or result in, an AOO with attendant challenge to plant safety systems.]

SR 3.8.1.11

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

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.11 (continued)

automatically achieve the required voltage and frequency within the specified time.

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

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

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

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

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.11 (continued)

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

SR 3.8.1.12

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

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

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

This SR is modified by two Notes. The reason for Note 1 is to minimize wear and tear on the DGs during testing. For the purpose of this testing, the DGs must be started from

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.12 (continued)

standby conditions, that is, with the engine coolant and oil continuously circulated and temperature maintained consistent with manufacturer recommendations. The reason for Note 2 is that during operation with the reactor critical, performance of this Surveillance could cause perturbations to the electrical distribution systems that could challenge continued steady state operation and, as a result, unit safety systems. Credit may be taken for unplanned events that satisfy this SR.

SR 3.8.1.13

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

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

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

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.13 (continued)

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

a. Performance of the SR will not render any safety system or component inoperable;

b. Performance of the SR will not cause perturbations to any of the electrical distribution systems that could result in a challenge to steady state operation or to plant safety systems; and

c. Performance of the SR, or failure of the SR, will not cause, or result in, an AOO with attendant challenge to plant safety systems.

SR 3.8.1.14

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

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

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.14 (continued)

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

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

SR 3.8.1.15

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

This SR is modified by two Notes. Note 1 ensures that the test is performed with the diesel sufficiently hot. The load band is provided to avoid routine overloading of the DG. Routine overloads may result in more frequent teardown inspections in accordance with vendor recommendations in order to maintain DG OPERABILITY. The requirement that the diesel has operated for at least [2] hours at full load conditions prior to performance of this Surveillance is based on manufacturer recommendations for achieving hot conditions. Momentary transients due to changing bus loads do not invalidate this test. Note 2 allows all DG starts to be preceded by an engine prelube period to minimize wear and tear on the diesel during testing.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.16

As required by Regulatory Guide 1.108 (Ref. 9), paragraph 2.a.(6), this Surveillance ensures that the manual synchronization and automatic load transfer from the DG to the offsite source can be made and that 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 autoclose signal on bus undervoltage, and the load sequence timers are reset.

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

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

SR 3.8.1.17

Demonstration of the test mode override ensures that the DG availability under accident conditions will not be compromised as the result of testing and the DG will automatically reset to ready to load operation if a LOCA actuation signal is received during operation in the test mode. Ready to load operation is defined as the DG running at rated speed and voltage with the DG output breaker open. These provisions for automatic switchover are required by IEEE-308 (Ref. 13), paragraph 6.2.6(2).

The requirement to automatically energize the emergency loads with offsite power is essentially identical to that of SR 3.8.1.12. The intent in the requirement associated with SR 3.8.1.17.b is to show that the emergency loading was not affected by the DG operation in test mode. In lieu of actual demonstration of connection and loading of loads, testing that adequately shows the capability of the

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.17 (continued)

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

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

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

SR 3.8.1.18

Under accident [and loss of offsite power] conditions loads are sequentially connected to the bus by the [automatic load sequencer]. The sequencing logic controls the permissive and starting signals to motor breakers to prevent overloading of the DGs due to high motor starting currents. The [10]% load sequence time interval tolerance ensures that sufficient time exists for the DG to restore frequency and voltage prior to applying the next load and that safety analysis assumptions regarding ESF equipment time delays are not violated. Reference 1 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. 9), paragraph 2.a.(2); takes into consideration unit conditions required to perform the Surveillance; and is intended to be consistent with expected fuel cycle lengths.

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

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.18 (continued)

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

a. Performance of the SR will not render any safety system or component inoperable;

b. Performance of the SR will not cause perturbations to any of the electrical distribution systems that could result in a challenge to steady state operation or to plant safety systems; and

c. Performance of the SR, or failure of the SR, will not cause, or result in, an AOO with attendant challenge to plant safety systems.]

SR 3.8.1.19

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

This Surveillance demonstrates the DG operation, as discussed in the Bases for SR 3.8.1.11, during a loss of offsite power actuation test signal in conjunction with an ESF actuation signal. In lieu of actual demonstration of connection and loading of loads, testing that adequately shows the capability of the DG system to perform these functions is acceptable. This testing may include any series of sequential, overlapping, or total steps so that the entire connection and loading sequence is verified.

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

This SR is modified by two Notes. The reason for Note 1 is to minimize wear and tear on the DGs during testing. For the purpose of this testing, the DGs must be started from

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.19 (continued)

standby conditions, that is, with the engine coolant and oil continuously circulated and temperature maintained consistent with manufacturer recommendations for DGs. The reason for Note 2 is that performing the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems. Credit may be taken for unplanned events that satisfy this SR.

SR 3.8.1.20

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

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

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

Diesel Generator Test Schedule

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

According to Regulatory Guide 1.9, Revision 3 (Ref. 3), each DG unit should be tested at least once every 31 days. Whenever a DG has experienced 4 or more valid failures in the last 25 valid tests, the maximum time between tests is reduced to 7 days. Four failures in 25 valid tests is a failure rate of 0.16, or the threshold of acceptable DG

(continued)

BASES

SURVEILLANCE REQUIREMENTS

Diesel Generator Test Schedule (continued)

performance, and hence may be an early indication of the degradation of DG reliability. When considered in the light of a long history of tests, however, 4 failures in the last 25 valid tests may only be a statistically probable distribution of random events. Increasing the test Frequency will allow for a more timely accumulation of additional test data upon which to base judgment of the reliability of the DG. The increased test Frequency must be maintained until seven consecutive, failure free tests have been performed.

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

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

REFERENCES

1. 10 CFR 50, Appendix A, GDC 17.
2. FSAR, Chapter [8].
3. Regulatory Guide 1.9, Rev. [3], [date].
4. FSAR, Chapter [6].
5. FSAR, Chapter [15].
6. Regulatory Guide 1.93, Rev. [ ], [date].
7. Generic Letter 84-15.
8. 10 CFR 50, Appendix A, GDC 18.
9. Regulatory Guide 1.108, Rev. [1], [August 1977].

(continued)

BASES

REFERENCES (continued)

10. Regulatory Guide 1.137, Rev. [ ], [date].

11. ANSI C84.1-1982.

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

13. IEEE Standard 308-[1978].

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 LCO 3.8.1, "AC Sources—Operating."

APPLICABLE SAFETY ANALYSES

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

a. The unit can be maintained in the shutdown or refueling condition for extended periods;

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and

c. Adequate AC electrical power is provided to mitigate events postulated during shutdown, such as a fuel handling accident.

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

During MODES 1, 2, 3, and 4, various deviations from the analysis assumptions and design requirements are allowed

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

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

a. The fact that time in an outage is limited. This is a risk prudent goal as well as a utility economic consideration.

b. Requiring appropriate compensatory measures for certain conditions. These may include administrative controls, reliance on systems that do not necessarily meet typical design requirements applied to systems credited in operating MODE analyses, or both.

c. Prudent utility consideration of the risk associated with multiple activities that could affect multiple systems.

d. Maintaining, to the extent practical, the ability to perform required functions (even if not meeting MODE 1, 2, 3, and 4 OPERABILITY requirements) with systems assumed to function during an event.

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

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

LCO

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

(continued)

BASES

LCO (continued)

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

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

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

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

Proper sequencing of loads, including tripping of nonessential loads, is a required function for DG OPERABILITY.

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

(continued)

BASES

LCO (continued)

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

APPLICABILITY

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

a. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel assemblies;

b. Systems needed to mitigate a fuel handling accident are available;

c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

The AC power requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.1.

ACTIONS

A.1

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

(continued)

BASES

ACTIONS (continued)

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

With the offsite circuit not available to all required trains, the option would still exist to declare all required features inoperable. Since this option may involve undesired administrative efforts, the allowance for sufficiently conservative actions is made. With the required DG inoperable, the minimum required diversity of AC power sources is not available. It is, therefore, required to suspend CORE ALTERATIONS, movement of irradiated fuel assemblies, and operations involving positive reactivity additions. The Required Action to suspend positive reactivity additions does not preclude actions to maintain or increase reactor vessel inventory provided the required SDM is maintained.

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

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

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

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BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.8.2.1

SR 3.8.2.1 requires the SRs from LCO 3.8.1 that are necessary for ensuring the OPERABILITY of the AC sources in other than MODES 1, 2, 3, and 4. SR 3.8.1.8 is not required to be met since only one offsite circuit is required to be OPERABLE. SR 3.8.1.17 is not required to be met because the required OPERABLE DG(s) is not required to undergo periods of being synchronized to the offsite circuit. SR 3.8.1.20 is excepted because starting independence is not required with DG(s) that are not required to be OPERABLE.

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

REFERENCES

None.

B 3.8 ELECTRICAL POWER SYSTEMS

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

BASES

BACKGROUND

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

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

For proper operation of the standby DGs, it is necessary to ensure the proper quality of the fuel oil. Regulatory Guide 1.137 (Ref. 2) addresses the recommended fuel oil practices as supplemented by ANSI N195-1976 (Ref. 3). The fuel oil properties governed by these SRs are the water and sediment content, the kinematic viscosity, specific gravity (or API gravity), and impurity level.

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

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

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BASES (continued)

APPLICABLE SAFETY ANALYSES

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

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

LCO

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

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

APPLICABILITY

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

(continued)

BASES

APPLICABILITY (continued)

air are required to be within limits when the associated DG is required to be OPERABLE.

ACTIONS

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

A.1

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

B.1

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

(continued)

BASES

ACTIONS

B.1 (continued)

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

C.1

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

D.1

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

(continued)

BASES

ACTIONS (continued)

E.1

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

F.1

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

SURVEILLANCE REQUIREMENTS

SR 3.8.3.1

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

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

SR 3.8.3.2

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

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.3.2 (continued)

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

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

SR 3.8.3.3

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

a. Sample the new fuel oil in accordance with ASTM D4057-[ ] (Ref. 6);

b. Verify in accordance with the tests specified in ASTM D975-[ ] (Ref. 6) that the sample has an absolute specific gravity at 60/60°F of $\geq$ 0.83 and $\leq$ 0.89, or an API gravity at 60°F of $\geq$ 27° and $\leq$ 39°, a kinematic viscosity at 40°C of $\geq$ 1.9 centistokes and $\leq$ 4.1 centistokes, and a flash point $\geq$ 125°F; and

c. Verify that the new fuel oil has a clear and bright appearance with proper color when tested in accordance with ASTM D4176-[ ] (Ref. 6).

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.3.3 (continued)

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

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

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

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

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

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.3.4

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

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

SR 3.8.3.5

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

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.3.6

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

REFERENCES

1. FSAR, Section [9.5.4.2].
2. Regulatory Guide 1.137.
3. ANSI N195-1976, Appendix B.
4. FSAR, Chapter [6].
5. FSAR, Chapter [15].
6. ASTM Standards: D4057-[ ]; D975-[ ]; D4175-[ ]; D1552-[ ]; D2622-[ ]; S2276, Method A.
7. ASTM Standards, D975, Table 1.
8. ASME, Boiler and Pressure Vessel Code, Section XI.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.4 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 and preferred AC vital bus power (via inverters). As required by 10 CFR 50, Appendix A, GDC 17 (Ref. 1), the DC electrical power system is designed to have sufficient independence, redundancy, and testability to perform its safety functions, assuming a single failure. The DC electrical power system also conforms to the recommendations of Regulatory Guide 1.6 (Ref. 2) and IEEE-308 (Ref. 3).

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

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

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

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

(continued)

BASES

BACKGROUND (continued)

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

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

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

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

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

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the FSAR, Chapter [6] (Ref. 6) and Chapter [15] (Ref. 7), assume that Engineered Safety Feature

(continued)

APPLICABLE SAFETY ANALYSES (continued)

(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 unit. This includes maintaining the DC sources OPERABLE during accident conditions in the event of:

a. An assumed loss of all offsite AC power or all onsite AC power; and

b. A worst case single failure.

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

LCO

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

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

APPLICABILITY

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

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

(continued)

BASES

APPLICABILITY (continued)

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

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

ACTIONS

A.1

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

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

B.1 and B.2

If the inoperable DC electrical power subsystem cannot be restored to OPERABLE status within the required Completion Time, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 5

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

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

SURVEILLANCE REQUIREMENTS

SR 3.8.4.1

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

SR 3.8.4.2

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

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

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

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.4.3

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

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

SR 3.8.4.4 and SR 3.8.4.5

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

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

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

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

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.4.6

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

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

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

SR 3.8.4.7

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

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

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

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.4.7 (continued)

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

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

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

SR 3.8.4.8

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

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

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.4.8 (continued)

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

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

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

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

REFERENCES

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

(continued)

BASES

REFERENCES (continued)

6. FSAR, Chapter [6].

7. FSAR, Chapter [15].

8. Regulatory Guide 1.93, December 1974.

9. IEEE-450-[1987].

10. Regulatory Guide 1.32, February 1977.

11. Regulatory Guide 1.129, December 1974.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.5 DC Sources—Shutdown

BASES

BACKGROUND	A description of the DC sources is provided in the Bases for LCO 3.8.4, "DC Sources—Operating."
APPLICABLE SAFETY ANALYSES	The initial conditions of Design Basis Accident (DBA) and transient analyses in the FSAR, Chapter [6] (Ref. 1) and Chapter [15] (Ref. 2), assume that Engineered Safety Feature (ESF) systems are OPERABLE. The DC electrical power system provides normal and emergency DC electrical power for the DGs, emergency auxiliaries, and control and switching during all MODES of operation. The OPERABILITY of the DC subsystems is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY. The OPERABILITY of the minimum DC electrical power sources during MODES 5 and 6 and during movement of irradiated fuel assemblies ensures that: a. The unit can be maintained in the shutdown or refueling condition for extended periods; b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and c. Adequate DC electrical power is provided to mitigate events postulated during shutdown, such as a fuel handling accident. The DC sources satisfy Criterion 3 of the NRC Policy Statement.
LCO	The DC electrical power subsystems, each subsystem consisting of two batteries, one battery charger per battery, and the corresponding control equipment and interconnecting cabling within the train, are required to be

(continued)

BASES

LCO (continued)

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

APPLICABILITY

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

a. Required features needed to mitigate a fuel handling accident are available;

b. Required features necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

c. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

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

ACTIONS

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

If two trains are required per LCO 3.8.10, the remaining train with DC power available may be capable of supporting sufficient systems to allow continuation of CORE ALTERATIONS and fuel movement. By allowing the option to declare required features inoperable with the associated DC power source(s) inoperable, appropriate restrictions will be implemented in accordance with the affected required features LCO ACTIONS. In many instances, this option may involve undesired administrative efforts. Therefore, the allowance for sufficiently conservative actions is made (i.e., to suspend CORE ALTERATIONS, movement of irradiated fuel assemblies, and operations involving positive reactivity additions). The Required Action to suspend positive reactivity additions does not preclude actions to

(continued)

BASES

ACTIONS

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

maintain or increase reactor vessel inventory, provided the required SDM is maintained.

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

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

SURVEILLANCE REQUIREMENTS

SR 3.8.5.1

SR 3.8.5.1 states that Surveillances required by SR 3.8.4.1 through SR 3.8.4.8 are applicable in these MODES. See the corresponding Bases for LCO 3.8.4 for a discussion of each SR.

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

REFERENCES

1. FSAR, Chapter [6].

2. FSAR, Chapter [15].

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.6 Battery Cell Parameters

BASES

BACKGROUND

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

APPLICABLE SAFETY ANALYSES

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

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

a. An assumed loss of all offsite AC power or all onsite AC power; and

b. A worst case single failure.

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

LCO

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

(continued)

BASES (continued)

APPLICABILITY

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

ACTIONS

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

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

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

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

(continued)

BASES

ACTIONS

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

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

B.1

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

SURVEILLANCE REQUIREMENTS

SR 3.8.6.1

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

SR 3.8.6.2

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

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.6.2 (continued)

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

SR 3.8.6.3

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

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

Table 3.8.6-1

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

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

The Category A limits specified for electrolyte level are based on manufacturer recommendations and are consistent with the guidance in IEEE-450 (Ref. 3), with the extra ¼ inch allowance above the high water level indication for operating margin to account for temperatures and charge effects. In addition to this allowance, footnote a to Table 3.8.6-1 permits the electrolyte level to be above the specified maximum level during equalizing charge, provided it is not overflowing. These limits ensure that the plates

(continued)

BASES

SURVEILLANCE REQUIREMENTS

Table 3.8.6-1 (continued)

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

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

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

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

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

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

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BASES

SURVEILLANCE REQUIREMENTS

Table 3.8.6-1 (continued)

Category C defines the limit 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 limit, the assurance of sufficient capacity described above no longer exists and the battery must be declared inoperable.

The Category C limit specified for electrolyte level (above the top of the plates and not overflowing) ensures 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. 3), which states that a cell voltage of 2.07 V or below, under float conditions and not caused by elevated temperature of the cell, indicates internal cell problems and may require cell replacement.

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

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

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

(continued)

BASES

SURVEILLANCE REQUIREMENTS

Table 3.8.6-1 (continued)

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

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

REFERENCES

1. FSAR, Chapter [6].
2. FSAR, Chapter [15].
3. IEEE-450-[1980].

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.7 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. The function of the inverter is to provide AC electrical power to the vital buses. The inverters can be powered from an internal AC source/rectifier or from the station battery. The station battery provides an uninterruptible power source for the instrumentation and controls for the Reactor Protective System (RPS) and the Engineered Safety Feature Actuation System (ESFAS). Specific details on inverters and their operating characteristics are found in the FSAR, Chapter [8] (Ref. 1).

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the FSAR, Chapter [6] (Ref. 2) and Chapter [15] (Ref. 3), assume Engineered Safety Feature systems are OPERABLE. The inverters are designed to provide the required capacity, capability, redundancy, and reliability to ensure the availability of necessary power to the RPS and ESFAS instrumentation and controls so that the fuel, Reactor Coolant System, and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for Section 3.2, Power Distribution Limits; Section 3.4, Reactor Coolant System (RCS); and Section 3.6, Containment Systems.

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

a. An assumed loss of all offsite AC electrical power or all onsite AC electrical power; and

b. A worst case single failure.

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

(continued)

BASES (continued)

LCO

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

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

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

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

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

(continued)

BASES (continued)

APPLICABILITY

The inverters are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure that:

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

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

Inverter requirements for MODES 5 and 6 are covered in the Bases for LCO 3.8.8, "Inverters—Shutdown."

ACTIONS

A.1

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

Required Action A.1 is modified by a Note, which states to enter the applicable conditions and Required Actions of LCO 3.8.9, "Distribution Systems—Operating," when Condition A is entered with one AC vital bus de-energized. This ensures the vital bus is re-energized within 2 hours.

Required Action A.1 allows 24 hours to fix the inoperable inverter and return it to service. The 24 hour limit is based upon engineering judgment, taking into consideration the time required to repair an inverter and the additional risk to which the unit is exposed because of the inverter inoperability. This has to be balanced against the risk of an immediate shutdown, along with the potential challenges to safety systems such a shutdown might entail. When the AC vital bus is powered from its constant voltage source, it is relying upon interruptible AC electrical power sources (offsite and onsite). The uninterruptible inverter source to the AC vital buses is the preferred source for powering instrumentation trip setpoint devices.

(continued)

BASES

ACTIONS (continued)	<u>B.1 and B.2</u> If the inoperable devices or components cannot be restored to OPERABLE status within the required Completion Time, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.
SURVEILLANCE REQUIREMENTS	<u>SR 3.8.7.1</u> This Surveillance verifies that the inverters are functioning properly with all required circuit breakers closed and AC vital buses energized from the inverter. The verification of proper voltage and frequency output ensures that the required power is readily available for the instrumentation of the RPS and ESFAS connected to the AC vital buses. The 7 day Frequency takes into account the redundant capability of the inverters and other indications available in the control room that alert the operator to inverter malfunctions.
REFERENCES	1. FSAR, Chapter [8]. 2. FSAR, Chapter [6]. 3. FSAR, Chapter [14].

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.8 Inverters—Shutdown

BASES

BACKGROUND

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

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the FSAR, Chapter [6] (Ref. 1) and Chapter [15] (Ref. 2), assume Engineered Safety Feature systems are OPERABLE. The DC to AC inverters are designed to provide the required capacity, capability, redundancy, and reliability to ensure the availability of necessary power to the Reactor Protective System and Engineered Safety Features Actuation System instrumentation and controls so that the fuel, Reactor Coolant System, and containment design limits are not exceeded.

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

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

a. The unit can be maintained in the shutdown or refueling condition for extended periods;

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and

c. Adequate power is available to mitigate events postulated during shutdown, such as a fuel handling accident.

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

(continued)

BASES (continued)

LCO

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

APPLICABILITY

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

a. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel in the core;

b. Systems needed to mitigate a fuel handling accident are available;

c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

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

ACTIONS

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

If two trains are required by LCO 3.8.10, "Distribution Systems—Shutdown," the remaining OPERABLE inverters may be capable of supporting sufficient required features to allow

(continued)

BASES

ACTIONS

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

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

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

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

SURVEILLANCE REQUIREMENTS

SR 3.8.8.1

This Surveillance verifies that the inverters are functioning properly with all required circuit breakers closed and AC vital buses energized from the inverter. The verification of proper voltage and frequency output ensures that the required power is readily available for the

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.8.1 (continued)

instrumentation connected to the AC vital buses. The 7 day Frequency takes into account the redundant capability of the inverters and other indications available in the control room that alert the operator to inverter malfunctions.

REFERENCES

1. FSAR, Chapter [6].

2. FSAR, Chapter [15].

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.9 Distribution Systems—Operating

BASES

BACKGROUND

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

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

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

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

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

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

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the FSAR, Chapter [6] (Ref. 1) and Chapter [15] (Ref. 2), assume ESF systems are OPERABLE. The AC, DC, and AC vital bus electrical power distribution systems are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, Reactor Coolant System, and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for Section 3.2, Power Distribution Limits; Section 3.4, Reactor Coolant System (RCS); and Section 3.6, Containment Systems.

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

a. An assumed loss of all offsite power or all onsite AC electrical power; and

b. A worst case single failure.

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

LCO

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

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

(continued)

BASES

LCO (continued)

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

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

APPLICABILITY

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

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

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

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

(continued)

BASES (continued)

ACTIONS

A.1

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

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

a. The potential for decreased safety if the unit operator's attention is diverted from the evaluations and actions necessary to restore power to the affected train, to the actions associated with taking the unit to shutdown within this time limit; and

b. The potential for an event in conjunction with a single failure of a redundant component in the train with AC power.

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

(continued)

BASES

ACTIONS

A.1 (continued)

distribution system. At this time, a DC circuit could again become inoperable, and AC distribution restored OPERABLE. This could continue indefinitely.

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

B.1

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

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

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

(continued)

BASES

ACTIONS

B.1 (continued)

a. The potential for decreased safety by requiring a change in unit conditions (i.e., requiring a shutdown) and not allowing stable operations to continue;

b. The potential for decreased safety by requiring entry into numerous Applicable Conditions and Required Actions for components without adequate vital AC power and not providing sufficient time for the operators to perform the necessary evaluations and actions for restoring power to the affected train; and

c. The potential for an event in conjunction with a single failure of a redundant component.

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

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

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

(continued)

BASES

ACTIONS (continued)

C.1

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

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

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

a. The potential for decreased safety by requiring a change in unit conditions (i.e., requiring a shutdown) while allowing stable operations to continue;

b. The potential for decreased safety by requiring entry into numerous applicable Conditions and Required Actions for components without DC power and not providing sufficient time for the operators to perform the necessary evaluations and actions for restoring power to the affected train; and

c. The potential for an event in conjunction with a single failure of a redundant component.

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

(continued)

BASES

ACTIONS

C.1 (continued)

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

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

D.1 and D.2

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

E.1

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

(continued)

BASES

ACTIONS

E.1 (continued)

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

SURVEILLANCE REQUIREMENTS

SR 3.8.9.1

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

REFERENCES

1. FSAR, Chapter [6].
2. FSAR, Chapter [15].
3. Regulatory Guide 1.93, December 1974.

Table B 3.8.9-1 (page 1 of 1)
AC and DC Electrical Power Distribution Systems

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

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

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.10 Distribution Systems—Shutdown

BASES

BACKGROUND

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

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident and transient analyses in the FSAR, Chapter [6] (Ref. 1) and Chapter [15] (Ref. 2), assume Engineered Safety Feature (ESF) systems are OPERABLE. The AC, DC, and AC vital bus electrical power distribution systems are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, Reactor Coolant System, and containment design limits are not exceeded.

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

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

a. The unit can be maintained in the shutdown or refueling condition for extended periods;

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and

c. Adequate power is provided to mitigate events postulated during shutdown, such as a fuel handling accident.

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

(continued)

BASES (continued)

LCO

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

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

APPLICABILITY

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

a. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel in the core;

b. Systems needed to mitigate a fuel handling accident are available;

c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition and refueling condition.

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

(continued)

BASES (continued)

ACTIONS

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

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

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

Notwithstanding performance of the above conservative Required Actions, a required shutdown cooling (SDC) subsystem may be inoperable. In this case, Required Actions A.2.1 through A.2.4 do not adequately address the concerns relating to coolant circulation and heat removal. Pursuant to LCO 3.0.6, the SDC ACTIONS would not be entered. Therefore, Required Action A.2.5 is provided to direct declaring SDC inoperable, which results in taking the appropriate SDC actions.

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

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.8.10.1

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

REFERENCES

1. FSAR, Chapter [6].
2. FSAR, Chapter [15].

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 canal, and refueling cavity during refueling ensures that the reactor remains subcritical during MODE 6. Refueling boron concentration is the soluble boron concentration in the coolant in each of these volumes having direct access to the reactor core during refueling.

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

GDC 26 of 10 CFR 50, Appendix A, requires that two independent reactivity control systems of different design principles be provided (Ref. 1). One of these systems must be capable of holding the reactor core subcritical under cold conditions. The Chemical 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 canal and the refueling cavity are then flooded with borated water from the refueling water tank into the open reactor vessel by gravity feeding or by the use of the Shutdown Cooling (SDC) System pumps.

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

(continued)

BASES

BACKGROUND (continued)

concentrated boric acid with the water in the refueling canal. The SDC System is in operation during refueling (see LCO 3.9.4, "Shutdown Cooling (SDC) and Coolant Circulation—High Water Level," and LCO 3.9.5, "Shutdown Cooling (SDC) and Coolant Circulation—Low Water Level") to provide forced circulation in the RCS and assist in maintaining the boron concentrations in the RCS, the refueling canal, and the refueling cavity above the COLR limit.

APPLICABLE SAFETY ANALYSES

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

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

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

The limiting boron dilution accident analyzed occurs in MODE 5 (Ref. 2). A detailed discussion of this event is provided in B 3.1.2, "SHUTDOWN MARGIN—$T_{avg} \leq 200°F$."

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

LCO

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

(continued)

BASES

LCO (continued)

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, "SHUTDOWN MARGIN (SDM)—$T_{avg} > 200°F$," and LCO 3.1.2, "SHUTDOWN MARGIN—$T_{avg} \leq 200°F$," ensure that an adequate amount of negative reactivity is available to shut down the reactor and to maintain it subcritical.

ACTIONS

A.1 and A.2

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

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

A.3

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

In determining the required combination of boration flow rate and concentration, there is no unique design basis event that must be satisfied. The only requirement is to restore the boron concentration to its required value as soon as possible. In order to raise the boron concentration as soon as possible, the operator should begin boration with the best source available for unit conditions.

(continued)

BASES

ACTIONS

A.3 (continued)

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

SURVEILLANCE REQUIREMENTS

SR 3.9.1.1

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

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

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26.
2. FSAR, Section [ ].

B 3.9 REFUELING OPERATIONS

B 3.9.2 Nuclear Instrumentation

BASES

BACKGROUND	The source range monitors (SRMs) are used during refueling operations to monitor the core reactivity condition. The installed SRMs are part of the Nuclear Instrumentation System (NIS). These detectors are located external to the reactor vessel and detect neutrons leaking from the core. The use of portable detectors is permitted, provided the LCO requirements are met. The installed SRMs 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. If used, portable detectors should be functionally equivalent to the NIS SRMs.
APPLICABLE SAFETY ANALYSES	Two OPERABLE SRMs are required to provide a signal to alert the operator to unexpected changes in core reactivity such as by a boron dilution accident or an improperly loaded fuel assembly. The safety analysis of the uncontrolled boron dilution accident is described in Reference 2. The analysis of the uncontrolled boron dilution accident shows that normally available SHUTDOWN MARGIN would be reduced, but there is sufficient time for the operator to take corrective actions. The SRMs satisfy Criterion 3 of the NRC Policy Statement.
LCO	This LCO requires two SRMs OPERABLE to ensure that redundant monitoring capability is available to detect changes in core reactivity.

(continued)

BASES (continued)

APPLICABILITY

In MODE 6, the SRMs must be OPERABLE to determine changes in core reactivity. There is no other direct means available to check core reactivity levels.

In MODES 2, 3, 4, and 5, the installed source range detectors and circuitry are required to be OPERABLE by LCO 3.3.2, "RPS Instrumentation Shutdown."

ACTIONS

A.1 and A.2

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

B.1

With no SRM OPERABLE, action to restore a monitor to OPERABLE status shall be initiated immediately. Once initiated, action shall be continued until an SRM is restored to OPERABLE status.

B.2

With no SRM 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 SRMs are OPERABLE. This stabilized condition is determined by performing SR 3.9.1.1 to verify that the required boron concentration exists.

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

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.9.2.1

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

The Frequency of 12 hours is consistent with the CHANNEL CHECK Frequency specified similarly for the same instruments in LCO 3.3.1, "Reactor Protection System."

SR 3.9.2.2

SR 3.9.2.2 is the performance of a CHANNEL CALIBRATION every 18 months. This SR is modified by a Note stating that neutron detectors are excluded from the CHANNEL CALIBRATION. The CHANNEL CALIBRATION for the source range neutron flux monitors consists of obtaining the detector plateau or preamp discriminator curves, evaluating those curves, and comparing the curves to the manufacturer's data. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage. Operating experience has shown these components usually pass the Surveillance when performed on the 18 month Frequency.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 13, GDC 26, GDC 28, and GDC 29.

2. FSAR, Section [ ].

B 3.9 REFUELING OPERATIONS

B 3.9.3 Containment Penetrations

BASES

BACKGROUND

During CORE ALTERATIONS or movement of fuel assemblies within containment with irradiated fuel in containment, a release of fission product radioactivity within 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, the potential for containment pressurization as a result of an accident is not likely; therefore, requirements to isolate the containment from the outside atmosphere can be less stringent. The LCO requirements are referred to as "containment closure" rather than "containment OPERABILITY." Containment closure means that all potential escape paths are closed or capable of being closed. Since there is no potential for containment pressurization, the Appendix J leakage criteria and tests are not required.

The containment serves to contain fission product radioactivity that may be released from the reactor core following an accident, such that offsite radiation exposures are maintained well within the requirements of 10 CFR 100. Additionally, the containment 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 irradiated fuel assemblies within containment, the equipment hatch must be held in place by at least four bolts. Good engineering practice dictates that the bolts required by this LCO be approximately equally spaced.

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

(continued)

BASES

BACKGROUND (continued)

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

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

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

[The minipurge system remains operational in MODE 6 and all four valves are also closed by the ESFAS.

or

The minipurge system is not used in MODE 6. All four [8] inch valves are secured in the closed position.]

The other containment penetrations that provide direct access from containment atmosphere to outside atmosphere

(continued)

BASES

BACKGROUND (continued)	must be isolated on at least one side. Isolation may be achieved by an OPERABLE automatic isolation valve, or by a manual isolation valve, blind flange, or equivalent. Equivalent isolation methods must be approved and may include use of a material that can provide a temporary, atmospheric pressure ventilation barrier for the other containment penetrations during fuel movements (Ref. 1).
APPLICABLE SAFETY ANALYSES	During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, the most severe radiological consequences result from a fuel handling accident. The fuel handling accident is a postulated event that involves damage to irradiated fuel (Ref. 2). Fuel handling accidents, analyzed in Reference 3, include dropping a single irradiated fuel assembly and handling tool or a heavy object onto other irradiated fuel assemblies. The requirements of LCO 3.9.6, "Refueling 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. The acceptance limits for offsite radiation exposure are contained in Standard Review Plan Section 15.7.4, Rev. 1 (Ref. 2), which defines "well within" 10 CFR 100 to be 25% or less of the 10 CFR 100 values. Containment penetrations satisfy Criterion 3 of the NRC Policy Statement.
LCO	This LCO limits the consequences of a fuel handling accident in containment by limiting the potential escape paths for fission product radioactivity released within containment. The LCO requires any penetration providing direct access from the containment atmosphere to the outside atmosphere to be closed except for the OPERABLE containment purge and 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 ensure that the automatic purge and exhaust valve closure times specified in the FSAR can be achieved and therefore meet the assumptions used in the safety analysis

(continued)

BASES

LCO (continued)

to ensure releases through the valves are terminated, such that the radiological doses are within the acceptance limit.

APPLICABILITY

The containment penetration requirements are applicable during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment because this is when there is a potential for a fuel handling accident. In MODES 1, 2, 3, and 4, containment penetration requirements are addressed by LCO 3.6.1, "Containment." In MODES 5 and 6, when CORE ALTERATIONS or movement of irradiated fuel assemblies within 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 Isolation System not capable of automatic actuation when the purge and exhaust valves are open, the unit must be placed in a condition in which the isolation function is not needed. This is accomplished by immediately suspending CORE ALTERATIONS and movement of irradiated fuel assemblies within containment. Performance of these actions shall not preclude completion of movement of a component to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.9.3.1

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

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.3.1 (continued)

OPERABLE automatic containment purge and exhaust isolation signal.

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

(continued)

BASES (continued)

REFERENCES

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

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 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. This conditional de-energizing of the SDC pump does not result in a challenge to the fission product barrier.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

SDC and Coolant Circulation—High Water Level satisfies Criterion 2 of the NRC Policy Statement.

LCO

Only one SDC loop is required for decay heat removal in MODE 6, with water level ≥ 23 ft above the top of the reactor vessel flange. Only one SDC loop is required because the volume of water above the reactor vessel flange provides backup decay heat removal capability. At least one SDC loop must be 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.

The LCO is modified by a Note that allows the required operating SDC loop to be removed from service for up to 1 hour in each 8 hour period, provided no operations are permitted that would cause a reduction of the RCS boron concentration. Boron concentration reduction is prohibited because uniform concentration distribution cannot be ensured without forced circulation. This permits operations such as core mapping or alterations in the vicinity of the reactor vessel hot leg nozzles, and RCS to 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 in operation in MODE 6, with the water level ≥ 23 ft above the top of the reactor vessel flange, to provide decay heat removal. The 23 ft level was selected because it corresponds to the 23 ft requirement established for fuel movement in LCO 3.9.6, "Refueling Water Level."

(continued)

BASES

APPLICABILITY (continued)

Requirements for the SDC System in other MODES are covered by LCOs in Section 3.4, Reactor Coolant System (RCS), and Section 3.5, Emergency Core Cooling Systems (ECCS). SDC loop requirements in MODE 6, with the water level < 23 ft above the top of the reactor vessel flange, are located in LCO 3.9.5, "Shutdown Cooling (SDC) 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 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 taken immediately to suspend loading irradiated fuel assemblies in the core. With no forced circulation cooling, decay heat removal from the core occurs by natural convection to the heat sink provided by the water above the core. A minimum refueling water level 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 loading a fuel assembly, is a prudent action under this condition.

A.3

If SDC loop requirements are not met, actions shall be initiated and continued in order to satisfy SDC loop requirements.

(continued)

BASES

ACTIONS (continued)	**A.4** If SDC loop requirements are not met, all containment penetrations to the outside atmosphere must be closed to prevent fission products, if released by a loss of decay heat event, from escaping the containment building. The 4 hour Completion Time allows fixing most SDC problems without incurring the additional action of violating the containment atmosphere.
SURVEILLANCE REQUIREMENTS	**SR 3.9.4.1** This Surveillance demonstrates that the SDC loop is in operation and circulating reactor coolant. The flow rate is determined by the flow rate necessary to provide sufficient decay heat removal capability and to prevent thermal and boron stratification in the core. The Frequency of 12 hours is sufficient, considering the flow, temperature, pump control, and alarm indications available to the operator in the control room for monitoring the SDC System.
REFERENCES	1. FSAR, Section [ ].

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 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 in MODE 6, with the water level < 23 ft above the top of the reactor vessel flange, to prevent this challenge.

SDC and Coolant Circulation—Low Water Level satisfies Criterion 2 of the NRC Policy Statement.

(continued)

BASES (continued)

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.

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, action shall be immediately 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. 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. An immediate Completion Time is necessary for an operator to initiate corrective actions.

(continued)

BASES

ACTIONS (continued)

B.1

If no SDC loop is in operation or no SDC loops are OPERABLE, 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 in operation or no SDC loops are OPERABLE, action 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.

B.3

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

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

SURVEILLANCE REQUIREMENTS

SR 3.9.5.1

This Surveillance demonstrates that one SDC loop is operating 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. In addition, this Surveillance demonstrates that the other SDC loop is OPERABLE.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.5.1 (continued)

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.

Verification that the required loops are OPERABLE and in operation ensures that loops can be placed in operation as needed, to maintain decay heat and retain forced circulation. The Frequency of 12 hours is considered reasonable, since other administrative controls are available and have proven to be acceptable by operating experience.

SR 3.9.5.2

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

REFERENCES

1. FSAR, Section [ ].

B 3.9 REFUELING OPERATIONS

B 3.9.6 Refueling Water Level

BASES

BACKGROUND

The movement of irradiated fuel assemblies or performance of CORE ALTERATIONS, except during latching and unlatching of control rod drive shafts, within containment requires a minimum water level of 23 ft above the top of the reactor vessel flange. During refueling this maintains sufficient water level in the containment, the refueling canal, the fuel transfer canal, the refueling cavity, and the spent fuel pool. Sufficient water is necessary to retain iodine fission product activity in the water in the event of a fuel handling accident (Refs. 1 and 2). Sufficient iodine activity would be retained to limit offsite doses from the accident to < 25% of 10 CFR 100 limits, as provided by the guidance of Reference 3.

APPLICABLE SAFETY ANALYSES

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

The fuel handling accident analysis inside containment is described in Reference 2. With a minimum water level of 23 ft and a minimum decay time of 72 hours prior to fuel handling, the analysis and test programs demonstrate that the iodine release due to a postulated fuel handling accident is adequately captured by the water and offsite doses are maintained within allowable limits (Ref. 4).

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

(continued)

BASES (continued)

LCO

A minimum refueling water level of 23 ft above the reactor vessel flange is required to ensure that the radiological consequences of a postulated fuel handling accident inside containment are within acceptable limits as provided by the guidance of Reference 3.

APPLICABILITY

LCO 3.9.6 is applicable during CORE ALTERATIONS, except during latching and unlatching of control rod drive shafts, and 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 and A.2

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

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

A.3

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

(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 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, March 23, 1972.

2. FSAR, Section [ ].

3. NUREG-0800, Section 15.7.4.

4. 10 CFR 100.10.

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11. ABSTRACT *(200 words or less)*

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

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