



Tennessee Valley Authority, Sequoyah Nuclear Plant, P.O. Box 2000, Soddy Daisy, Tennessee 37384

October 28, 2020

10 CFR 50.4  
10 CFR 50.71(e)

ATTN: Document Control Desk  
U.S. Nuclear Regulatory Commission  
Washington, D.C. 20555-0001

Sequoyah Nuclear Plant, Units 1 and 2  
Renewed Facility Operating License Nos. DPR-77 and DPR-79  
NRC Docket Nos. 50-327 and 50-328

Subject: **Revisions to the Sequoyah Nuclear Plant Units 1 and 2 Technical Specification Bases**

References: TVA Letter to NRC, "Revisions to the Sequoyah Nuclear Plant Units 1 and 2 Technical Specification Bases," dated May 31, 2019.

Pursuant to the Sequoyah Nuclear Plant (SQN) Technical Specification 5.5.12, "Technical Specifications (TS) Bases Control Program," these changes to the SQN TS Bases are submitted in accordance with 10 CFR 50.71(e). The previous revisions of the SQN TS Bases were submitted in the referenced letter. The enclosure to this letter provides a description of the TS Bases revisions with attachments of the updated pages.

There are no new regulatory commitments contained in this letter. If you have any questions, please contact Mr. Andrew McNeil, Site Licensing Manager (Acting), at (423) 843-8098.

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I certify that I am duly authorized by TVA, and that, to the best of my knowledge and belief, the information contained herein accurately presents changes made since the previous submittal, necessary to reflect information and analyses submitted to the Commission or prepared pursuant to Commission requirements.

Respectfully,



Scott Hunnewell  
Interim Site Vice President  
Sequoyah Nuclear Plant

Enclosure:

Description of Revisions for the Sequoyah Nuclear Plant (SQN), Units 1 and 2  
Technical Specification Bases

cc (Enclosure):

NRC Regional Administrator – Region II  
NRC Senior Resident Inspector – Sequoyah Nuclear Plant

## **ENCLOSURE**

### **DESCRIPTION OF REVISIONS FOR THE SEQUOYAH NUCLEAR PLANT (SQN), UNITS 1 AND 2 TECHNICAL SPECIFICATION BASES**

Revisions 60 to the SQN, Units 1 and 2 Technical Specification (TS) Bases were approved on August 27, 2019, and implemented on October 4, 2019. These Bases revisions are associated with TS Change 17-03, "Unbalanced Voltage Relays," that was approved by NRC on August 27, 2019, for SQN Units 1 and 2 under Amendment Nos. 345 and 339, respectively. The TS amendment added Condition C to Limiting Condition for Operation (LCO) 3.3.5, "Loss of Power (LOP) Diesel Generator (DG) Start Instrumentation," for actions to take when unbalanced voltage relays are determined to be inoperable. The unbalanced voltage relays were added to TS Table 3.3.5-1, "LOP DG Start Instrumentation Function," as a new function for operability.

Revisions 61 to the SQN, Units 1 and 2 TS Bases were approved on May 24, 2018. These TS Bases revisions are for replacement of certain containment isolation valves represented on Table B 3.6.3-1, "Containment Isolation Valve Completion Times." The steam generator blowdown containment isolation sampling valves, which were flow solenoid valves (FSVs), have been replaced with flow control valves (FCVs). The implementation of the Units 1 and 2 TS Bases occurred on November 21, 2019, and September 21, 2020, after physical plant modifications during each unit's respective refueling outage.

A revised effective page listing for the SQN TSs are provided with the Bases change pages associated with the above revisions.

**Attachments:**

1. Sequoyah Nuclear Plant, Unit 1, Technical Specification Bases - Changed Pages
2. Sequoyah Nuclear Plant, Unit 2, Technical Specification Bases - Changed Pages

**ATTACHMENT 1**  
**SEQUOYAH NUCLEAR PLANT, UNIT 1,**  
**TECHNICAL SPECIFICATION BASES**  
**CHANGED PAGES**

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# SEQUOYAH NUCLEAR PLANT

## Technical Specification Bases

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## B 2.0 SAFETY LIMITS (SLs)

### B 2.1.1 Reactor Core

#### BASES

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**BACKGROUND** GDC 10 (Ref. 1) requires that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). This is accomplished by having a departure from nucleate boiling (DNB) design basis, which corresponds to a 95% probability at a 95% confidence level (the 95/95 DNB criterion) that DNB will not occur and by requiring that fuel centerline temperature stays below the melting temperature.

The restrictions of this SL prevent overheating of the fuel cladding (due to departure from nucleate boiling) and overheating of the fuel pellet (centerline fuel melt (CFM)), either of which could result in cladding perforation, which would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady state peak linear heat rate (LHR) below the level at which fuel centerline melting occurs. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime, where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of DNB and the corresponding significant reduction in heat transfer coefficient from the outer surface of the cladding to the reactor coolant water. Inside the steam film, high cladding temperatures are reached, and a cladding water (zirconium water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB. The DNB correlations have been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

To meet the DNB Design Basis, a statistical core design (SCD) process has been used to develop an appropriate statistical DNBR design limit. Uncertainties in plant operating parameters, nuclear and thermal parameters, and fuel fabrication parameters are considered statistically such that there is at least a 95 percent probability at a 95 percent

BASES

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## BACKGROUND (continued)

confidence level that the minimum DNBR for the limiting rod is greater than or equal to the DNBR limit. This DNBR uncertainty derived from the SCD analysis, combined with the applicable DNB critical heat flux correlation limit, establishes the statistical DNBR design limit which must be met in plant safety analysis using values of input parameters without adjustment for uncertainty.

Operation above the maximum local linear heat generation rate for fuel melting could result in excessive fuel pellet temperature and cause melting of the fuel at its centerline. Fuel centerline melting occurs when the local LHR, or power peaking, in a region of the fuel is high enough to cause the fuel centerline temperature to reach the melting point of the fuel. Expansion of the pellet upon centerline melting may cause the pellet to stress the cladding to the point of failure, allowing an uncontrolled release of activity to the reactor coolant. The melting point of uranium dioxide varies slightly with burnup. As uranium is depleted and fission products produced, the net effect is a decrease in the melting point. Fuel centerline temperature is not a directly measurable parameter during operation. The maximum local fuel pin centerline temperature is maintained by limiting the local linear heat generation rate in the fuel. The local linear heat generation rate in the fuel is limited so that the maximum fuel centerline temperature will not exceed the acceptance criteria in the safety analysis.

The curves provided in Figure 2.1.1-1 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature for which the minimum DNBR is no less than the safety analysis DNBR limit, or the average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid.

These lines are bounding for all fuel types. The curves provided in Figure 2.1.1-1 are based upon enthalpy rise hot channel factors that result in acceptable DNBR performance of each fuel type. Acceptable DNBR performance is assured by operation within the DNB-based Limiting Safety Limit System Settings (Reactor Trip System trip limits). The plant trip setpoints are verified to be less than the limits defined by the safety limit lines provided in Figure 2.1.1-1 converted from power to delta-temperature and adjusted for uncertainty.

The limiting heat flux conditions for DNB are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion assuming the axial power imbalance or Delta-I ( $\Delta I$ ) is within the limits of the  $f_1(\Delta I)$  function of the Overtemperature Delta Temperature trip. When the axial power imbalance exceeds the tolerance (or deadband) of the  $f_1(\Delta I)$  trip reset function, the Overtemperature Delta Temperature trip setpoint is reduced by the values

BASES

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BACKGROUND (continued)

in the COLR to provide protection required by the core safety limits.

Similarly, the limiting linear heat generation rate conditions for CFM are higher than those calculated for the range of all control rods from fully withdrawn to the maximum allowable control rod insertion assuming the axial power imbalance or Delta-I ( $\Delta I$ ) is within the limits of the  $f_2(\Delta I)$  function of the Overpower-Delta Temperature trip. When the axial power imbalance exceeds the tolerance (or deadband) of the  $f_2(\Delta I)$  trip reset function, the Overpower-Delta Temperature trip setpoint is reduced by the values specified in the COLR to provide protection required by the core safety limits.

The proper functioning of the Reactor Protection System (RPS) and steam generator safety valves prevents violation of the reactor core SLs.

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APPLICABLE  
SAFETY  
ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and AOOs. The reactor core SLs are established to preclude violation of the following fuel design criteria:

- a. There must be at least 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB and
- b. The hot fuel pellet in the core must not experience centerline fuel melting.

The Reactor Trip System setpoints (Ref. 2), in combination with the LCOs, are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System (RCS) temperature, pressure, RCS Flow,  $\Delta I$ , and THERMAL POWER level that would result in a departure from nucleate boiling ratio (DNBR) of less than the DNBR limit and preclude the existence of flow instabilities.

Automatic enforcement of these reactor core SLs is provided by the appropriate operation of the RPS and the steam generator safety valves.

The SLs represent a design requirement for establishing the RPS trip setpoints identified previously. LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," or the assumed initial conditions of the safety analyses (as indicated in the UFSAR, Ref. 2) provide more restrictive limits to ensure that the SLs are not exceeded.

## BASES

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**SAFETY LIMITS** Figure 2.1.1-1 shows the loci of points of THERMAL POWER, RCS pressure, and average temperature for which the minimum DNBR is not less than the safety analyses limit, that fuel centerline temperature remains below melting, that the average enthalpy in the hot leg is less than or equal to the enthalpy of saturated liquid, or that the exit quality is within the limits defined by the DNBR correlation.

The reactor core SLs are established to preclude violation of the following fuel design criteria:

- a. There must be at least a 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB and
- b. There must be at least a 95% probability at a 95% confidence level that the hot fuel pellet in the core does not experience centerline fuel melting.

The reactor core SLs are used to define the various RPS functions such that the above criteria are satisfied during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). To ensure that the RPS precludes the violation of the above criteria, additional criteria are applied to the Overtemperature and Overpower  $\Delta T$  reactor trip functions. That is, it must be demonstrated that the average enthalpy in the hot leg is less than or equal to the saturation enthalpy and that the core exit quality is within the limits defined by the DNBR correlation. Appropriate functioning of the RPS ensures that for variations in the THERMAL POWER, RCS Pressure, RCS average temperature, RCS flow rate, and  $\Delta I$  that the reactor core SLs will be satisfied during steady state operation, normal operational transients, and AOOs.

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**APPLICABILITY** SL 2.1.1 only applies in MODES 1 and 2 because these are the only MODES in which the reactor is critical. Automatic protection functions are required to be OPERABLE during MODES 1 and 2 to ensure operation within the reactor core SLs. The steam generator safety valves or automatic protection actions serve to prevent RCS heatup to the reactor core SL conditions or to initiate a reactor trip function, which forces the unit into MODE 3. Setpoints for the reactor trip functions are specified in LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation." In MODES 3, 4, 5, and 6, Applicability is not required since the reactor is not generating significant THERMAL POWER.

BASES

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SAFETY LIMIT  
VIOLATIONS

The following SL violation responses are applicable to the reactor core SLs. If SL 2.1.1 is violated, the requirement to go to MODE 3 places the unit in a MODE in which this SL is not applicable.

The allowed Completion Time of 1 hour recognizes the importance of bringing the unit to a MODE of operation where this SL is not applicable, and reduces the probability of fuel damage.

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REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.
  2. UFSAR, Section 7.2.
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## B 2.0 SAFETY LIMITS (SLs)

### B 2.1.2 Reactor Coolant System (RCS) Pressure SL

#### BASES

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**BACKGROUND** The SL on RCS pressure protects the integrity of the RCS against overpressurization. In the event of fuel cladding failure, fission products are released into the reactor coolant. The RCS then serves as the primary barrier in preventing the release of fission products into the atmosphere. By establishing an upper limit on RCS pressure, the continued integrity of the RCS is ensured. According to 10 CFR 50, Appendix A, GDC 14, "Reactor Coolant Pressure Boundary," and GDC 15, "Reactor Coolant System Design" (Ref. 1), the reactor coolant pressure boundary (RCPB) design conditions are not to be exceeded during normal operation and anticipated operational occurrences (AOOs). Also, in accordance with GDC 28, "Reactivity Limits" (Ref. 1), reactivity accidents, including rod ejection, do not result in damage to the RCPB greater than limited local yielding.

The design pressure of the RCS is 2500 psia. During normal operation and AOOs, RCS pressure is limited from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME Code (Ref. 2). To ensure system integrity, all RCS components are hydrostatically tested at 125% of design pressure, according to the ASME Code requirements prior to initial operation when there is no fuel in the core. Following inception of unit operation, RCS components shall be pressure tested, in accordance with the requirements of ASME Code, Section XI (Ref. 3).

Overpressurization of the RCS could result in a breach of the RCPB. If such a breach occurs in conjunction with a fuel cladding failure, fission products could enter the containment atmosphere, raising concerns relative to limits on radioactive releases specified in 10 CFR 100, "Reactor Site Criteria" (Ref. 4).

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**APPLICABLE SAFETY ANALYSES** The RCS pressurizer safety valves, the main steam safety valves (MSSVs), and the reactor high pressure trip have settings established to ensure that the RCS pressure SL will not be exceeded.

The RCS pressurizer safety valves are sized to prevent system pressure from exceeding the design pressure by more than 10%, as specified in Section III of the ASME Code for Nuclear Power Plant Components (Ref. 2). The transient that establishes the required relief capacity, and hence valve size requirements and lift settings, is a complete loss of

BASES

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APPLICABLE SAFETY ANALYSES (continued)

external load without a direct reactor trip. During the transient, no control actions are assumed, except that the safety valves on the secondary plant are assumed to open when the steam pressure reaches the secondary plant safety valve settings, and nominal feedwater supply is maintained.

The Reactor Trip System setpoints (Ref. 5), together with the settings of the MSSVs, provide pressure protection for normal operation and AOOs. The reactor high pressure trip setpoint is specifically set to provide protection against overpressurization (Ref. 5). The safety analyses for both the high pressure trip and the RCS pressurizer safety valves are performed using conservative assumptions relative to pressure control devices.

More specifically, no credit is taken for operation of any of the following:

- a. Pressurizer power operated relief valves (PORVs);
- b. Steam Dump System;
- c. Reactor Control System;
- d. Pressurizer Level Control System; or
- e. Pressurizer spray valve.

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SAFETY LIMITS

The maximum transient pressure allowed in the RCS pressure vessel under the ASME Code, Section III, is 110% of design pressure. The maximum transient pressure allowed in the RCS piping, valves, and fittings under USAS, Section B31.1 (Ref. 6) is 120% of design pressure. The most limiting of these two allowances is the 110% of design pressure; therefore, the SL on maximum allowable RCS pressure is 2735 psig.

BASES

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APPLICABILITY SL 2.1.2 applies in MODES 1, 2, 3, 4, and 5 because this SL could be approached or exceeded in these MODES due to overpressurization events. The SL is not applicable in MODE 6 because the reactor vessel head closure bolts are not fully tightened, making it unlikely that the RCS can be pressurized.

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SAFETY LIMIT VIOLATIONS If the RCS pressure SL is violated when the reactor is in MODE 1 or 2, the requirement is to restore compliance and be in MODE 3 within 1 hour.

Exceeding the RCS pressure SL may cause immediate RCS failure and create a potential for radioactive releases in excess of 10 CFR 100, "Reactor Site Criteria," limits (Ref. 4).

The allowable Completion Time of 1 hour recognizes the importance of reducing power level to a MODE of operation where the potential for challenges to safety systems is minimized.

If the RCS pressure SL is exceeded in MODE 3, 4, or 5, RCS pressure must be restored to within the SL value within 5 minutes. Exceeding the RCS pressure SL in MODE 3, 4, or 5 is more severe than exceeding this SL in MODE 1 or 2, since the reactor vessel temperature may be lower and the vessel material, consequently, less ductile. As such, pressure must be reduced to less than the SL within 5 minutes. The action does not require reducing MODES, since this would require reducing temperature, which would compound the problem by adding thermal gradient stresses to the existing pressure stress.

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- REFERENCES
1. 10 CFR 50, Appendix A, GDC 14, GDC 15, and GDC 28.
  2. ASME, Boiler and Pressure Vessel Code, Section III, Article NB-7000, 1971.
  3. ASME, Boiler and Pressure Vessel Code, Section XI, Article IWX-5000.
  4. 10 CFR 100.
  5. UFSAR, Section 7.2.
  6. USAS B31.1, Standard Code for Pressure Piping, American Society of Mechanical Engineers, 1967.
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## B 3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

### BASES

LCOs	LCO 3.0.1 through LCO 3.0.9 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.
LCO 3.0.1	LCO 3.0.1 establishes the Applicability statement within each individual Specification as the requirement for when the LCO is required to be met (i.e., when the unit is in the MODES or other specified conditions of the Applicability statement of each Specification).
LCO 3.0.2	<p>LCO 3.0.2 establishes that upon discovery of a failure to meet an LCO, the associated ACTIONS shall be met. The Completion Time of each Required Action for an ACTIONS Condition is applicable from the point in time that an ACTIONS Condition is entered. The Required Actions establish those remedial measures that must be taken within specified Completion Times when the requirements of an LCO are not met. This Specification establishes that:</p> <ol style="list-style-type: none"> <li>a. Completion of the Required Actions within the specified Completion Times constitutes compliance with a Specification; and</li> <li>b. Completion of the Required Actions is not required when an LCO is met within the specified Completion Time, unless otherwise specified.</li> </ol> <p>There are two basic types of Required Actions. The first type of Required Action specifies a time limit in which the LCO must be met. This time limit is the Completion Time to restore an inoperable system or component to OPERABLE status or to restore variables to within specified limits. If this type of Required Action is not completed within the specified Completion Time, a shutdown may be required to place the unit in a MODE or condition in which the Specification is not applicable. (Whether stated as a Required Action or not, correction of the entered Condition is an action that may always be considered upon entering ACTIONS.) The second type of Required Action specifies the remedial measures that permit continued operation of the unit that is not further restricted by the Completion Time. In this case, compliance with the Required Actions provides an acceptable level of safety for continued operation.</p> <p>Completing the Required Actions is not required when an LCO is met or is no longer applicable, unless otherwise stated in the individual Specifications.</p>

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LCO 3.0.2 (continued)

The nature of some Required Actions of some Conditions necessitates that, once the Condition is entered, the Required Actions must be completed even though the associated Conditions no longer exist. The individual LCO's ACTIONS specify the Required Actions where this is the case. An example of this is in LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits."

The Completion Times of the Required Actions are also applicable when a system or component is removed from service intentionally. The reasons for intentionally relying on the ACTIONS include, but are not limited to, performance of Surveillances, preventive maintenance, corrective maintenance, or investigation of operational problems. Entering ACTIONS for these reasons must be done in a manner that does not compromise safety. Intentional entry into ACTIONS should not be made for operational convenience. Additionally, if intentional entry into ACTIONS would result in redundant equipment being inoperable, alternatives should be used instead. Doing so limits the time both subsystems/trains of a safety function are inoperable and limits the time conditions exist which may result in LCO 3.0.3 being entered. Individual Specifications may specify a time limit for performing an SR when equipment is removed from service or bypassed for testing. In this case, the Completion Times of the Required Actions are applicable when this time limit expires, if the equipment remains removed from service or bypassed.

When a change in MODE or other specified condition is required to comply with Required Actions, the unit may enter a MODE or other specified condition in which another Specification becomes applicable. In this case, the Completion Times of the associated Required Actions would apply from the point in time that the new Specification becomes applicable, and the ACTIONS Condition(s) are entered.

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LCO 3.0.3

LCO 3.0.3 establishes the actions that must be implemented when an LCO is not met and:

- a. An associated Required Action and Completion Time is not met and no other Condition applies; or
- b. The condition of the unit is not specifically addressed by the associated ACTIONS. This means that no combination of Conditions stated in the ACTIONS can be made that exactly corresponds to the actual condition of the unit. Sometimes, possible combinations of Conditions are such that entering LCO 3.0.3 is warranted; in such cases, the ACTIONS specifically state a Condition corresponding to such combinations and also that LCO 3.0.3 be entered immediately.

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### LCO 3.0.3 (continued)

This Specification delineates the time limits for placing the unit in a safe MODE or other specified condition when operation cannot be maintained within the limits for safe operation as defined by the LCO and its ACTIONS. It is not intended to be used as an operational convenience that permits routine voluntary removal of redundant systems or components from service in lieu of other alternatives that would not result in redundant systems or components being inoperable.

Upon entering LCO 3.0.3, 1 hour is allowed to prepare for an orderly shutdown before initiating a change in unit operation. This includes time to permit the operator to coordinate the reduction in electrical generation with the load dispatcher to ensure the stability and availability of the electrical grid. The time limits specified to reach lower MODES of operation permit the shutdown to proceed in a controlled and orderly manner that is well within the specified maximum cooldown rate and within the capabilities of the unit, assuming that only the minimum required equipment is OPERABLE. This reduces thermal stresses on components of the Reactor Coolant System and the potential for a plant upset that could challenge safety systems under conditions to which this Specification applies. The use and interpretation of specified times to complete the actions of LCO 3.0.3 are consistent with the discussion of Section 1.3, Completion Times.

A unit shutdown required in accordance with LCO 3.0.3 may be terminated and LCO 3.0.3 exited if any of the following occurs:

- a. The LCO is now met;
- b. A Condition exists for which the Required Actions have now been performed; or
- c. ACTIONS exist that do not have expired Completion Times. These Completion Times are applicable from the point in time that the Condition is initially entered and not from the time LCO 3.0.3 is exited.

The time limits of LCO 3.0.3 allow 37 hours for the unit to be in MODE 5 when a shutdown is required during MODE 1 operation. If the unit is in a lower MODE of operation when a shutdown is required, the time limit for reaching the next lower MODE applies. If a lower MODE is reached in less time than allowed, however, the total allowable time to reach MODE 5, or other applicable MODE, is not reduced. For example, if MODE 3 is reached in 2 hours, then the time allowed for reaching MODE 4 is the next 11 hours, because the total time for reaching

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LCO 3.0.3 (continued)

MODE 4 is not reduced from the allowable limit of 13 hours. Therefore, if remedial measures are completed that would permit a return to MODE 1, a penalty is not incurred by having to reach a lower MODE of operation in less than the total time allowed.

In MODES 1, 2, 3, and 4, LCO 3.0.3 provides actions for Conditions not covered in other Specifications. The requirements of LCO 3.0.3 do not apply in MODES 5 and 6 because the unit is already in the most restrictive Condition required by LCO 3.0.3. The requirements of LCO 3.0.3 do not apply in other specified conditions of the Applicability (unless in MODE 1, 2, 3, or 4) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken.

Exceptions to LCO 3.0.3 are provided in instances where requiring a unit shutdown, in accordance with LCO 3.0.3, would not provide appropriate remedial measures for the associated condition of the unit. An example of this is in LCO 3.7.13, "Spent Fuel Pool Water Level." LCO 3.7.13 has an Applicability of "Whenever irradiated fuel assemblies are in the spent fuel pool." Therefore, this LCO can be applicable in any or all MODES. If the LCO and the Required Actions of LCO 3.7.13 are not met while in MODE 1, 2, or 3, there is no safety benefit to be gained by placing the unit in a shutdown condition. The Required Actions of LCO 3.7.13 of "Suspend all movement of fuel assemblies and crane operations with loads in the fuel storage areas" and "Restore spent fuel pool water level to within limit" are the appropriate Required Actions to complete in lieu of the actions of LCO 3.0.3. These exceptions are addressed in the individual Specifications.

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LCO 3.0.4

LCO 3.0.4 establishes limitations on changes in MODES or other specified conditions in the Applicability when an LCO is not met. It allows placing the unit in a MODE or other specified condition stated in that Applicability (e.g., the Applicability desired to be entered) when unit conditions are such that the requirements of the LCO would not be met, in accordance with LCO 3.0.4.a, LCO 3.0.4.b, or LCO 3.0.4.c.

LCO 3.0.4.a allows entry into a MODE or other specified condition in the Applicability with the LCO not met when the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time. Compliance with Required Actions that permit continued operation of the unit for an unlimited period of time in a MODE or other specified condition provides an acceptable level of safety for continued operation. This is without regard to the status of the unit before or after the MODE change. Therefore, in such cases, entry into a MODE or other specified condition

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### LCO 3.0.4 (continued)

in the Applicability may be made in accordance with the provisions of the Required Actions.

LCO 3.0.4.b allows entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering the MODE or other specified condition in the Applicability, and establishment of risk management actions, if appropriate.

The risk assessment may use quantitative, qualitative, or blended approaches, and the risk assessment will be conducted using the plant program, procedures, and criteria in place to implement 10 CFR 50.65(a)(4), which requires that risk impacts of maintenance activities be assessed and managed. The risk assessment, for the purposes of LCO 3.0.4.b, must take into account all inoperable Technical Specification equipment regardless of whether the equipment is included in the normal 10 CFR 50.65(a)(4) risk assessment scope. The risk assessments will be conducted using the procedures and guidance endorsed by Regulatory Guide 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants." Regulatory Guide 1.182 endorses the guidance in Section 11 of NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." These documents address general guidance for conduct of the risk assessment, quantitative and qualitative guidelines for establishing risk management actions, and example risk management actions. These include actions to plan and conduct other activities in a manner that controls overall risk, increased risk awareness by shift and management personnel, actions to reduce the duration of the condition, actions to minimize the magnitude of risk increases (establishment of backup success paths or compensatory measures), and determination that the proposed MODE change is acceptable. Consideration should also be given to the probability of completing restoration such that the requirements of the LCO would be met prior to the expiration of ACTIONS Completion Times that would require exiting the Applicability.

LCO 3.0.4.b may be used with single, or multiple systems and components unavailable. NUMARC 93-01 provides guidance relative to consideration of simultaneous unavailability of multiple systems and components.

The results of the risk assessment shall be considered in determining the acceptability of entering the MODE or other specified condition in the Applicability, and any corresponding risk management actions. The LCO 3.0.4.b risk assessments do not have to be documented.

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## LCO 3.0.4 (continued)

The Technical Specifications allow continued operation with equipment unavailable in MODE 1 for the duration of the Completion Time. Since this is allowable, and since in general the risk impact in that particular MODE bounds the risk of transitioning into and through the applicable MODES or other specified conditions in the Applicability of the LCO, the use of the LCO 3.0.4.b allowance should be generally acceptable, as long as the risk is assessed and managed as stated above. However, there is a small subset of systems and components that have been determined to be more important to risk and use of the LCO 3.0.4.b allowance is prohibited. The LCOs governing these systems and components contain Notes prohibiting the use of LCO 3.0.4.b by stating that LCO 3.0.4.b is not applicable.

LCO 3.0.4.c allows entry into a MODE or other specified condition in the Applicability with the LCO not met based on a Note in the Specification which states LCO 3.0.4.c is applicable. These specific allowances permit entry into MODES or other specified conditions in the Applicability when the associated ACTIONS to be entered do not provide for continued operation for an unlimited period of time and a risk assessment has not been performed. This allowance may apply to all the ACTIONS or to a specific Required Action of a Specification. The risk assessments performed to justify the use of LCO 3.0.4.b usually only consider systems and components. For this reason, LCO 3.0.4.c is typically applied to Specifications which describe values and parameters (e.g., Containment Air Temperature, Containment Pressure, and Moderator Temperature Coefficient), and may be applied to other Specifications based on NRC plant specific approval.

The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to OPERABLE status before entering an associated MODE or other specified condition in the Applicability.

The provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any unit shutdown. In this context, a unit shutdown is defined as a change in MODE or other specified condition in the Applicability associated with transitioning from MODE 1 to MODE 2, MODE 2 to MODE 3, MODE 3 to MODE 4, and MODE 4 to MODE 5.

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### LCO 3.0.4 (continued)

Upon entry into a MODE or other specified condition in the Applicability with the LCO not met, LCO 3.0.1 and LCO 3.0.2 require entry into the applicable Conditions and Required Actions until the Condition is resolved, until the LCO is met, or until the unit is not within the Applicability of the Technical Specification.

Surveillances do not have to be performed on the associated inoperable equipment (or on variables outside the specified limits), as permitted by SR 3.0.1. Therefore, utilizing LCO 3.0.4 is not a violation of SR 3.0.1 or SR 3.0.4 for any Surveillances that have not been performed on inoperable equipment. However, SRs must be met to ensure OPERABILITY prior to declaring the associated equipment OPERABLE (or variable within limits) and restoring compliance with the affected LCO.

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### LCO 3.0.5

LCO 3.0.5 establishes the allowance for restoring equipment to service under administrative controls when it has been removed from service or declared inoperable to comply with ACTIONS. The sole purpose of this Specification is to provide an exception to LCO 3.0.2 (e.g., to not comply with the applicable Required Action(s)) to allow the performance of required testing to demonstrate:

- a. The OPERABILITY of the equipment being returned to service; or
- b. The OPERABILITY of other equipment.

The administrative controls ensure the time the equipment is returned to service in conflict with the requirements of the ACTIONS is limited to the time absolutely necessary to perform the required testing to demonstrate OPERABILITY. This Specification does not provide time to perform any other preventive or corrective maintenance.

An example of demonstrating the OPERABILITY of the equipment being returned to service is reopening a containment isolation valve that has been closed to comply with Required Actions and must be reopened to perform the required testing.

An example of demonstrating the OPERABILITY of other equipment is taking an inoperable channel or trip system out of the tripped condition to prevent the trip function from occurring during the performance of required testing on another channel in the other trip system. A similar example of demonstrating the OPERABILITY of other equipment is taking an inoperable channel or trip system out of the tripped condition to permit the logic to function and indicate the appropriate response during the performance of required testing on another channel in the same trip system.

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### LCO 3.0.6

LCO 3.0.6 establishes an exception to LCO 3.0.2 for supported systems that have a support system LCO specified in the Technical Specifications (TS). This exception is provided because LCO 3.0.2 would require that the Conditions and Required Actions of the associated inoperable supported system LCO be entered solely due to the inoperability of the support system. This exception is justified because the actions that are required to ensure the unit is maintained in a safe condition are specified in the support system LCO's Required Actions. These Required Actions may include entering the supported system's Conditions and Required Actions or may specify other Required Actions.

When a support system is inoperable and there is an LCO specified for it in the TS, the supported system(s) are required to be declared inoperable if determined to be inoperable as a result of the support system inoperability. However, it is not necessary to enter into the supported systems' Conditions and Required Actions unless directed to do so by the support system's Required Actions. The potential confusion and inconsistency of requirements related to the entry into multiple support and supported systems' LCOs' Conditions and Required Actions are eliminated by providing all the actions that are necessary to ensure the unit is maintained in a safe condition in the support system's Required Actions.

However, there are instances where a support system's Required Action may either direct a supported system to be declared inoperable or direct entry into Conditions and Required Actions for the supported system. This may occur immediately or after some specified delay to perform some other Required Action. Regardless of whether it is immediate or after some delay, when a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.

Specification 5.5.13, "Safety Function Determination Program (SFDP)," ensures loss of safety function is detected and appropriate actions are taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other limitations, remedial actions, or compensatory actions may be identified as a result of the support system inoperability and corresponding exception to entering supported system Conditions and Required Actions. The SFDP implements the requirements of LCO 3.0.6.

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### LCO 3.0.6 (continued)

The following examples use Figure B 3.0-1 to illustrate loss of safety function conditions that may result when a TS support system is inoperable. In this figure, the fifteen systems that comprise Train A are independent and redundant to the fifteen systems that comprise Train B. To correctly use the figure to illustrate the SFDP provisions for a cross train check, the figure establishes a relationship between support and supported systems as follows: the figure shows System 1 as a support system for System 2 and System 3; System 2 as a support system for System 4 and System 5; and System 4 as a support system for System 8 and System 9. Specifically, a loss of safety function may exist when a support system is inoperable and:

- a. A system redundant to system(s) supported by the inoperable support system is also inoperable (EXAMPLE B 3.0.6-1);
- b. A system redundant to system(s) in turn supported by the inoperable supported system is also inoperable (EXAMPLE B 3.0.6-2); or
- c. A system redundant to support system(s) for the supported systems (a) and (b) above is also inoperable (EXAMPLE B 3.0.6-3).

For the following examples, refer to Figure B 3.0-1.

#### EXAMPLE B 3.0.6-1

If System 2 of Train A is inoperable and System 5 of Train B is inoperable, a loss of safety function exists in Systems 5, 10, and 11.

#### EXAMPLE B 3.0.6-2

If System 2 of Train A is inoperable, and System 11 of Train B is inoperable, a loss of safety function exists in System 11.

#### EXAMPLE B 3.0.6-3

If System 2 of Train A is inoperable, and System 1 of Train B is inoperable, a loss of safety function exists in Systems 2, 4, 5, 8, 9, 10 and 11.

If an evaluation determines that a loss of safety function exists, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

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LCO 3.0.6 (continued)

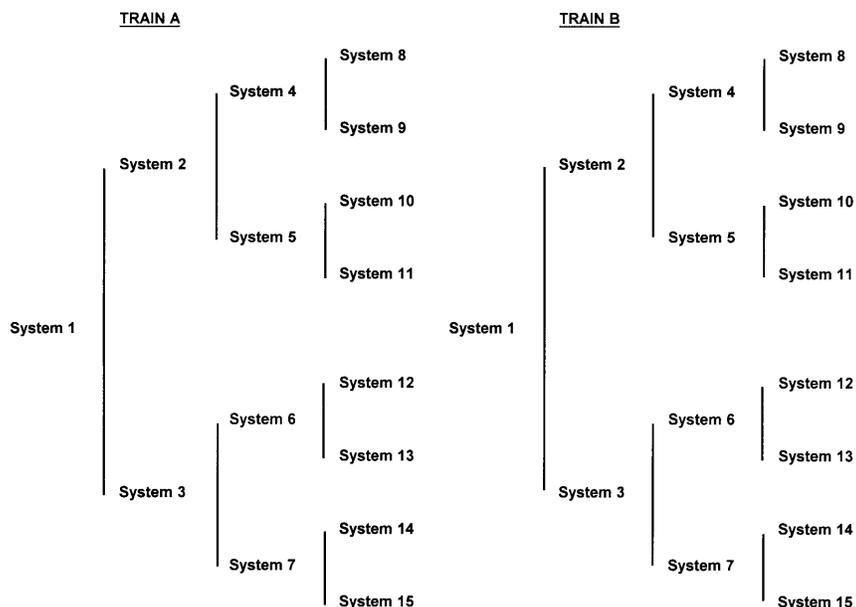


Figure B 3.0-1  
Configuration of Trains and Systems

This loss of safety function does not require the assumption of additional single failures or loss of offsite power. Since operations are being restricted in accordance with the ACTIONS of the support system, any resulting temporary loss of redundancy or single failure protection is taken into account. Similarly, the ACTIONS for inoperable offsite circuit(s) and inoperable diesel generator(s) provide the necessary restriction for cross train inoperabilities. This explicit cross train verification for inoperable AC electrical power sources also acknowledges that supported system(s) are not declared inoperable solely as a result of inoperability of a normal or emergency electrical power source (refer to the definition of OPERABILITY).

When loss of safety function is determined to exist, and the SFDP requires entry into the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists, consideration must be given to the specific type of function affected. Where a loss of function is solely due to a single Technical Specification support system (e.g., loss of automatic start due to inoperable instrumentation, or loss of pump suction

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LCO 3.0.6 (continued)

source due to low tank level) the appropriate LCO is the LCO for the support system. The ACTIONS for a support system LCO adequately address the inoperabilities of that system without reliance on entering its supported system LCO. When the loss of function is the result of multiple support systems, the appropriate LCO is the LCO for the supported system.

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LCO 3.0.7

There are certain special tests and operations required to be performed at various times over the life of the unit. These special tests and operations are necessary to demonstrate select unit performance characteristics, to perform special maintenance activities, and to perform special evolutions. Test Exception LCO 3.1.8, "PHYSICS TEST Exceptions - Mode 2," allows specified Technical Specification (TS) requirements to be changed to permit performances of these special tests and operations, which otherwise could not be performed if required to comply with the requirements of these TS. Unless otherwise specified, all the other TS requirements remain unchanged. This will ensure all appropriate requirements of the MODE or other specified condition not directly associated with or required to be changed to perform the special test or operation will remain in effect.

The Applicability of a Test Exception LCO represents a condition not necessarily in compliance with the normal requirements of the TS. Compliance with Test Exception LCOs is optional. A special operation may be performed either under the provisions of the appropriate Test Exception LCO or under the other applicable TS requirements. If it is desired to perform the special operation under the provisions of the Test Exception LCO, the requirements of the Test Exception LCO shall be followed.

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LCO 3.0.8

LCO 3.0.8 establishes conditions under which systems are considered to remain capable of performing their intended safety function when associated snubbers are not capable of providing their associated support function(s). This LCO states that the supported system is not considered to be inoperable solely due to one or more snubbers not capable of performing their associated support function(s). This is appropriate because a limited length of time is allowed for maintenance, testing, or repair of one or more snubbers not capable of performing their associated support function(s) and appropriate compensatory measures are specified in the snubber requirements, which are located outside of the Technical Specifications (TS) under licensee control. The snubber requirements do not meet the criteria in 10 CFR 50.36(c)(2)(ii), and, as such, are appropriate for control by the licensee.

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## BASES

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### LCO 3.0.8 (continued)

If the allowed time expires and the snubber(s) are unable to perform their associated support function(s), the affected supported system's LCO(s) must be declared not met and the Conditions and Required Actions entered in accordance with LCO 3.0.2.

LCO 3.0.8.a applies when one or more snubbers are not capable of providing their associated support function(s) to a single train or subsystem of a multiple train or subsystem supported system or to a single train or subsystem supported system. LCO 3.0.8.a allows 72 hours to restore the snubber(s) before declaring the supported system inoperable. The 72 hour Completion Time is reasonable based on the low probability of a seismic event concurrent with an event that would require operation of the supported system occurring while the snubber(s) are not capable of performing their associated support function and due to the availability of the redundant train of the supported system.

LCO 3.0.8.b applies when one or more snubbers are not capable of providing their associated support function(s) to more than one train or subsystem of a multiple train or subsystem supported system. LCO 3.0.8.b allows 12 hours to restore the snubber(s) before declaring the supported system inoperable. The 12 hour Completion Time is reasonable based on the low probability of a seismic event concurrent with an event that would require operation of the supported system occurring while the snubber(s) are not capable of performing their associated support function.

LCO 3.0.8 requires that risk be assessed and managed. Industry and NRC guidance on the implementation of 10 CFR 50.65(a)(4) (the Maintenance Rule) does not address seismic risk. However, use of LCO 3.0.8 should be considered with respect to other plant maintenance activities, and integrated into the existing Maintenance Rule process to the extent possible so that maintenance on any unaffected train or subsystem is properly controlled, and emergent issues are properly addressed. The risk assessment need not be quantified, but may be a qualitative awareness of the vulnerability of systems and components when one or more snubbers are not able to perform their associated support function.

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### LCO 3.0.9

LCO 3.0.9 establishes conditions under which systems described in the Technical Specifications are considered to remain OPERABLE when required barriers are not capable of providing their related support function(s).

Barriers are doors, walls, floor plugs, curbs, hatches, installed structures or components, or other devices, not explicitly described in Technical Specifications, that support the performance of the safety function of systems described in the Technical Specifications. This LCO states that the supported system is not considered to be inoperable solely because required barriers not capable of performing their related support function(s) under the described conditions. LCO 3.0.9 allows 30 days before declaring the supported system(s) inoperable and the LCO(s) associated with the supported system(s) not met. A maximum time is placed on each use of this allowance to ensure that required barriers are restored. However, the allowable duration may be less than the specified maximum time based on the risk assessment.

If the allowed time expires and the barriers are unable to perform their related support function(s), the supported system's LCO(s) must be declared not met and the Conditions and Required Actions entered in accordance with LCO 3.0.2.

This provision does not apply to barriers which support ventilation systems or to fire barriers. The Technical Specifications for ventilation systems provide specific Conditions for inoperable barriers. Fire barriers are addressed by other regulatory requirements and associated plant programs. This provision does not apply to barriers that are not required to support system OPERABILITY (see NRC Regulatory Issue Summary 2001-09, "Control of Hazard Barriers," dated April 2, 2001).

The provisions of LCO 3.0.9 are justified because of the low risk associated with required barriers not being capable of performing their related support function. This provision is based on consideration of the following initiating event categories:

- Loss of coolant accidents;
- High energy line breaks;
- Feedwater line breaks;
- Internal flooding;
- External flooding;
- Turbine missile ejection; and
- Tornado or high wind.

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## LCO 3.0.9 (continued)

The risk impact of the barriers which cannot perform their related support function(s) must be addressed pursuant to the risk assessment and management provision of the Maintenance Rule, 10 CFR 50.65 (a)(4), and the associated implementation guidance, Regulatory Guide 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants." Regulatory Guide 1.182 endorses the guidance in Section 11 of NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." This guidance provides for the consideration of dynamic plant configuration issues, emergent conditions, and other aspects pertinent to plant operation with the barriers unable to perform their related support function(s). These considerations may result in risk management and other compensatory actions being required during the period that barriers are unable to perform their related support function(s).

LCO 3.0.9 may be applied to one or more trains or subsystems of a system supported by barriers that cannot provide their related support function(s), provided that risk is assessed and managed (including consideration of the effects on Large Early Release and from external events). If applied concurrently to more than one train or subsystem of a multiple train or subsystem supported system, the barriers supporting each of these trains or subsystems must provide their related support function(s) for different categories of initiating events. For example, LCO 3.0.9 may be applied for up to 30 days for more than one train of a multiple train supported system if the affected barrier for one train protects against internal flooding and the affected barrier for the other train protects against tornado missiles. In this example, the affected barrier may be the same physical barrier but serve different protection functions for each train.

If during the time that LCO 3.0.9 is being used, the required OPERABLE train or subsystem becomes inoperable, it must be restored to OPERABLE status within 24 hours. Otherwise, the train(s) or subsystem(s) supported by barriers that cannot perform their related support function(s) must be declared inoperable and the associated LCOs declared not met. This 24 hour period provides time to respond to emergent conditions that would otherwise likely lead to entry into LCO 3.0.3 and a rapid plant shutdown, which is not justified given the low probability of an initiating event which would require the barrier(s) not capable of performing their related support function(s). During this 24 hour period, the plant risk associated with the existing conditions is assessed and managed in accordance with 10 CFR 50.65(a)(4).

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## B 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

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SRs	SR 3.0.1 through SR 3.0.4 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated. SR 3.0.2 and SR 3.0.3 apply in Chapter 5 only when invoked by a Chapter 5 Specification.
SR 3.0.1	<p>SR 3.0.1 establishes the requirement that SRs must be met during the MODES or other specified conditions in the Applicability for which the requirements of the LCO apply, unless otherwise specified in the individual SRs. This Specification is to ensure that Surveillances are performed to verify the OPERABILITY of systems and components, and that variables are within specified limits. Failure to meet a Surveillance within the specified Frequency, in accordance with SR 3.0.2, constitutes a failure to meet an LCO. Surveillances may be performed by means of any series of sequential, overlapping, or total steps provided the entire Surveillance is performed within the specified Frequency. Additionally, the definitions related to instrument testing (e.g., CHANNEL CALIBRATION) specify that these tests are performed by means of any series of sequential, overlapping, or total steps.</p> <p>Systems and components are assumed to be OPERABLE when the associated SRs have been met. Nothing in this Specification, however, is to be construed as implying that systems or components are OPERABLE when:</p> <ol style="list-style-type: none"> <li>a. The systems or components are known to be inoperable, although still meeting the SRs; or</li> <li>b. The requirements of the Surveillance(s) are known not to be met between required Surveillance performances.</li> </ol> <p>Surveillances do not have to be performed when the unit is in a MODE or other specified condition for which the requirements of the associated LCO are not applicable, unless otherwise specified. The SRs associated with a test exception are only applicable when the test exception is used as an allowable exception to the requirements of a Specification.</p> <p>Unplanned events may satisfy the requirements (including applicable acceptance criteria) for a given SR. In this case, the unplanned event may be credited as fulfilling the performance of the SR. This allowance includes those SRs whose performance is normally precluded in a given MODE or other specified condition.</p> <p>Surveillances, including Surveillances invoked by Required Actions, do not have to be performed on inoperable equipment because the ACTIONS define the remedial measures that apply. Surveillances have</p>

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SR 3.0.1 (continued)

to be met and performed in accordance with SR 3.0.2, prior to returning equipment to OPERABLE status.

Upon completion of maintenance, appropriate post maintenance testing is required to declare equipment OPERABLE. This includes ensuring applicable Surveillances are not failed and their most recent performance is in accordance with SR 3.0.2. Post maintenance testing may not be possible in the current MODE or other specified conditions in the Applicability due to the necessary unit parameters not having been established. In these situations, the equipment may be considered OPERABLE provided testing has been satisfactorily completed to the extent possible and the equipment is not otherwise believed to be incapable of performing its function. This will allow operation to proceed to a MODE or other specified condition where other necessary post maintenance tests can be completed.

Some examples of this process are:

- a. Auxiliary feedwater (AFW) pump turbine maintenance during refueling that requires testing at steam pressures greater than 800 psi. However, if other appropriate testing is satisfactorily completed, the AFW System can be considered OPERABLE. This allows startup and other necessary testing to proceed until the plant reaches the steam pressure required to perform the testing.
- b. High pressure safety injection (SI) maintenance during shutdown that requires system functional tests at a specified pressure. Provided other appropriate testing is satisfactorily completed, startup can proceed with SI considered OPERABLE. This allows operation to reach the specified pressure to complete the necessary post maintenance testing.

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SR 3.0.2

SR 3.0.2 establishes the requirements for meeting the specified Frequency for Surveillances and any Required Action with a Completion Time that requires the periodic performance of the Required Action on a "once per . . ." interval.

SR 3.0.2 permits a 25 percent extension of the interval specified in the Frequency. This extension facilitates Surveillance scheduling and considers plant operating conditions that may not be suitable for conducting the Surveillance (e.g., transient conditions or other ongoing Surveillance or maintenance activities).

When a Section 5.5, "Programs and Manuals," specification states that the provisions of SR 3.0.2 are applicable, a 25 percent extension of the

BASES

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SR 3.0.2 (continued)

testing interval, whether stated in the specification or incorporated by reference, is permitted.

The 25 percent extension does not significantly degrade the reliability that results from performing the Surveillance at its specified Frequency. This is based on the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the SRs. The exceptions to SR 3.0.2 are those Surveillances for which the 25 percent extension of the interval specified in the Frequency does not apply. These exceptions are stated in the individual Specifications. The requirements of regulations take precedence over the TS. An example of where SR 3.0.2 does not apply is in the Containment Leakage Rate Testing Program. This program establishes testing requirements and Frequencies in accordance with the requirements of regulations. The TS cannot in and of themselves extend a test interval specified in the regulations. As stated in SR 3.0.2, the 25 percent extension also does not apply to the initial portion of a periodic Completion Time that requires performance on a "once per ..." basis. The 25 percent extension applies to each performance after the initial performance. The initial performance of the Required Action, whether it is a particular Surveillance or some other remedial action, is considered a single action with a single Completion Time. One reason for not allowing the 25 percent extension to this Completion Time is that such an action usually verifies that no loss of function has occurred by checking the status of redundant or diverse components or accomplishes the function of the inoperable equipment in an alternative manner.

The provisions of SR 3.0.2 are not intended to be used repeatedly merely as an operational convenience to extend Surveillance intervals (other than those consistent with refueling intervals) or periodic Completion Time intervals beyond those specified.

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SR 3.0.3

SR 3.0.3 establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a Surveillance has not been completed within the specified Frequency. A delay period of up to 24 hours or up to the limit of the specified Frequency, whichever is greater, applies from the point in time that it is discovered that the Surveillance has not been performed in accordance with SR 3.0.2, and not at the time that the specified Frequency was not met.

When a Section 5.5, "Programs and Manuals," specification states that the provisions of SR 3.0.3 are applicable, it permits the flexibility to defer declaring the testing requirement not met in accordance with SR 3.0.3 when the testing has not been completed within the testing interval

BASES

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SR 3.0.3 (continued)

(including the allowance of SR 3.0.2 if invoked by the Section 5.5 specification).

This delay period provides adequate time to complete Surveillances that have been missed. This delay period permits the completion of a Surveillance before complying with Required Actions or other remedial measures that might preclude completion of the Surveillance.

The basis for this delay period includes consideration of unit conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, the safety significance of the delay in completing the required Surveillance, and the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the requirements.

When a Surveillance with a Frequency based not on time intervals, but upon specified unit conditions, operating situations, or requirements of regulations (e.g., prior to entering MODE 1 after each fuel loading, or in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions, etc.) is discovered to not have been performed when specified, SR 3.0.3 allows for the full delay period of up to the specified Frequency to perform the Surveillance. However, since there is not a time interval specified, the missed Surveillance should be performed at the first reasonable opportunity.

SR 3.0.3 provides a time limit for, and allowances for the performance of, Surveillances that become applicable as a consequence of MODE changes imposed by Required Actions.

Failure to comply with specified Frequencies for SRs is expected to be an infrequent occurrence. Use of the delay period established by SR 3.0.3 is a flexibility which is not intended to be used as an operational convenience to extend Surveillance intervals. While up to 24 hours or the limit of the specified Frequency is provided to perform the missed Surveillance, it is expected that the missed Surveillance will be performed at the first reasonable opportunity. The determination of the first reasonable opportunity should include consideration of the impact on plant risk (from delaying the Surveillance as well as any plant configuration changes required or shutting the plant down to perform the Surveillance) and impact on any analysis assumptions, in addition to unit conditions, planning, availability of personnel, and the time required to perform the Surveillance. This risk impact should be managed through the program in place to implement 10 CFR 50.65(a)(4) and its implementation guidance, NRC Regulatory Guide 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants." This Regulatory Guide addresses consideration of temporary and

BASES

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SR 3.0.3 (continued)

aggregate risk impacts, determination of risk management action thresholds, and risk management action up to and including plant shutdown. The missed Surveillance should be treated as an emergent condition as discussed in the Regulatory Guide. The risk evaluation may use quantitative, qualitative, or blended methods. The degree of depth and rigor of the evaluation should be commensurate with the importance of the component. Missed Surveillances for important components should be analyzed quantitatively. If the results of the risk evaluation determine the risk increase is significant, this evaluation should be used to determine the safest course of action. All missed Surveillances will be placed in the licensee's Corrective Action Program.

If a Surveillance is not completed within the allowed delay period, then the equipment is considered inoperable or the variable is considered outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon expiration of the delay period. If a Surveillance is failed within the delay period, then the equipment is inoperable, or the variable is outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon the failure of the Surveillance.

Completion of the Surveillance within the delay period allowed by this Specification, or within the Completion Time of the ACTIONS, restores compliance with SR 3.0.1.

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SR 3.0.4

SR 3.0.4 establishes the requirement that all applicable SRs must be met before entry into a MODE or other specified condition in the Applicability.

This Specification ensures that system and component OPERABILITY requirements and variable limits are met before entry into MODES or other specified conditions in the Applicability for which these systems and components ensure safe operation of the unit. The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to OPERABLE status before entering an associated MODE or other specified condition in the Applicability.

A provision is included to allow entry into a MODE or other specified condition in the Applicability when an LCO is not met due to a Surveillance not being met in accordance with LCO 3.0.4.

However, in certain circumstances, failing to meet an SR will not result in SR 3.0.4 restricting a MODE change or other specified condition change. When a system, subsystem, division, component, device, or variable is inoperable or outside its specified limits, the associated SR(s) are not

BASES

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## SR 3.0.4 (continued)

required to be performed, per SR 3.0.1, which states that surveillances do not have to be performed on inoperable equipment. When equipment is inoperable, SR 3.0.4 does not apply to the associated SR(s) since the requirement for the SR(s) to be performed is removed. Therefore, failing to perform the Surveillance(s) within the specified Frequency does not result in an SR 3.0.4 restriction to changing MODES or other specified conditions of the Applicability. However, since the LCO is not met in this instance, LCO 3.0.4 will govern any restrictions that may (or may not) apply to MODE or other specified condition changes. SR 3.0.4 does not restrict changing MODES or other specified conditions of the Applicability when a Surveillance has not been performed within the specified Frequency, provided the requirement to declare the LCO not met has been delayed in accordance with SR 3.0.3.

The provisions of SR 3.0.4 shall not prevent entry into MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of SR 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any unit shutdown. In this context, a unit shutdown is defined as a change in MODE or other specified condition in the Applicability associated with transitioning from MODE 1 to MODE 2, MODE 2 to MODE 3, MODE 3 to MODE 4, and MODE 4 to MODE 5.

The precise requirements for performance of SRs are specified such that exceptions to SR 3.0.4 are not necessary. The specific time frames and conditions necessary for meeting the SRs are specified in the Frequency, in the Surveillance, or both. This allows performance of Surveillances when the prerequisite condition(s) specified in a Surveillance procedure require entry into the MODE or other specified condition in the Applicability of the associated LCO prior to the performance or completion of a Surveillance. A Surveillance that could not be performed until after entering the LCO's Applicability, would have its Frequency specified such that it is not "due" until the specific conditions needed are met. Alternately, the Surveillance may be stated in the form of a Note, as not required (to be met or performed) until a particular event, condition, or time has been reached. Further discussion of the specific formats of SRs' annotation is found in Section 1.4, Frequency.

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## B 3.1 REACTIVITY CONTROL SYSTEMS

### B 3.1.1 SHUTDOWN MARGIN (SDM)

#### BASES

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BACKGROUND	<p>According to GDC 26 (Ref. 1), the reactivity control systems must be redundant and capable of holding the reactor core subcritical when shut down under cold conditions. Maintenance of the SDM ensures that postulated reactivity events will not damage the fuel.</p> <p>SDM requirements provide sufficient reactivity margin to ensure that acceptable fuel design limits will not be exceeded for normal shutdown and anticipated operational occurrences (AOOs). As such, the SDM defines the degree of subcriticality that would be obtained immediately following the insertion or scram of all shutdown and control rods, assuming that the single rod cluster assembly of highest reactivity worth is fully withdrawn.</p> <p>The system design requires that two independent reactivity control systems be provided, and that one of these systems be capable of maintaining the core subcritical under cold conditions. These requirements are provided by the use of movable control assemblies and soluble boric acid in the Reactor Coolant System (RCS). The Control Rod System can compensate for the reactivity effects of the fuel and water temperature changes accompanying power level changes over the range from full load to no load. In addition, the Control Rod System, together with the boration system, provides the SDM during power operation and is capable of making the core subcritical rapidly enough to prevent exceeding acceptable fuel damage limits, assuming that the rod of highest reactivity worth remains fully withdrawn. The soluble boron system can compensate for fuel depletion during operation and all xenon burnout reactivity changes and maintain the reactor subcritical under cold conditions.</p> <p>During power operation, SDM control is ensured by operating with the shutdown banks fully withdrawn and the control banks within the limits of LCO 3.1.6, "Control Bank Insertion Limits." When the unit is in the shutdown and refueling modes, the SDM requirements are met by means of adjustments to the RCS boron concentration.</p>
APPLICABLE SAFETY ANALYSES	<p>The minimum required SDM is assumed as an initial condition in safety analyses. The safety analysis (Ref. 2) establishes a SDM that ensures specified acceptable fuel design limits are not exceeded for normal operation and AOOs, with the assumption of the highest worth rod stuck out on scram. For MODE 5, the primary safety analysis that relies on the SDM limits is the boron dilution analysis.</p>

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BASES

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## APPLICABLE SAFETY ANALYSES (continued)

The acceptance criteria for the SDM requirements are that specified acceptable fuel design limits are maintained. This is done by ensuring that:

- a. The reactor can be made subcritical from all operating conditions, transients, and Design Basis Events,
- b. The reactivity transients associated with postulated accident conditions are controllable within acceptable limits (departure from nucleate boiling ratio (DNBR), fuel centerline temperature limits for AOOs, and  $\leq 280$  cal/gm fuel energy deposition for the rod ejection accident), and
- c. The reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

The most limiting accident for the SDM requirements is based on a main steam line break (MSLB), as described in the accident analysis (Ref. 2). The increased steam flow resulting from a pipe break in the main steam system causes an increased energy removal from the affected steam generator (SG), and consequently the RCS. This results in a reduction of the reactor coolant temperature. The resultant coolant shrinkage causes a reduction in pressure. In the presence of a negative moderator temperature coefficient, this cooldown causes an increase in core reactivity. As RCS temperature decreases, the severity of an MSLB decreases until the MODE 5 value is reached. The most limiting MSLB, with respect to potential fuel damage before a reactor trip occurs, is a double ended break of a main steam line inside containment initiated at the end of core life. The positive reactivity addition from the moderator temperature decrease will terminate when the affected SG boils dry, thus terminating RCS heat removal and cooldown. Following the MSLB, a post trip return to power may occur; however, no fuel damage occurs as a result of the post trip return to power, and THERMAL POWER does not violate the Safety Limit (SL) requirement of SL 2.1.1.

In addition to the limiting MSLB transient, the SDM requirements must also protect against:

- a. Inadvertent boron dilution,
- b. An uncontrolled rod withdrawal from subcritical or low power condition,

## BASES

## APPLICABLE SAFETY ANALYSES (continued)

- c. Startup of an inactive reactor coolant pump (RCP), and
- d. Rod ejection.

Each of these events is discussed below.

In the boron dilution analysis, the required SDM defines the reactivity difference between an initial subcritical boron concentration and the corresponding critical boron concentration. These values, in conjunction with the configuration of the RCS and the assumed dilution flow rate, directly affect the results of the analysis. This event is most limiting at the beginning of core life, when critical boron concentrations are highest.

Depending on the system initial conditions and reactivity insertion rate, the uncontrolled rod withdrawal transient is terminated by either a high power level trip or an overtemperature  $\Delta T$  trip. In all cases, power level, RCS pressure, linear heat rate, and the DNBR do not exceed allowable limits.

The startup of an inactive RCP will not result in a "cold water" criticality, even if the maximum difference in temperature exists between the SG and the core. The maximum positive reactivity addition that can occur due to an inadvertent RCP start is less than half the minimum required SDM. Startup of an idle RCP cannot, therefore, produce a return to power from the hot standby condition.

The ejection of a control rod rapidly adds reactivity to the reactor core, causing both the core power level and heat flux to increase with corresponding increases in reactor coolant temperatures and pressure. The ejection of a rod also produces a time dependent redistribution of core power.

SDM satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii). Even though it is not directly observed from the control room, SDM is considered an initial condition process variable because it is periodically monitored to ensure that the unit is operating within the bounds of accident analysis assumptions.

## LCO

SDM is a core design condition that can be ensured during operation through control rod positioning (control and shutdown banks) and through the soluble boron concentration.

The MSLB (Ref. 2) and the boron dilution (Ref. 3) accidents are the most limiting analyses that establish the SDM value of the LCO. For MSLB accidents, if the LCO is violated, there is a potential to exceed the DNBR limit and to exceed 10 CFR 100, "Reactor Site Criteria," limits (Ref. 4).

BASES

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## LCO (continued)

For the boron dilution accident, if the LCO is violated, the minimum required time assumed for operator action to terminate dilution may no longer be applicable.

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## APPLICABILITY

In MODE 2 with  $k_{\text{eff}} < 1.0$  and in MODES 3, 4, and 5, the SDM requirements are applicable to provide sufficient negative reactivity to meet the assumptions of the safety analyses discussed above. In MODE 6, the shutdown reactivity requirements are given in LCO 3.9.1, "Boron Concentration." In MODES 1 and 2, SDM is ensured by complying with LCO 3.1.5, "Shutdown Bank Insertion Limits," and LCO 3.1.6.

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## ACTIONS

A.1

If the SDM requirements are not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. It is assumed that boration will be continued until the SDM requirements are met.

In the determination of the required combination of boration flow rate and boron concentration, there is no unique requirement that must be satisfied. Since it is imperative to raise the boron concentration of the RCS as soon as possible, the boron concentration should be a highly concentrated solution, such as that normally found in the boric acid tank, or the refueling water storage tank. The operator should borate with the best source available for the plant conditions.

In determining the boration flow rate, the time in core life must be considered. For instance, the most difficult time in core life to increase the RCS boron concentration is at the beginning of cycle when the boron concentration may approach or exceed 2000 ppm. Assuming that a value of 1%  $\Delta k/k$  (1000 pcm) must be recovered and a boration flow rate of 50 gpm, it is possible to increase the boron concentration of the RCS by 147 ppm in approximately 46 minutes. If a boron worth of 6.8 pcm/ppm is assumed, this combination will increase the SDM by 1%  $\Delta k/k$  or 1000 pcm. These boration parameters represent Sequoyah typical values and are provided for the purpose of offering a specific example.

BASES

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SURVEILLANCE  
REQUIREMENTSSR 3.1.1.1

In MODES 1 and 2 with  $k_{\text{eff}} \geq 1.0$ , SDM is verified by observing that the requirements of LCO 3.1.5 and LCO 3.1.6 are met. In the event that a rod is known to be untrippable, however, SDM verification must account for the worth of the untrippable rod as well as another rod of maximum worth.

In MODE 2 with  $k_{\text{eff}} < 1.0$  and in MODES 3, 4, and 5, the SDM is verified by performing a reactivity balance calculation, considering the listed reactivity effects:

- a. RCS boron concentration,
- b. Control bank position,
- c. RCS average temperature,
- d. Fuel burnup based on gross thermal energy generation,
- e. Xenon concentration,
- f. Samarium concentration, and
- g. Isothermal temperature coefficient (ITC).

Using the ITC accounts for Doppler reactivity in this calculation because the reactor is subcritical, and the fuel temperature will be changing at the same rate as the RCS.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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## REFERENCES

1. 10 CFR 50, Appendix A, GDC 26.
  2. UFSAR, Section 15.4.2.
  3. UFSAR, Section 15.2.4.
  4. 10 CFR 100.
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## B 3.1 REACTIVITY CONTROL SYSTEMS

### B 3.1.2 Core Reactivity

#### BASES

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**BACKGROUND** According to GDC 26, GDC 28, and GDC 29 (Ref. 1), reactivity shall be controllable, such that subcriticality is maintained under cold conditions, and acceptable fuel design limits are not exceeded during normal operation and anticipated operational occurrences. Therefore, reactivity balance is used as a measure of the predicted versus measured core reactivity during power operation. The periodic confirmation of core reactivity is necessary to ensure that Design Basis Accident (DBA) and transient safety analyses remain valid. A large reactivity difference could be the result of unanticipated changes in fuel, control rod worth, or operation at conditions not consistent with those assumed in the predictions of core reactivity, and could potentially result in a loss of SDM or violation of acceptable fuel design limits. Comparing predicted versus measured core reactivity validates the nuclear methods used in the safety analysis and supports the SDM demonstrations (LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") in ensuring the reactor can be brought safely to cold, subcritical conditions.

When the reactor core is critical or in normal power operation, a reactivity balance exists and the net reactivity is zero. A comparison of predicted and measured reactivity is convenient under such a balance, since parameters are being maintained relatively stable under steady state power conditions. The positive reactivity inherent in the core design is balanced by the negative reactivity of the control components, thermal feedback, neutron leakage, and materials in the core that absorb neutrons, such as burnable absorbers producing zero net reactivity. Excess reactivity can be inferred from the boron letdown curve (or critical boron curve), which provides an indication of the soluble boron concentration in the Reactor Coolant System (RCS) versus cycle burnup. Periodic measurement of the RCS boron concentration for comparison with the predicted value with specific variables (such as rod height, temperature, pressure, and power), provides a convenient method of ensuring that core reactivity is within design expectations and that the calculational models used to generate the safety analysis are adequate.

In order to achieve the required fuel cycle energy output, the uranium enrichment, in the new fuel loading and in the fuel remaining from the previous cycle, provides excess positive reactivity beyond that required to sustain steady state operation throughout the cycle. When the reactor is critical at RTP and moderator temperature, the excess positive reactivity is compensated by burnable absorbers (if any), control rods, whatever neutron poisons (mainly xenon and samarium) are present in the fuel, and the RCS boron concentration.

## BASES

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### BACKGROUND (continued)

When the core is producing THERMAL POWER, the fuel is being depleted and excess reactivity is decreasing. As the fuel depletes, the RCS boron concentration is reduced to decrease negative reactivity and maintain constant THERMAL POWER. The boron letdown curve is based on steady state operation at RTP. Therefore, deviations from the predicted boron letdown curve may indicate deficiencies in the design analysis, deficiencies in the calculational models, or abnormal core conditions, and must be evaluated.

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### APPLICABLE SAFETY ANALYSES

The acceptance criteria for core reactivity are that the reactivity balance limit ensures plant operation is maintained within the assumptions of the safety analyses.

Accurate prediction of core reactivity is either an explicit or implicit assumption in the accident analysis evaluations. Every accident evaluation (Ref. 2) is, therefore, dependent upon accurate evaluation of core reactivity. In particular, SDM and reactivity transients, such as control rod withdrawal accidents or rod ejection accidents, are very sensitive to accurate prediction of core reactivity. These accident analysis evaluations rely on computer codes that have been qualified against available test data, operating plant data, and analytical benchmarks. Monitoring reactivity balance additionally ensures that the nuclear methods provide an accurate representation of the core reactivity.

Design calculations and safety analyses are performed for each fuel cycle for the purpose of predetermining reactivity behavior and the RCS boron concentration requirements for reactivity control during fuel depletion.

The comparison between measured and predicted initial core reactivity provides a normalization for the calculational models used to predict core reactivity. If the measured and predicted RCS boron concentrations for identical core conditions at beginning of cycle life (BOL) do not agree, then the assumptions used in the reload cycle design analysis or the calculational models used to predict soluble boron requirements may not be accurate. If reasonable agreement between measured and predicted core reactivity exists at BOL, then the prediction may be normalized to the measured boron concentration. Thereafter, any significant deviations in the measured boron concentration from the predicted boron letdown curve that develop during fuel depletion may be an indication that the calculational model is not adequate for core burnups beyond BOL, or that an unexpected change in core conditions has occurred.

BASES

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APPLICABLE SAFETY ANALYSES (continued)

The normalization of predicted RCS boron concentration to the measured value is typically performed after reaching RTP following startup from a refueling outage, with the control rods in their normal positions for power operation. The normalization is performed at BOL conditions, so that core reactivity relative to predicted values can be continually monitored and evaluated as core conditions change during the cycle.

Core reactivity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

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LCO

Long term core reactivity behavior is a result of the core physics design and cannot be easily controlled once the core design is fixed. During operation, therefore, the LCO can only be ensured through measurement and tracking, and appropriate actions taken as necessary. Large differences between actual and predicted core reactivity may indicate that the assumptions of the DBA and transient analyses are no longer valid, or that the uncertainties in the Nuclear Design Methodology are larger than expected. A limit on the reactivity balance of  $\pm 1\% \Delta k/k$  has been established based on engineering judgment. A 1% deviation in reactivity from that predicted is larger than expected for normal operation and should therefore be evaluated.

When measured core reactivity is within  $1\% \Delta k/k$  of the predicted value at steady state thermal conditions, the core is considered to be operating within acceptable design limits. Since deviations from the limit are normally detected by comparing predicted and measured steady state RCS critical boron concentrations, the difference between measured and predicted values would be approximately 100 ppm (depending on the boron worth) before the limit is reached. These values are well within the uncertainty limits for analysis of boron concentration samples, so that spurious violations of the limit due to uncertainty in measuring the RCS boron concentration are unlikely.

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APPLICABILITY

The limits on core reactivity must be maintained during MODES 1 and 2 because a reactivity balance must exist when the reactor is critical or producing THERMAL POWER. As the fuel depletes, core conditions are changing, and confirmation of the reactivity balance ensures the core is operating as designed. This Specification does not apply in MODES 3, 4, and 5 because the reactor is shut down and the reactivity balance is not changing.

In MODE 6, fuel loading results in a continually changing core reactivity. Boron concentration requirements (LCO 3.9.1, "Boron Concentration") ensure that fuel movements are performed within the bounds of the safety analysis. A SDM demonstration is required during the first startup following operations that could have altered core reactivity (e.g., fuel movement, control rod replacement, control rod shuffling).

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BASES

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ACTIONS

A.1 and A.2

Should an anomaly develop between measured and predicted core reactivity, an evaluation of the core design and safety analysis must be performed. Core conditions are evaluated to determine their consistency with input to design calculations. Measured core and process parameters are evaluated to determine that they are within the bounds of the safety analysis, and safety analysis calculational models are reviewed to verify that they are adequate for representation of the core conditions. The required Completion Time of 7 days is based on the low probability of a DBA occurring during this period, and allows sufficient time to assess the physical condition of the reactor and complete the evaluation of the core design and safety analysis.

Following evaluations of the core design and safety analysis, the cause of the reactivity anomaly may be resolved. If the cause of the reactivity anomaly is a mismatch in core conditions at the time of RCS boron concentration sampling, then a recalculation of the RCS boron concentration requirements may be performed to demonstrate that core reactivity is behaving as expected. If an unexpected physical change in the condition of the core has occurred, it must be evaluated and corrected, if possible. If the cause of the reactivity anomaly is in the calculation technique, then the calculational models must be revised to provide more accurate predictions. If any of these results are demonstrated, and it is concluded that the reactor core is acceptable for continued operation, then the boron letdown curve may be renormalized and power operation may continue. If operational restriction or additional SRs are necessary to ensure the reactor core is acceptable for continued operation, then they must be defined.

The required Completion Time of 7 days is adequate for preparing whatever operating restrictions or Surveillances that may be required to allow continued reactor operation.

B.1

If the core reactivity cannot be restored to within the 1%  $\Delta k/k$  limit, 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. If the SDM for MODE 3 is not met, then the boration required by SR 3.1.1.1 would occur. The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.1.2.1

Core reactivity is verified by periodic comparisons of measured and predicted RCS boron concentrations. The comparison is made, considering that other core conditions are fixed or stable, including control rod position, moderator temperature, fuel temperature, fuel depletion, xenon concentration, and samarium concentration. The Surveillance is performed prior to entering MODE 1 as an initial check on core conditions and design calculations at BOL. The SR is modified by a Note. The Note indicates that the normalization of predicted core reactivity to the measured value may take place, if required, within the first 60 effective full power days (EFPD) after each fuel loading. This allows sufficient time for core conditions to reach steady state, but prevents operation for a large fraction of the fuel cycle without establishing a benchmark for the design calculations. The required subsequent Frequency, following the initial 60 EFPD after entering MODE 1, is acceptable, based on the slow rate of core changes due to fuel depletion and the presence of other indicators (QPTR, AFD, etc.) for prompt indication of an anomaly.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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REFERENCES

1. 10 CFR 50, Appendix A, GDC 26, GDC 28, and GDC 29.
  2. UFSAR, Chapter 15.
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## B 3.1 REACTIVITY CONTROL SYSTEMS

### B 3.1.3 Moderator Temperature Coefficient (MTC)

#### BASES

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**BACKGROUND** According to GDC 11 (Ref. 1), the reactor core and its interaction with the Reactor Coolant System (RCS) must be designed for inherently stable power operation, even in the possible event of an accident. In particular, the net reactivity feedback in the system must compensate for any unintended reactivity increases.

The MTC relates a change in core reactivity to a change in reactor coolant temperature (a positive MTC means that reactivity increases with increasing moderator temperature; conversely, a negative MTC means that reactivity decreases with increasing moderator temperature). The reactor is designed to operate with a negative MTC over the largest possible range of fuel cycle operation. Therefore, a coolant temperature increase will cause a reactivity decrease, so that the coolant temperature tends to return toward its initial value. Reactivity increases that cause a coolant temperature increase will thus be self limiting, and stable power operation will result.

MTC values are predicted at selected burnups during the safety evaluation analysis and are confirmed to be acceptable by measurements. Both initial and reload cores are designed so that the beginning of cycle life (BOL) MTC is less than zero when THERMAL POWER is at RTP. The actual value of the MTC is dependent on core characteristics, such as fuel loading and reactor coolant soluble boron concentration. The core design may require additional fixed distributed poisons to yield an MTC at BOL within the range analyzed in the plant accident analysis. The end of cycle life (EOL) MTC is also limited by the requirements of the accident analysis. Fuel cycles that are designed to achieve high burnups or that have changes to other characteristics are evaluated to ensure that the MTC does not exceed the EOL limit.

The limitations on MTC are provided to ensure that the value of this coefficient remains within the limiting conditions assumed in the UFSAR accident and transient analyses.

BASES

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## BACKGROUND (continued)

If the LCO limits are not met, the unit response during transients may not be as predicted. The core could violate criteria that prohibit a return to criticality, or the departure from nucleate boiling ratio criteria of the approved correlation may be violated, which could lead to a loss of the fuel cladding integrity.

The SRs for measurement of the MTC at the beginning and near the end of the fuel cycle are adequate to confirm that the MTC remains within its limits, since this coefficient changes slowly, due principally to the reduction in RCS boron concentration associated with fuel burnup.

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APPLICABLE  
SAFETY  
ANALYSES

The acceptance criteria for the specified MTC are:

- a. The MTC values must remain within the bounds of those used in the accident analysis (Ref. 2) and
- b. The MTC must be such that inherently stable power operations result during normal operation and accidents, such as overheating and overcooling events.

The UFSAR, Chapter 15 (Ref. 2), contains analyses of accidents that result in both overheating and overcooling of the reactor core. MTC is one of the controlling parameters for core reactivity in these accidents. Both the most positive value and most negative value of the MTC are important to safety, and both values must be bounded. Values used in the analyses consider worst case conditions to ensure that the accident results are bounding (Ref. 3).

The consequences of accidents that cause core overheating must be evaluated when the MTC is positive. Such accidents include the rod withdrawal transient from either zero (Ref. 4) or RTP, loss of main feedwater flow, and loss of forced reactor coolant flow. The consequences of accidents that cause core overcooling must be evaluated when the MTC is negative. Such accidents include sudden feedwater flow increase and sudden decrease in feedwater temperature.

BASES

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## APPLICABLE SAFETY ANALYSES (continued)

In order to ensure a bounding accident analysis, the MTC is assumed to be its most limiting value for the analysis conditions appropriate to each accident. The bounding value is determined by considering rodged and unrodged conditions, whether the reactor is at full or zero power, and whether it is the BOL or EOL. The most conservative combination appropriate to the accident is then used for the analysis (Ref. 2).

MTC values are bounded in reload safety evaluations assuming steady state conditions at BOL and EOL. An EOL measurement is conducted at conditions when the RCS boron concentration reaches approximately 300 ppm. The measured value may be extrapolated to project the EOL value, in order to confirm reload design predictions.

MTC satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii). Even though it is not directly observed and controlled from the control room, MTC is considered an initial condition process variable because of its dependence on boron concentration.

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## LCO

LCO 3.1.3 requires the MTC to be maintained within specified limits of the COLR to ensure that the core operates within the assumptions of the accident analysis. During the reload core safety evaluation, the MTC is analyzed to determine that its values remain within the bounds of the original accident analysis during operation.

Assumptions made in safety analyses require that the MTC be less positive than a given upper bound and more positive than a given lower bound. The MTC is most positive at BOL; this upper bound must not be exceeded. This maximum upper limit occurs at BOL, all rods out (ARO), hot zero power conditions. At EOL the MTC takes on its most negative value, when the lower bound becomes important. This LCO exists to ensure that both the upper and lower bounds are not exceeded.

During operation, therefore, the conditions of the LCO can only be ensured through measurement. The Surveillance checks at BOL and EOL on MTC provide confirmation that the MTC is behaving as anticipated so that the acceptance criteria are met.

The LCO establishes a maximum positive value that cannot be exceeded. The BOL positive limit and the EOL negative limit are established in the COLR to allow specifying limits for each particular cycle. This permits the unit to take advantage of improved fuel management and changes in unit operating schedule.

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**BASES**

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**APPLICABILITY** Technical Specifications place both LCO and SR values on MTC, based on the safety analysis assumptions described above.

In MODE 1, the limits on MTC must be maintained to ensure that any accident initiated from THERMAL POWER operation will not violate the design assumptions of the accident analysis. In MODE 2 with the reactor critical, the BOL limit must be maintained to ensure that startup and subcritical accidents (such as the uncontrolled control rod assembly or group withdrawal) will not violate the assumptions of the accident analysis. The EOL MTC limit must be maintained in MODES 2 and 3, in addition to MODE 1, to ensure that cooldown accidents will not violate the assumptions of the accident analysis. In MODES 4, 5, and 6, this LCO is not applicable, since no Design Basis Accidents using the MTC as an analysis assumption are initiated from these MODES.

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**ACTIONS**A.1

If the BOL MTC limit is violated, administrative withdrawal limits for control banks must be established to maintain the MTC within its limits. The MTC becomes more negative with control bank insertion and decreased boron concentration. A Completion Time of 24 hours provides enough time for evaluating the MTC measurement and computing the required bank withdrawal limits.

As cycle burnup is increased, the RCS boron concentration will be reduced. The reduced boron concentration causes the MTC to become more negative. Using physics calculations, the time in cycle life at which the calculated MTC will meet the LCO requirement can be determined. At this point in core life Condition A no longer exists. The unit is no longer in the Required Action, so the administrative withdrawal limits are no longer in effect.

B.1

If the required administrative withdrawal limits at BOL are not established within 24 hours, the unit must be brought to at least MODE 2 with  $k_{\text{eff}} < 1.0$  to prevent operation with an MTC that is more positive than that assumed in safety analyses.

The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

BASES

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ACTIONS (continued)

C.1

Exceeding the EOL MTC limit means that the safety analysis assumptions for the EOL accidents that use a bounding negative MTC value may be invalid. If the EOL MTC limit is exceeded, the plant must be brought to a MODE or condition in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least MODE 4 within 12 hours.

The allowed Completion Time is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

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SURVEILLANCE  
REQUIREMENTS

SR 3.1.3.1

This SR requires measurement of the MTC at BOL prior to entering MODE 1 in order to demonstrate compliance with the most positive MTC LCO. Meeting the limit prior to entering MODE 1 ensures that the limit will also be met at higher power levels.

The BOL MTC value for ARO will be inferred from isothermal temperature coefficient measurements obtained during the physics tests after refueling. The ARO value can be directly compared to the BOL MTC limit of the LCO. If required, measurement results and predicted design values can be used to establish administrative withdrawal limits for control banks.

## BASES

## SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.3.2

In similar fashion, the LCO demands that the MTC be less negative than the specified value for EOL full power conditions. This measurement may be performed at any THERMAL POWER, but its results must be extrapolated to the conditions of RTP and all banks withdrawn in order to make a proper comparison with the LCO value. Because the RTP MTC value will gradually become more negative with further core depletion and boron concentration reduction, a 300 ppm SR value of MTC should necessarily be less negative than the EOL LCO limit. The 300 ppm SR value is sufficiently less negative than the EOL LCO limit value to ensure that the LCO limit will be met when the 300 ppm Surveillance criterion is met.

SR 3.1.3.2 is modified by three Notes that include the following requirements:

- a. The SR is not required to be performed until 7 effective full power days (EFPDs) after reaching the equivalent of an equilibrium RTP all rods out (ARO) boron concentration of 300 ppm.
- b. If the 300 ppm Surveillance limit is not met, it is possible that the EOL limit on MTC could be reached before the planned EOL. Because the MTC changes slowly with core depletion, the Frequency of 14 effective full power days is sufficient to avoid exceeding the EOL limit on MTC.
- c. If the measured MTC at 60 ppm is more positive than the 60 ppm Surveillance limit, the EOL limit on MTC will not be exceeded because of the gradual manner in which MTC changes with core burnup. The 60 ppm Surveillance is only performed if the 300 ppm Surveillance limit was not met (see note b). If the 60 ppm Surveillance limit is met, no further Surveillance of EOL MTC is required for the remainder of the fuel cycle. If the 60 ppm Surveillance limit is not met, then Surveillance of EOL MTC is required for the remainder of the fuel cycle as described in note b.

## REFERENCES

1. 10 CFR 50, Appendix A, GDC 11.
2. UFSAR, Chapter 15.
3. BAW 10169P-A, "B&W Safety Analysis Methodology for Recirculating Steam Generator Plants," October 1989.
4. UFSAR, Section 15.2.1.

## B 3.1 REACTIVITY CONTROL SYSTEMS

### B 3.1.4 Rod Group Alignment Limits

#### BASES

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**BACKGROUND** The OPERABILITY (i.e., trippability) of the shutdown and control rods is an initial assumption in all safety analyses that assume rod insertion upon reactor trip. Maximum rod misalignment is an initial assumption in the safety analysis that directly affects core power distributions and assumptions of available SDM.

The applicable criteria for these reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10, "Reactor Design," GDC 26, "Reactivity Control System Redundancy and Capability" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plants" (Ref. 2).

Mechanical or electrical failures may cause a control or shutdown rod to become inoperable or to become misaligned from its group. Rod inoperability or misalignment may cause increased power peaking, due to the asymmetric reactivity distribution and a reduction in the total available rod worth for reactor shutdown. Therefore, rod alignment and OPERABILITY are related to core operation in design power peaking limits and the core design requirement of a minimum SDM.

Limits on rod alignment have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

Rod cluster control assemblies (RCCAs), or rods, are moved by their control rod drive mechanisms (CRDMs). Each CRDM moves its RCCA one step (approximately 5/8 inch) at a time, but at varying rates (steps per minute) depending on the signal output from the Rod Control System.

The RCCAs are divided among control banks and shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control. A group consists of two or more RCCAs that are electrically paralleled to step simultaneously. If a bank of RCCAs consists of two groups, the groups are moved in a staggered fashion, but always within one step of each other. Each unit has four control banks and four shutdown banks.

The shutdown banks are maintained either in the fully inserted or fully withdrawn position. The control banks are moved in an overlap pattern, using the following withdrawal sequence: When Control Bank A reaches a predetermined height in the core, Control Bank B begins to move out with Control Bank A. Control Bank A stops at the position of maximum

## BASES

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### BACKGROUND (continued)

withdrawal, and Control Bank B continues to move out. When Control Bank B reaches a predetermined height, Control Bank C begins to move out with Control Bank B. This sequence continues until Control Banks A, B, and C are at the fully withdrawn position, and Control Bank D is approximately halfway withdrawn. The insertion sequence is the opposite of the withdrawal sequence. The control rods are arranged in a radially symmetric pattern, so that control bank motion does not introduce radial asymmetries in the core power distributions.

The axial position of shutdown rods and control rods is indicated by two separate and independent systems that are the Demand Position Indication System (commonly called group step counters) and the Rod Position Indication System.

The Demand Position Indication System counts the pulses from the rod control system that moves the rods. There is one step counter for each group of rods. Individual rods in a group receive the same signal to move and should, therefore, all be at the same position indicated by the group step counter for that group. The Demand Position Indication System is considered highly precise ( $\pm 1$  step or  $\pm 5/8$  inch). If a rod does not move one step for each demand pulse, the step counter will still count the pulse and incorrectly reflect the position of the rod.

The Rod Position Indication System provides an indication of actual rod position, but at a lower precision than the step counters. This system is based on inductive analog signals from a series of coils spaced along a hollow tube. The Rod Position Indication System is capable of monitoring rod position within at least  $\pm 12$  steps.

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### APPLICABLE SAFETY ANALYSES

Control rod misalignment accidents are analyzed in the safety analysis (Ref. 3). The acceptance criteria for addressing control rod inoperability or misalignment are that:

- a. There be no violations of:
  1. Specified acceptable fuel design limits; or
  2. Reactor Coolant System (RCS) pressure boundary integrity; and
- b. The core remains subcritical after accident transients.

## BASES

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### APPLICABLE SAFETY ANALYSES (continued)

There are three RCCA misalignment accidents which are analyzed. They include one or more dropped RCCAs, a dropped RCCA bank, and a statically misaligned RCCA (Ref. 4). A different type of misalignment occurs if one rod fails to insert upon a reactor trip and remains stuck fully withdrawn. This condition requires an evaluation to determine that sufficient reactivity worth is held in the control rods to meet the SDM requirement, with the maximum worth rod stuck fully withdrawn.

For the dropped RCCA(s) misalignment accident, a negative reactivity insertion will result. For those dropped RCCA(s) that do not result in a reactor trip, power may be reestablished either by reactivity feedback or control bank withdrawal. Following a dropped rod event in manual rod control, the plant will establish a new equilibrium condition. The equilibrium process without control system interaction is monotonic, thus removing power overshoot as a concern and establishing the automatic rod control mode of operation as the limiting case.

For the dropped RCCA bank misalignment accident, a reactivity insertion of greater than 500 pcm which will be detected by the power range negative neutron flux rate trip circuitry. The reactor is then tripped. The core is not adversely affected during this period since power is decreasing rapidly. Following the reactor trip, normal shutdown procedures are followed to further cool down the plant.

Two types of analysis are performed in regard to static rod misalignment (Ref. 3). With control banks at their insertion limits, one type of analysis considers the case when any one rod is completely inserted into the core. The second type of analysis considers the case of a completely withdrawn single rod and Control Bank D is fully inserted to its insertion limit. Satisfying limits on departure from nucleate boiling ratio in both of these cases bounds the situation when a rod is misaligned from its group by  $\pm 12$  steps.

Another type of misalignment occurs if one RCCA fails to insert upon a reactor trip and remains stuck fully withdrawn. This condition is assumed in the evaluation to determine that the required SDM is met with the maximum worth RCCA also fully withdrawn (Ref. 4).

Shutdown and control rod OPERABILITY and alignment are directly related to power distributions and SDM, which are initial conditions assumed in safety analyses. Therefore they satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

BASES

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LCO

The limits on shutdown or control rod alignments ensure that the assumptions in the safety analysis will remain valid. The requirements on control rod OPERABILITY ensure that upon reactor trip, the assumed reactivity will be available and will be inserted. The control rod OPERABILITY requirements (i.e., trippability) are separate from the alignment requirements, which ensure that the RCCAs and banks maintain the correct power distribution and rod alignment. The rod OPERABILITY requirement is satisfied provided the rod will fully insert in the required rod drop time assumed in the safety analysis. Rod control malfunctions that result in the inability to move a rod (e.g., rod lift coil failures), but that do not impact trippability, do not result in rod inoperability.

The requirement to maintain the rod alignment to within plus or minus 12 steps is conservative. The minimum misalignment assumed in safety analysis is 10% of span (14.4 inches), and in some cases a total misalignment from fully withdrawn to fully inserted is assumed.

Failure to meet the requirements of this LCO may produce unacceptable power peaking factors and linear heat rates (LHRs), or unacceptable SDMs, that may constitute initial conditions inconsistent with the safety analysis.

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APPLICABILITY

The requirements on RCCA OPERABILITY and alignment are applicable in MODES 1 and 2 because these are the only MODES in which neutron (or fission) power is generated, and the OPERABILITY (i.e., trippability) and alignment of rods have the potential to affect the safety of the plant. In MODES 3, 4, 5, and 6, the alignment limits do not apply because, except for control rod OPERABILITY testing, the control rods are bottomed and the reactor is shut down and not producing fission power. In the shutdown MODES, the OPERABILITY of the shutdown and control rods has the potential to affect the required SDM, but this effect can be compensated for by an increase in the boron concentration of the RCS. See LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," for SDM in MODES 3, 4, and 5 and LCO 3.9.1, "Boron Concentration," for boron concentration requirements during refueling.

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ACTIONS

A.1.1 and A.1.2

When one or more rods are inoperable (i.e., untrippable), there is a possibility that the required SDM may be adversely affected. Under these conditions, it is important to determine the SDM, and if it is less than the required value, initiate boration until the required SDM is recovered. The Completion Time of 1 hour is adequate for determining SDM and, if necessary, for initiating emergency boration and restoring SDM.

In this situation, SDM verification must include the worth of the untrippable rod, as well as a rod of maximum worth.

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## BASES

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### ACTIONS (continued)

#### A.2

If the inoperable rod(s) cannot be restored to OPERABLE status, the plant must be brought to a MODE or condition in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours.

The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

#### B.1

When a rod becomes misaligned, it can usually be moved and is trippable. If the rod can be realigned within the Completion Time of 1 hour, local xenon redistribution during this short interval will not be significant, and operation may proceed without further restriction.

An alternative to realigning a single misaligned RCCA to the group average position is to align the remainder of the group to the position of the misaligned RCCA. However, this must be done without violating the bank sequence, overlap, and insertion limits specified in LCO 3.1.5, "Shutdown Bank Insertion Limits," and LCO 3.1.6, "Control Bank Insertion Limits." The Completion Time of 1 hour gives the operator sufficient time to adjust the rod positions in an orderly manner.

#### B.2.1.1 and B.2.1.2

With a misaligned rod, SDM must be verified to be within limit or boration must be initiated to restore SDM to within limit.

In many cases, realigning the remainder of the group to the misaligned rod may not be desirable.

Power operation may continue with one RCCA misaligned but trippable (OPERABLE), provided that SDM is verified within 1 hour. The Completion Time of 1 hour represents the time necessary for determining the actual unit SDM and, if necessary, aligning and starting the necessary systems and components to initiate boration.

## BASES

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### ACTIONS (continued)

#### B.2.2, B.2.3, B.2.4, B.2.5, and B.2.6

For continued operation with a misaligned rod, RTP must be reduced, SDM must periodically be verified within limits, hot channel factors ( $F_Q(X,Y,Z)$  and  $F_{\Delta H}(X,Y)$ ) must be verified within limits, and the safety analyses must be re-evaluated to confirm continued operation is permissible.

Reduction of power to 75% RTP ensures that local LHR increases resulting from a misaligned RCCA will not cause the core design criteria to be exceeded (Ref. 5). The Completion Time of 2 hours gives the operator sufficient time to accomplish an orderly power reduction without challenging the Reactor Protection System.

When a rod is known to be misaligned, there is a potential to impact the SDM. Since the core conditions can change with time, periodic verification of SDM is required. A Frequency of 12 hours is sufficient to ensure this requirement continues to be met.

Verifying that  $F_Q(X,Y,Z)$  and  $F_{\Delta H}(X,Y)$  are within the required limits ensures that current operation at 75% RTP with a rod misaligned is not resulting in power distributions that may invalidate safety analysis assumptions at full power. The Completion Time of 72 hours allows sufficient time to obtain flux maps of the core power distribution using the incore flux mapping system and to calculate  $F_Q(X,Y,Z)$  and  $F_{\Delta H}(X,Y)$ .

Once current conditions have been verified acceptable, time is available to perform evaluations of accident analysis to determine that core limits will not be exceeded during a Design Basis Event for the duration of operation under these conditions. The accident analyses presented in UFSAR Chapter 15 (Ref. 5) that may be adversely affected will be evaluated to ensure that the analysis results remain valid for the duration of continued operation under these conditions. A Completion Time of 5 days is sufficient time to obtain the required input data and to perform the analysis.

#### C.1

When Required Actions cannot be completed within their Completion Time, the unit must be brought to a MODE in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours, which

## BASES

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### ACTIONS (continued)

obviates concerns about the development of undesirable xenon or power distributions. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging the plant systems.

#### D.1.1 and D.1.2

More than one control rod becoming misaligned from its group average position is not expected, and has the potential to reduce SDM. Therefore, SDM must be evaluated. One hour allows the operator adequate time to determine SDM. Restoration of the required SDM, if necessary, requires increasing the RCS boron concentration to provide negative reactivity, as described in the Bases of LCO 3.1.1. The required Completion Time of 1 hour for initiating boration is reasonable, based on the time required for potential xenon redistribution, the low probability of an accident occurring, and the steps required to complete the action. This allows the operator sufficient time to align the required valves and start the boric acid pumps. Boration will continue until the required SDM is restored.

#### D.2

If more than one rod is found to be misaligned or becomes misaligned because of bank movement, the unit conditions fall outside of the accident analysis assumptions. Since automatic sequencing of the control banks would continue to cause misalignment, the unit must be brought to a MODE or Condition in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours.

The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

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### SURVEILLANCE REQUIREMENTS

#### SR 3.1.4.1

Verification that individual rod positions are within alignment limits allows the operator to detect a rod that is beginning to deviate from its expected position.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.4.2

Verifying each control rod is OPERABLE would require that each rod be tripped. However, in MODES 1 and 2, tripping each control rod would result in radial or axial power tilts, or oscillations. Exercising each individual control rod provides increased confidence that all rods continue to be OPERABLE without exceeding the alignment limit, even if they are not regularly tripped. Moving each control rod by greater than or equal to 10 steps in either direction will not cause radial or axial power tilts, or oscillations, to occur.

To ensure minimum  $F_{\Delta H}$  peaking factor margins are maintained in accordance with TS 3.2.2 during SR 3.1.4.2 rod testing, margin penalties are typically assigned during the test, as described in the Nuclear Design Report. The minimum predicted  $F_{\Delta H}$  future margin, including penalties, should be verified prior to performing the test to ensure adequate margin will be maintained. In the event that a potential negative margin condition exists, compliance with the associated Conditions and Required Actions of TS 3.2.2 should be verified prior to performing rod testing.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

Between required performances of SR 3.1.4.2 (determination of control rod OPERABILITY by movement), if a control rod(s) is discovered to be immovable, but remains trippable, the control rod(s) is considered to be OPERABLE. At any time, if a control rod(s) is immovable, a determination of the trippability (OPERABILITY) of the control rod(s) must be made, and appropriate action taken.

SR 3.1.4.3

Verification of rod drop times allows the operator to determine that the maximum rod drop time permitted is consistent with the assumed rod drop time used in the safety analysis. Measuring rod drop times prior to reactor criticality, after reactor vessel head installation, ensures that the reactor internals and rod drive mechanism will not interfere with rod motion or rod drop time, and that no degradation in these systems has occurred that would adversely affect control rod motion or drop time. This testing is performed with all RCPs operating and the average moderator temperature  $\geq 500^{\circ}\text{F}$  to simulate a reactor trip under actual conditions. Fully withdrawn shall be the condition where shutdown and control banks are at a position within the interval of  $\geq 222$  and  $\leq 231$  steps withdrawn, inclusive.

BASES

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SURVEILLANCE REQUIREMENTS (continued)

This Surveillance is performed during a plant outage, due to the plant conditions needed to perform the SR and the potential for an unplanned plant transient if the Surveillance were performed with the reactor at power.

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REFERENCES

1. 10 CFR 50, Appendix A, GDC 10 and GDC 26.
  2. 10 CFR 50.46.
  3. UFSAR, Section 15.2.3.
  4. UFSAR, Section 15.4.2.
  5. UFSAR, Chapter 15.
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## B 3.1 REACTIVITY CONTROL SYSTEMS

### B 3.1.5 Shutdown Bank Insertion Limits

#### BASES

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**BACKGROUND** The insertion limits of the shutdown and control rods are initial assumptions in all safety analyses that assume rod insertion upon reactor trip. The insertion limits directly affect core power and fuel burnup distributions and assumptions of available ejected rod worth, SDM and initial reactivity insertion rate.

The applicable criteria for these reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10, "Reactor Design," GDC 26, "Reactivity Control System Redundancy and Protection," GDC 28, "Reactivity Limits" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 2). Limits on control rod insertion have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

The rod cluster control assemblies (RCCAs) are divided among control banks and shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control. A group consists of two or more RCCAs that are electrically paralleled to step simultaneously. A bank of RCCAs consists of two groups that are moved in a staggered fashion, but always within one step of each other. Each unit has four control banks and four shutdown banks. See LCO 3.1.4, "Rod Group Alignment Limits," for control and shutdown rod OPERABILITY and alignment requirements, and LCO 3.1.7, "Rod Position Indication," for position indication requirements.

The control banks are used for precise reactivity control of the reactor. The positions of the control banks are normally automatically controlled by the Rod Control System, but they can also be manually controlled. They are capable of adding negative reactivity very quickly (compared to borating). The control banks must be maintained above designed insertion limits and are typically near the fully withdrawn position during normal full power operations.

Hence, they are not capable of adding a large amount of positive reactivity. Boration or dilution of the Reactor Coolant System (RCS) compensates for the reactivity changes associated with large changes in RCS temperature. The design calculations are performed with the assumption that the shutdown banks are withdrawn first. The shutdown banks can be fully withdrawn without the core going critical. This provides available negative reactivity in the event of boration errors. The

BASES

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BACKGROUND (continued)

shutdown banks are controlled manually by the control room operator. During normal unit operation, the shutdown banks are either fully withdrawn or fully inserted. They are moved quarterly or following maintenance to ensure trippability but are returned to the withdrawn position when the testing is completed. The shutdown banks must be withdrawn from the core, prior to withdrawing any control banks during an approach to criticality. They affect core power and burnup distribution, and add negative reactivity to shut down the reactor upon receipt of a reactor trip signal.

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APPLICABLE  
SAFETY  
ANALYSES

On a reactor trip, all RCCAs (shutdown banks and control banks), except the most reactive RCCA, are assumed to insert into the core. The shutdown banks shall be at or above their insertion limits and available to insert the maximum amount of negative reactivity on a reactor trip signal. The control banks may be partially inserted in the core, as allowed by LCO 3.1.6, "Control Bank Insertion Limits." The shutdown bank and control bank insertion limits are established to ensure that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM (see LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") following a reactor trip from full power. The combination of control banks and shutdown banks (less the most reactive RCCA, which is assumed to be fully withdrawn) is sufficient to take the reactor from full power conditions at rated temperature to zero power, and to maintain the required SDM at rated no load temperature (Ref. 3). The shutdown bank insertion limit also limits the reactivity worth of an ejected shutdown rod.

The acceptance criteria for addressing shutdown and control rod bank insertion limits and inoperability or misalignment is that:

- a. There be no violations of:
  - 1. Specified acceptable fuel design limits or
  - 2. RCS pressure boundary integrity and
- b. The core remains subcritical after accident transients.

As such, the shutdown bank insertion limits affect safety analyses involving core reactivity and SDM (Ref. 3).

The shutdown bank insertion limits preserve an initial condition assumed in the safety analyses and, as such, satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

BASES

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LCO The shutdown banks must be within their insertion limits any time the reactor is critical or approaching criticality. This ensures that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM following a reactor trip.

The shutdown bank insertion limits are defined in the COLR.

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APPLICABILITY The shutdown banks must be within their insertion limits, with the reactor in MODES 1 and 2. This ensures that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM following a reactor trip. The shutdown banks do not have to be within their insertion limits in MODE 3, unless an approach to criticality is being made. In MODE 3, 4, 5, or 6, the shutdown banks, except for control rod OPERABILITY testing, are fully inserted in the core and contribute to the SDM. Refer to LCO 3.1.1 for SDM requirements in MODE 2  $k_{\text{eff}} < 1.0$ , MODES 3, 4, and 5. LCO 3.9.1, "Boron Concentration," ensures adequate SDM in MODE 6.

The Applicability requirements have been modified by a Note indicating the LCO requirement is suspended during SR 3.1.4.2. This SR verifies the freedom of the rods to move, and requires the shutdown bank to move below the LCO limits, which would normally violate the LCO.

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ACTIONS A.1, A.2, A.3, A.4, A.5, A.6, and A.7

When one shutdown bank is inserted beyond the insertion limit due to performance of trippability testing per SR 3.1.4.2 and is immovable due to a malfunction in the rod control system, 72 hours are provided to restore the shutdown banks to within limits. Additionally, immediate verification is required to prove that the shutdown bank is less than or equal to 18 steps below the insertion limit as measured by the group demand position indicators, the individual control rod alignment limits of LCOs 3.1.4 and 3.1.6 are met, there are no reactor coolant system boron dilution activities, and there are no power level increases taking place. Checks are performed for each reload core to ensure that bank insertions of up to 18 steps will not result in power distributions which violate the DNB criterion for ANS Condition II transients, (moderate frequency transients analyzed in Section 15.2 of the UFSAR). Administrative requirements on the initial controlling bank position will ensure that this insertion and an additional controlling bank insertion of five steps or less will not violate the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 during the repair period. If the controlling bank is inserted more than five steps deeper than its initial position, a calculation will be performed to ensure that the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is met. Since no dilution or power level increases are allowed, shutdown margin will be maintained as long as the controlling bank is far enough above its insertion limit to compensate for the inserted worth of the bank that is beyond its insertion limit. Furthermore, a verification of SDM is

## BASES

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### ACTIONS (continued)

required within 12 hours and when the controlling bank is inserted more than 5 steps from the initial position. The requirement to be in compliance with LCOs 3.1.4 and 3.1.6 ensures that the rods are trippable, and power distribution is acceptable during the time allowed to restore the inserted rod. The 12 hour requirement to verify the SDM is within limits ensures the SDM requirements of LCO 3.1.1 are met during the repair period. Furthermore, the requirement to verify the SDM is within limits when a controlling bank is inserted five steps or more also ensures that SDM requirements of LCO 3.1.1 are met during the repair period. If any of these Required Actions are not met, Condition C must be applied.

The Completion Time of 72 hours is based on operating experience and provides an acceptable time for evaluating and repairing problems with the rod control system, while restricting the probability of a more severe (i.e., ANS Condition III or IV) accident or transient condition occurring concurrently with the insertion limit violation.

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

When one or more shutdown banks is not within insertion limits for reasons other than Condition A, 2 hours is allowed to restore the shutdown banks to within the insertion limits. This is necessary because the available SDM may be significantly reduced, with one or more of the shutdown banks not within their insertion limits. Also, verification of SDM or initiation of boration within 1 hour is required, since the SDM in MODES 1 and 2 is ensured by adhering to the control and shutdown bank insertion limits (see LCO 3.1.1). If shutdown banks are not within their insertion limits, then SDM will be verified by performing a reactivity balance calculation, considering the effects listed in the Bases for SR 3.1.1.1.

The allowed Completion Time of 2 hours provides an acceptable time for evaluating and repairing minor problems without allowing the plant to remain in an unacceptable condition for an extended period of time.

#### C.1

If the Required Action(s) of Condition A or B are not met within the associated Completion Times, the unit must be brought to a MODE where the LCO is not applicable. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.1.5.1

Verification that the shutdown banks are within their insertion limits prior to an approach to criticality ensures that when the reactor is critical, or being taken critical, the shutdown banks will be available to shut down the reactor, and the required SDM will be maintained following a reactor trip. This SR and Frequency ensure that the shutdown banks are withdrawn before the control banks are withdrawn during a unit startup.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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REFERENCES

1. 10 CFR 50, Appendix A, GDC 10, GDC 26, and GDC 28.
  2. 10 CFR 50.46.
  3. UFSAR, Chapter 15.
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## B 3.1 REACTIVITY CONTROL SYSTEMS

### B 3.1.6 Control Bank Insertion Limits

#### BASES

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**BACKGROUND** The insertion limits of the shutdown and control rods are initial assumptions in the safety analyses that assume rod insertion upon reactor trip. The insertion limits directly affect core power and fuel burnup distributions and assumptions of available SDM, and initial reactivity insertion rate.

The applicable criteria for these reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10, "Reactor Design," GDC 26, "Reactivity Control System Redundancy and Protection," GDC 28, "Reactivity Limits" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 2). Limits on control rod insertion have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

The rod cluster control assemblies (RCCAs) are divided among control banks and shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control. A group consists of two or more RCCAs that are electrically paralleled to step simultaneously. A bank of RCCAs consists of two groups that are moved in a staggered fashion, but always within one step of each other. Each unit has four control banks and four shutdown banks. See LCO 3.1.4, "Rod Group Alignment Limits," for control and shutdown rod OPERABILITY and alignment requirements, and LCO 3.1.7, "Rod Position Indication," for position indication requirements.

The control bank insertion limits are specified in the COLR. The control banks are required to be at or above the insertion limit lines.

Overlap is the distance travelled together by two control banks. The predetermined position of control bank C, at which control bank D will begin to move with bank C on a withdrawal is shown on the COLR Figure. The fully withdrawn position is defined in the COLR.

BASES

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BACKGROUND (continued)

The control banks are used for precise reactivity control of the reactor. The positions of the control banks are normally controlled automatically by the Rod Control System, but can also be manually controlled. They are capable of adding reactivity very quickly (compared to borating or diluting).

The power density at any point in the core must be limited, so that the fuel design criteria are maintained. Together, LCO 3.1.4, LCO 3.1.5, "Shutdown Bank Insertion Limits," LCO 3.1.6, LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," provide limits on control component operation and on monitored process variables, which ensure that the core operates within the fuel design criteria.

The shutdown and control bank insertion and alignment limits, AFD, and QPTR are process variables that together characterize and control the three dimensional power distribution of the reactor core. Additionally, the control bank insertion limits control the reactivity that could be added in the event of a rod ejection accident, and the shutdown and control bank insertion limits ensure the required SDM is maintained.

Operation within the subject LCO limits will prevent fuel cladding failures that would breach the primary fission product barrier and release fission products to the reactor coolant in the event of a loss of coolant accident (LOCA), loss of flow, ejected rod, or other accident requiring termination by a Reactor Trip System (RTS) trip function.

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APPLICABLE  
SAFETY  
ANALYSES

The shutdown and control bank insertion limits, AFD, and QPTR LCOs are required to prevent power distributions that could result in fuel cladding failures in the event of a LOCA, loss of flow, ejected rod, or other accident requiring termination by an RTS trip function.

The acceptance criteria for addressing shutdown and control bank insertion limits and inoperability or misalignment are that:

- a. There be no violations of:
  - 1. Specified acceptable fuel design limits or
  - 2. Reactor Coolant System pressure boundary integrity and
- b. The core remains subcritical after accident transients.

BASES

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APPLICABLE SAFETY ANALYSES (continued)

As such, the shutdown and control bank insertion limits affect safety analyses involving core reactivity and power distributions (Ref. 3).

The SDM requirement is ensured by limiting the control and shutdown bank insertion limits so that allowable inserted worth of the RCCAs is such that sufficient reactivity is available in the rods to shut down the reactor to hot zero power with a reactivity margin that assumes the maximum worth RCCA remains fully withdrawn upon trip (Ref. 3).

Operation at the insertion limits or AFD limits may approach the maximum allowable linear heat generation rate or peaking factor with the allowed QPTR present. Operation at the insertion limit may indicate the maximum ejected RCCA worth could be equal to the limiting value in the fuel cycle that has sufficiently high ejected RCCA worths.

The control and shutdown bank insertion limits ensure that safety analyses assumptions for SDM, ejected rod worth, and power distribution peaking factors are preserved (Ref. 3).

The insertion limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii), in that they are initial conditions assumed in the safety analyses.

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LCO

The limits on control banks sequence, overlap, and physical insertion, as defined in the COLR, must be maintained because they serve the function of preserving power distribution, ensuring that the SDM is maintained, ensuring that ejected rod worth is maintained, and ensuring adequate negative reactivity insertion is available on trip. The overlap between control banks provides more uniform rates of reactivity insertion and withdrawal and is imposed to maintain acceptable power peaking during control bank motion.

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APPLICABILITY

The control bank sequence, overlap, and physical insertion limits shall be maintained with the reactor in MODES 1 and 2 with  $k_{\text{eff}} \geq 1.0$ . These limits must be maintained, since they preserve the assumed power distribution, ejected rod worth, SDM, and reactivity rate insertion assumptions. Applicability in MODE 2 with  $k_{\text{eff}} < 1.0$ , MODES 3, 4, and 5 is not required, since neither the power distribution nor ejected rod worth assumptions would be exceeded in these MODES.

The applicability requirements have been modified by a Note indicating the LCO requirements are suspended during the performance of SR 3.1.4.2. This SR verifies the freedom of the rods to move, and requires the control bank to move below the LCO limits, which would violate the LCO.

BASES

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ACTIONS

A.1, A.2, A.3, A.4, A.5, A.6, and A.7

When one control bank is inserted beyond the insertion limit due to performance of trippability testing per SR 3.1.4.2 and is immovable due to malfunctions in the rod control system, 72 hours are provided to restore the control banks to within limits. Additionally, immediate verification is required to prove that the control bank is less than or equal to 18 steps below the insertion limit as measured by the group demand position indicators, the individual rod alignment limits of LCOs 3.1.4 and 3.1.5 are met, there are no reactor coolant system boron concentration dilution activities, and there are no power level increases taking place. Checks are performed for each reload core to ensure that bank insertions of up to 18 steps will not result in power distributions which violate the DNB criterion for ANS Condition II transients (moderate frequency transients analyzed in Section 15.2 of the UFSAR). Administrative requirements on the initial controlling bank position will ensure that this insertion and an additional controlling bank insertion of five steps or less will not violate the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 during the repair period. If the controlling bank is inserted more than five steps deeper than its initial position, a calculation will be performed to ensure that the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is met. Since no dilution or power level increases are allowed, shutdown margin will be maintained as long as the controlling bank is far enough above its insertion limit to compensate for the inserted worth of the bank that is beyond its insertion limit. Furthermore, a verification of SDM is required within 12 hours and when the controlling bank is inserted more than 5 steps from the initial position. The requirement to be in compliance with LCOs 3.1.4 and 3.1.5 ensures that the rods are trippable, and power distribution is acceptable during the time allowed to restore the inserted bank. The 12 hour requirement to verify the SDM is within limits ensures the SDM requirements of LCO 3.1.1 are met during the repair period. Furthermore, the requirement to verify the SDM is within limits when a controlling bank is inserted five steps or more also ensures that SDM requirements of LCO 3.1.1 are met during the repair period. If any of these Required Actions are not met, Condition D must be applied.

The Condition is modified by a Note that specifies it only applies to control banks inserted beyond the insertion limit that are not controlling banks. A controlling bank is defined as a control bank that is less than fully withdrawn as defined in the COLR, with the exception of fully withdrawn banks that have been inserted for the performance of SR 3.1.4.2 (rod freedom of movement Surveillance).

The Completion Time of 72 hours is based on operating experience and provides an acceptable time for evaluating and repairing problems with the rod control system, while restricting the probability of a more severe (i.e., ANS Condition III or IV) accident or transient condition occurring concurrently with the insertion limit violation.

## BASES

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### ACTIONS (continued)

#### B.1.1, B.1.2, B.2, C.1.1, C.1.2, and C.2

When the control banks are outside the acceptable insertion limits, they must be restored to within those limits. This restoration can occur in two ways:

- a. Reducing power to be consistent with rod position or
- b. Moving rods to be consistent with power.

Also, verification of SDM or initiation of boration to regain SDM is required within 1 hour, since the SDM in MODES 1 and 2 normally ensured by adhering to the control and shutdown bank insertion limits (see LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") has been upset. If control banks are not within their insertion limits, then SDM will be verified by performing a reactivity balance calculation, considering the effects listed in the Bases for SR 3.1.1.1.

Similarly, if the control banks are found to be out of sequence or in the wrong overlap configuration, they must be restored to meet the limits.

Operation beyond the LCO limits is allowed for a short time period in order to take conservative action because the simultaneous occurrence of either a LOCA, loss of flow accident, ejected rod accident, or other accident during this short time period, together with an inadequate power distribution or reactivity capability, has an acceptably low probability.

The allowed Completion Time of 2 hours for restoring the banks to within the insertion, sequence, and overlap limits provides an acceptable time for evaluating and repairing minor problems without allowing the plant to remain in an unacceptable condition for an extended period of time.

#### D.1

If Required Action(s) of Condition A, B, or C are not met within the associated Completion Times, the plant must be brought to at least MODE 2 with  $k_{\text{eff}} < 1.0$ , where the LCO is not applicable. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.1.6.1

This Surveillance is required to ensure that the reactor does not achieve criticality with the control banks below their insertion limits.

The estimated critical position (ECP) depends upon a number of factors, one of which is xenon concentration. If the ECP was calculated long before criticality, xenon concentration could change to make the ECP substantially in error. Conversely, determining the ECP immediately before criticality could be an unnecessary burden. There are a number of unit parameters requiring operator attention at that point. Performing the ECP calculation within 4 hours prior to criticality avoids a large error from changes in xenon concentration, but allows the operator some flexibility to schedule the ECP calculation with other startup activities.

SR 3.1.6.2

Verification of the control bank insertion limits is sufficient to detect control banks that may be approaching the insertion limits.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.1.6.3

When control banks are maintained within their insertion limits as checked by SR 3.1.6.2 above, it is unlikely that their sequence and overlap will not be in accordance with requirements provided in the COLR.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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REFERENCES

1. 10 CFR 50, Appendix A, GDC 10, GDC 26, GDC 28.
  2. 10 CFR 50.46.
  3. UFSAR, Chapter 15.
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## B 3.1 REACTIVITY CONTROL SYSTEMS

### B 3.1.7 Rod Position Indication

#### BASES

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**BACKGROUND** According to GDC 13 (Ref. 1), instrumentation to monitor variables and systems over their operating ranges during normal operation, anticipated operational occurrences, and accident conditions must be OPERABLE. LCO 3.1.7 is required to ensure OPERABILITY of the control and shutdown rod position indicators to determine rod positions and thereby ensure compliance with the control rod alignment and insertion limits.

The OPERABILITY, including position indication, of the shutdown and control rods is an initial assumption in all safety analyses that assume rod insertion upon reactor trip. Maximum rod misalignment is an initial assumption in the safety analysis that directly affects core power distributions and assumptions of available SDM. Rod position indication is required to assess OPERABILITY and misalignment.

Mechanical or electrical failures may cause a rod to become inoperable or to become misaligned from its group. Control rod inoperability or misalignment may cause increased power peaking, resulting from the asymmetric reactivity distribution and a reduction in the total available rod worth for reactor shutdown. Therefore, control rod alignment and OPERABILITY are related to core operation in design power peaking limits and the core design requirement of a minimum SDM.

Limits on control rod alignment and OPERABILITY have been established, and rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

Rod cluster control assemblies (RCCAs), or rods, are moved out of the core (up or withdrawn) or into the core (down or inserted) by their control rod drive mechanisms. The RCCAs are divided among control banks and shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control.

The axial position of shutdown rods and control rods are determined by two separate and independent systems: the Demand Position Indication System (commonly called group step counters) and the Rod Position Indication System.

BASES

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BACKGROUND (continued)

The Demand Position Indication System counts the pulses from the Rod Control System that move the rods. There is one step counter for each group of rods. Individual rods in a group receive the same signal to move and should, therefore, be at the same position indicated by the group step counter for that group. The Demand Position Indication System is considered highly precise ( $\pm 1$  step or  $\pm 5/8$  inch). If a rod does not move one step for each demand pulse, the step counter will still count the pulse and incorrectly reflect the position of the rod.

The Rod Position Indication System provides an indication of actual control rod position, but at a lower precision than the step counters. This system is based on inductive analog signals from a series of coils spaced along a hollow tube. A deviation of  $\pm 12$  steps between the group step counter and a rod position indication is based on normal Rod Position Indication System indication accuracy of  $\pm 5\%$  span with a maximum uncertainty of 10% span between the group step counter and the rod position indication.

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APPLICABLE  
SAFETY  
ANALYSES

Control and shutdown rod position accuracy is essential during power operation. Power peaking, ejected rod worth, or SDM limits may be violated in the event of a Design Basis Accident (Ref. 2), with control or shutdown rods operating outside their limits undetected. Therefore, the acceptance criteria for rod position indication are that rod positions must be known with sufficient accuracy in order to verify the core is operating within the group sequence, overlap, design peaking limits, ejected rod worth, and with minimum SDM (LCO 3.1.5, "Shutdown Bank Insertion Limits," and LCO 3.1.6, "Control Bank Insertion Limits"). The rod positions must also be known in order to verify the alignment limits are preserved (LCO 3.1.4, "Rod Group Alignment Limits"). Rod positions are continuously monitored to provide operators with information that ensures the plant is operating within the bounds of the accident analysis assumptions.

The rod position indicator channels satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii). The rod position indicators monitor rod position, which is an initial condition of the accident.

## BASES

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### LCO

LCO 3.1.7 specifies that one Rod Position Indication System shall be OPERABLE for each rod. Additionally, one Demand Position Indication System shall be OPERABLE for each group within a bank. For the rod position indicators to be OPERABLE requires meeting the SR of the LCO and the following:

- a. The Rod Position Indication System indicates within 12 steps of the group step counter demand position as required by LCO 3.1.4, "Rod Group Alignment Limits,"
- b. For the Rod Position Indication System there are no failed coils, and
- c. The Demand Position Indication System has been calibrated either in the fully inserted position or a check is performed between the two step counters in the same bank. Shutdown Banks C and D each contain a single group. Therefore, validation of movement for Shutdown Banks C and D can only be performed with a comparison of the single group to the corresponding RPI movement.

The 12 step agreement limit between the Demand Position Indication System and the Rod Position Indication System indicates that the Demand Position Indication System is adequately calibrated, and can be used for indication of the measurement of control rod bank position.

A deviation of less than the allowable limit, given in LCO 3.1.4, in position indication for a single control rod, ensures high confidence that the position uncertainty of the corresponding control rod group is within the assumed values used in the analysis (that specified control rod group insertion limits).

These requirements ensure that control rod position indication during power operation and PHYSICS TESTS is accurate, and that design assumptions are not challenged.

OPERABILITY of the position indicator channels ensures that inoperable, misaligned, or mispositioned control rods can be detected. Therefore, power peaking, ejected rod worth, and SDM can be controlled within acceptable limits.

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### APPLICABILITY

The requirements of the Rod Position Indication and step counters are only applicable in MODES 1 and 2 (consistent with LCO 3.1.4, LCO 3.1.5, and LCO 3.1.6), because these are the only MODES in which power is generated, and the OPERABILITY and alignment of rods have the potential to affect the safety of the plant. In the shutdown MODES, the OPERABILITY of the shutdown and control banks has the potential to affect the required SDM, but this effect can be compensated for by an increase in the boron concentration of the Reactor Coolant System.

## BASES

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### ACTIONS

The ACTIONS Table is modified by a Note indicating that a separate Condition entry is allowed for each inoperable rod position indicator and each demand position indicator. This is acceptable because the Required Actions for each Condition provide appropriate compensatory actions for each inoperable position indicator.

A second Note has been added to provide clarification that LCO 3.0.4.a and LCO 3.0.4.b are not applicable for Required Action A.2.1 and A.2.2 following startup from a refueling outage, or following entry into MODE 5 of sufficient duration to safely repair an inoperable rod position indication.

#### A.1

When one Rod Position Indication channel per bank fails, the position of the rod may be determined indirectly by use of the movable incore detectors. Based on experience, normal power operation does not require excessive movement of banks. If a bank has been significantly moved, the Required Action of C.1 or C.2 below is required. Therefore, verification of RCCA position within the Completion Time of 12 hours is adequate for allowing continued full power operation, since the probability of simultaneously having a rod significantly out of position and an event sensitive to that rod position is small.

#### A.2.1 and A.2.2

When one RPI channel per bank fails, the position of the rod may still be determined indirectly by use of the movable incore detectors and reviewing the parameters of the rod control system for indications of unintended rod movement for the rod with the inoperable position indication. Therefore, verification of RCCA position within 8 hours and every 31 days thereafter is adequate for allowing continued full power operation as long as a review of the parameters of the rod control system for indications of unintended rod movement for the rod with the inoperable position indication is performed within 16 hours and every 8 hours thereafter. Furthermore, if the rod control system parameters indicate unintended movement or if the rod with an inoperable position indicator is moved greater than 12 steps, then the verification of the RCCA position must be performed within 8 hours. As long as these compensatory actions are met, reactor operation can then continue until the end of the current cycle or until an entry into MODE 5 of sufficient duration that the repair of the inoperable rod position indication can safely be performed.

Required Actions A.2.1 and A.2.2 are modified by a Note directing that these Required Actions may only be applied to one inoperable rod position indicator.

## BASES

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### ACTIONS (continued)

#### A.3

Reduction of THERMAL POWER to < 50% RTP puts the core into a condition where rod position is not significantly affecting core peaking factors (Ref. 3).

The allowed Completion Time of 8 hours is reasonable, based on operating experience, for reducing power to < 50% RTP from full power conditions without challenging plant systems and allowing for rod position determination by Required Action A.1 above.

#### B.1, B.2, B.3, and B.4

When more than one Rod Position Indication per bank fails, additional actions are necessary to ensure that acceptable power distribution limits are maintained, minimum SDM is maintained, and the potential effects of rod misalignment on associated accident analyses are limited. Placing the Rod Control System in manual assures unplanned rod motion will not occur. Together with the indirect position determination available via movable incore detectors will minimize the potential for rod misalignment. The immediate Completion Time for placing the Rod Control System in manual reflects the urgency with which unplanned rod motion must be prevented while in this Condition.

Monitoring and recording reactor coolant  $T_{avg}$  helps assure that significant changes in power distribution and SDM are avoided. The once per hour Completion Time is acceptable because only minor fluctuations in RCS temperature are expected at steady state plant operating conditions.

The position of the rods may be determined indirectly by use of the movable incore detectors. Verification of control rod position once per 12 hours is adequate for allowing continued full power operation for a limited, 24 hour period, since the probability of simultaneously having a rod significantly out of position and an event sensitive to that rod position is small. The 24 hour Completion Time provides sufficient time to troubleshoot and restore the Rod Position Indication System to operation while avoiding the plant challenges associated with the shutdown without full rod position indication.

Based on operating experience, normal power operation does not require excessive rod movement. If one or more rods has been significantly moved, the Required Action of C.1 or C.2 below is required.

## BASES

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### ACTIONS (continued)

#### C.1 and C.2

These Required Actions clarify that when one or more rods with inoperable position indicators have been moved in excess of 24 steps in one direction, since the position was last determined, the Required Actions of A.1 and A.2, or B.1, as applicable are still appropriate but must be initiated promptly under Required Action C.1 to begin verifying that these rods are still properly positioned, relative to their group positions.

If, the rod positions have not been determined, THERMAL POWER must be reduced to  $< 50\%$  RTP within 8 hours to avoid undesirable power distributions that could result from continued operation at  $\geq 50\%$  RTP, if one or more rods are misaligned by more than 24 steps.

#### D.1.1 and D.1.2

With one demand position indicator per bank inoperable, the rod positions can be determined by the Rod Position Indication System. Since normal power operation does not require excessive movement of rods, verification by administrative means that the rod position indicators are OPERABLE and the most withdrawn rod and the least withdrawn rod are  $\leq 12$  steps apart within the allowed Completion Time of once every 12 hours is adequate.

#### D.2

Reduction of THERMAL POWER to  $< 50\%$  RTP puts the core into a condition where rod position is not significantly affecting core peaking factor limits (Ref. 3). The allowed Completion Time of 8 hours provides an acceptable period of time to verify the rod positions per Required Actions C.1 and C.2 or reduce power to  $< 50\%$  RTP.

#### E.1

If the Required Actions cannot be completed within the associated Completion Time, 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. The allowed Completion Time is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.1.7.1

Verification that the Rod Position Indication agrees with the demand position within 12 steps ensures that the Rod Position Indication is operating correctly. This verification will be performed at 20 steps and 215 steps of rod travel.

This Surveillance is performed prior to reactor criticality after each removal of the reactor head, as there is the potential for unnecessary plant transients if the SR were performed with the reactor at power.

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REFERENCES

1. 10 CFR 50, Appendix A, GDC 13.
  2. UFSAR, Section 7.7.1.
  3. UFSAR, Chapter 15.
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## B 3.1 REACTIVITY CONTROL SYSTEMS

### B 3.1.8 PHYSICS TESTS Exceptions - MODE 2

#### BASES

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##### BACKGROUND

The primary purpose of the MODE 2 PHYSICS TESTS exceptions is to permit relaxations of existing LCOs to allow certain PHYSICS TESTS to be performed.

Section XI of 10 CFR 50, Appendix B (Ref. 1), requires that a test program be established to ensure that structures, systems, and components will perform satisfactorily in service. The 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 plant. Requirements for notification of the NRC, for the purpose of conducting tests and experiments, are specified in 10 CFR 50.59 (Ref. 2).

The key objectives of a test program are to (Ref. 3):

- a. Ensure that the facility has been adequately designed,
- b. Validate the analytical models used in the design and analysis,
- c. Verify the assumptions used to predict unit response,
- d. Ensure that installation of equipment in the facility has been accomplished in accordance with the design, and
- e. Verify that the operating and emergency procedures are adequate.

To accomplish these objectives, testing is performed prior to initial criticality, during startup, during low power operations, during power ascension, at high power, and after each refueling. The PHYSICS TESTS requirements for reload fuel cycles ensure that the operating characteristics of the core are consistent with the design predictions and that the core can be operated as designed (Ref. 4).

PHYSICS TESTS procedures are written and approved in accordance with established formats. The procedures include the information necessary to permit a detailed execution of the testing required to ensure that the design intent is met. PHYSICS TESTS are performed in accordance with these procedures and test results are approved prior to continued power escalation and long term power operation.

## BASES

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### BACKGROUND (continued)

The PHYSICS TESTS required for reload fuel cycles (Ref. 4) in MODE 2 are listed below:

- a. Critical Boron Concentration - Control Rods Withdrawn;
- b. Critical Boron Concentration - Control Rods Inserted;
- c. Control Rod Worth; and
- d. Isothermal Temperature Coefficient (ITC).

These and other supplementary tests may be required to calibrate the nuclear instrumentation or to diagnose operational problems. These tests may cause the operating controls and process variables to deviate from their LCO requirements during their performance.

- a. The Critical Boron Concentration - Control Rods Withdrawn Test measures the critical boron concentration at hot zero power (HZP). With all rods out, the lead control bank is at or near its fully withdrawn position. HZP is where the core is critical ( $k_{\text{eff}} = 1.0$ ), and the Reactor Coolant System (RCS) is at design temperature and pressure for zero power. Performance of this test should not violate any of the referenced LCOs.
- b. The Critical Boron Concentration - Control Rods Inserted Test measures the critical boron concentration at HZP, with a bank having a worth of at least 1%  $\Delta k/k$  when fully inserted into the core. This test is used to measure the boron reactivity coefficient. With the core at HZP and all banks fully withdrawn, the boron concentration of the reactor coolant is gradually lowered in a continuous manner. The selected bank is then inserted to make up for the decreasing boron concentration until the selected bank has been moved over its entire range of travel. The reactivity resulting from each incremental bank movement is measured with a reactivity computer. The difference between the measured critical boron concentration with all rods fully withdrawn and with the bank inserted is determined. The boron reactivity coefficient is determined by dividing the measured bank worth by the measured boron concentration difference. Performance of this test could violate LCO 3.1.4, "Rod Group Alignment Limits," LCO 3.1.5, "Shutdown Bank Insertion Limit," or LCO 3.1.6, "Control Bank Insertion Limits."

BASES

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BACKGROUND (continued)

- c. The Control Rod Worth Test is used to measure the reactivity worth of selected control banks. This test is performed at HZP and has three alternative methods of performance. The first method, the Boron Exchange Method, varies the reactor coolant boron concentration and moves the selected control bank in response to the changing boron concentration. The reactivity changes are measured with a reactivity computer. This sequence is repeated for the remaining control banks. The second method, the Rod Swap Method, measures the worth of a predetermined reference bank using the Boron Exchange Method above. The reference bank is then nearly fully inserted into the core. The selected bank is then inserted into the core as the reference bank is withdrawn. The HZP critical conditions are then determined with the selected bank fully inserted into the core. The worth of the selected bank is inferred, based on the position of the reference bank with respect to the selected bank. This sequence is repeated as necessary for the remaining control banks. The third method, the Boron Endpoint Method, moves the selected control bank over its entire length of travel and then varies the reactor coolant boron concentration to achieve HZP criticality again. The difference in boron concentration is the worth of the selected control bank. This sequence is repeated for the remaining control banks. Performance of this test could violate LCO 3.1.4, LCO 3.1.5, or LCO 3.1.6.
  
- d. The ITC Test measures the ITC of the reactor. This test is performed at HZP and has two methods of performance. The first method, the Slope Method, varies RCS temperature in a slow and continuous manner. The reactivity change is measured with a reactivity computer as a function of the temperature change. The ITC is the slope of the reactivity versus the temperature plot. The test is repeated by reversing the direction of the temperature change, and the final ITC is the average of the two calculated ITCs. The second method, the Endpoint Method, changes the RCS temperature and measures the reactivity at the beginning and end of the temperature change. The ITC is the total reactivity change divided by the total temperature change. The test is repeated by reversing the direction of the temperature change, and the final ITC is the average of the two calculated ITCs. Performance of this test could violate LCO 3.4.2, "RCS Minimum Temperature for Criticality."

BASES

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APPLICABLE  
SAFETY  
ANALYSES

The fuel is protected by LCOs that preserve the initial conditions of the core assumed during the safety analyses. The methods for development of the LCOs that are excepted by this LCO are described in the Core Operating Limit Methodology for Westinghouse Designed PWRs (Ref. 5). The above mentioned PHYSICS TESTS, and other tests that may be required to calibrate nuclear instrumentation or to diagnose operational problems, may require the operating control or process variables to deviate from their LCO limitations.

The UFSAR defines requirements for initial testing of the facility, including PHYSICS TESTS. Table 14.1-2 summarizes the zero, low power, and power tests. Requirements for reload fuel cycle PHYSICS TESTS are defined in ANSI/ANS-19.6.1-1997 (Ref. 4). Although these PHYSICS TESTS are generally accomplished within the limits for the LCOs, conditions may occur when one or more LCOs must be suspended to make completion of PHYSICS TESTS possible or practical. This is acceptable as long as the fuel design criteria are not violated. When one or more of the requirements specified in LCO 3.1.3, "Moderator Temperature Coefficient (MTC)," LCO 3.1.4, LCO 3.1.5, LCO 3.1.6, and LCO 3.4.2 are suspended for PHYSICS TESTS, the fuel design criteria are preserved as long as the power level is limited to  $\leq 5\%$  RTP, the reactor coolant temperature is kept  $\geq 531^\circ\text{F}$ , and SDM is within the limits provided in the COLR.

The PHYSICS TESTS include measurement of core nuclear parameters or the exercise of control components that affect process variables. Among the process variables involved are AFD and QPTR, representing initial conditions of the unit safety analyses. Also involved are the movable control components (control and shutdown rods), that are required to shut down the reactor. The limits for these variables are specified for each fuel cycle in the COLR.

As described in LCO 3.0.7, compliance with Test Exception LCOs is optional, and therefore no criteria of 10 CFR 50.36(c)(2)(ii) apply. Test Exception LCOs provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs. A discussion of the criteria satisfied for the other LCOs is provided in their respective Bases.

BASES

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LCO This LCO allows the reactor parameters of MTC and minimum temperature for criticality to be outside their specified limits. In addition, it allows selected control and shutdown rods to be positioned outside of their specified alignment and insertion limits. One power range neutron flux channel may be bypassed, reducing the number of required channels from 4 to 3. Operation beyond specified limits is permitted for the purpose of performing PHYSICS TESTS and poses no threat to fuel integrity, provided the SRs are met.

The requirements of LCO 3.1.3, LCO 3.1.4, LCO 3.1.5, LCO 3.1.6, and LCO 3.4.2 may be suspended and the number of required channels for LCO 3.3.1, "RTS Instrumentation," Functions 2, 3, 6 and 16.e may be reduced to 3 required channels during the performance of PHYSICS TESTS provided:

- a. RCS lowest loop average temperature is  $\geq 531^{\circ}\text{F}$ ,
- b. SDM is within the limits provided in the COLR, and
- c. THERMAL POWER is  $\leq 5\%$  RTP.

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APPLICABILITY This LCO is applicable when performing low power PHYSICS TESTS. The Applicability is stated as "during PHYSICS TESTS initiated in MODE 2" to ensure that the 5% RTP maximum power level is not exceeded. Should the THERMAL POWER exceed 5% RTP, and consequently the unit enter MODE 1, this Applicability statement prevents exiting this Specification and its Required Actions.

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ACTIONS A.1 and A.2

If the SDM requirement is not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. The operator should begin boration with the best source available for the plant conditions. Boration will be continued until SDM is within limit.

Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable LCOs to within specification.

B.1

When THERMAL POWER is  $> 5\%$  RTP, the only acceptable action is to open the reactor trip breakers (RTBs) to prevent operation of the reactor beyond its design limits. Immediately opening the RTBs will shut down the reactor and prevent operation of the reactor outside of its design limits.

BASES

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ACTIONS (continued)

C.1

When the RCS lowest  $T_{avg}$  is  $< 531^{\circ}\text{F}$ , the appropriate action is to restore  $T_{avg}$  to within its specified limit. The allowed Completion Time of 15 minutes provides time for restoring  $T_{avg}$  to within limits without allowing the plant to remain in an unacceptable condition for an extended period of time. Operation with the reactor critical and with temperature below  $531^{\circ}\text{F}$  could violate the assumptions for accidents analyzed in the safety analyses.

D.1

If the Required Actions cannot be completed within the associated Completion Time, 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 an additional 15 minutes. The Completion Time of 15 additional minutes is reasonable, based on operating experience, for reaching MODE 3 in an orderly manner and without challenging plant systems.

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SURVEILLANCE  
REQUIREMENTS

SR 3.1.8.1

The power range and intermediate range neutron detectors must be verified to be OPERABLE in MODE 2 by LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation." A CHANNEL OPERATIONAL TEST is performed on each power range and intermediate range channel prior to initiation of the PHYSICS TESTS. This will ensure that the RTS is properly aligned to provide the required degree of core protection during the performance of the PHYSICS TESTS.

SR 3.1.8.2

Verification that the RCS lowest loop  $T_{avg}$  is  $\geq 531^{\circ}\text{F}$  will ensure that the unit is not operating in a condition that could invalidate the safety analyses. Verification of the RCS temperature during the performance of the PHYSICS TESTS will ensure that the initial conditions of the safety analyses are not violated.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.8.3

Verification that the THERMAL POWER is  $\leq 5\%$  RTP will ensure that the plant is not operating in a condition that could invalidate the safety analyses. Verification of the THERMAL POWER during the performance of the PHYSICS TESTS will ensure that the initial conditions of the safety analyses are not violated.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.1.8.4

The SDM is verified by performing a reactivity balance calculation, considering the following reactivity effects:

- a. RCS boron concentration,
- b. Control bank position,
- c. RCS average temperature,
- d. Fuel burnup based on gross thermal energy generation,
- e. Xenon concentration,
- f. Samarium concentration,
- g. Isothermal temperature coefficient (ITC), when below the point of adding heat (POAH),
- h. Moderator temperature defect, when above the POAH, and
- i. Doppler defect, when above the POAH.

Using the ITC accounts for Doppler reactivity in this calculation when the reactor is subcritical or critical but below the POAH, and the fuel temperature will be changing at the same rate as the RCS.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

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- REFERENCES
1. 10 CFR 50, Appendix B, Section XI.
  2. 10 CFR 50.59.
  3. Regulatory Guide 1.68, Revision 2, August, 1978.
  4. ANSI/ANS-19.6.1-1997.
  5. BAW-10163P-A, "Core Operating Limit Methodology for Westinghouse Designed PWRs," June 1989.
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## B 3.2 POWER DISTRIBUTION LIMITS

### B 3.2.1 Heat Flux Hot Channel Factor (F<sub>Q</sub>(X,Y,Z))

#### BASES

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**BACKGROUND** The purpose of the limits on the values of F<sub>Q</sub>(X,Y,Z) is to limit the local (i.e., pellet) peak power density. The value of F<sub>Q</sub>(X,Y,Z) varies along the axial height (Z) of the core and by assembly location, X, Y.

F<sub>Q</sub>(X,Y,Z) is defined as the maximum local fuel rod linear power density divided by the average fuel rod linear power density. Therefore, F<sub>Q</sub>(X,Y,Z) is a measure of the peak fuel pellet power within the reactor core.

During power operation, the global power distribution is limited by LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO(QPTR)," which are directly and continuously measured process variables. These LCOs, along with LCO 3.1.6, "Control Bank Insertion Limits," maintain the core limits on power distributions on a continuous basis.

F<sub>Q</sub>(X,Y,Z) varies with fuel loading patterns, control bank insertion, fuel burnup, and changes in axial power distribution.

F<sub>Q</sub>(X,Y,Z) is measured periodically using the incore detector system. These measurements are generally taken with the core at or near equilibrium conditions.

Using the measured three dimensional power distributions, it is possible to derive a measured value for F<sub>Q</sub>(X,Y,Z). However, because this value represents an equilibrium condition, it does not include the variations in the value of F<sub>Q</sub>(X,Y,Z) which are present during nonequilibrium situations such as load following or power ascension.

To account for these possible variations, "the F<sub>Q</sub>(X,Y,Z) limits, BQDES(X,Y,Z) and BCDES(X,Y,Z), have been adjusted by pre-calculated factors (MQ(X,Y,Z) and MC(X,Y,Z) respectively) to account for the calculated worst case transient conditions."

Core monitoring and control under non-equilibrium conditions are accomplished by operating the core within the limits of the appropriate LCOs, including the limits on AFD, QPTR, and control rod insertion.

BASES

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APPLICABLE  
SAFETY  
ANALYSES

This LCO precludes core power distributions that violate the following fuel design criteria:

- a. During a large break loss of coolant accident (LOCA), the peak cladding temperature must not exceed 2200°F (Ref. 1),
- b. During a loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a departure from nucleate boiling (DNB) condition,
- c. During an ejected rod accident, the energy deposition to the fuel must not exceed 280 cal/gm (Ref. 2), and
- d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 3).

Limits on F<sub>Q</sub>(X,Y,Z) ensure that the value of the initial total peaking factor assumed in the accident analyses remains valid. Other criteria must also be met (e.g., maximum cladding oxidation, maximum hydrogen generation, coolable geometry, and long term cooling). However, the peak cladding temperature is typically most limiting.

F<sub>Q</sub>(X,Y,Z) limits assumed in the LOCA analysis are typically limiting relative to (i.e., lower than) the F<sub>Q</sub>(X,Y,Z) limit assumed in safety analyses for other postulated accidents. Therefore, this LCO provides conservative limits for other postulated accidents.

F<sub>Q</sub>(X,Y,Z) satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

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LCO

The Heat Flux Hot Channel Factor, F<sub>Q</sub>(X,Y,Z), shall be limited by the following relationships:

$$F_Q(X,Y,Z) \leq (F_Q^{RTP} / P) K(Z) \text{ for } P > 0.5$$

$$F_Q(X,Y,Z) \leq (F_Q^{RTP} / 0.5) K(Z) \text{ for } P \leq 0.5$$

where: F<sub>Q</sub><sup>RTP</sup> is the F<sub>Q</sub>(X,Y,Z) limit at RTP provided in the COLR,

K(Z) is the normalized F<sub>Q</sub>(X,Y,Z) as a function of core height provided in the COLR, and

$$P = \text{THERMAL POWER} / \text{RTP}$$

## BASES

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### LCO (continued)

For SQN, the actual values of  $F_Q^{RTP}$  and  $K(Z)$  are given in the COLR; however,  $F_Q^{RTP}$  is normally a number on the order of 2.62.

Measured  $F_Q(X,Y,Z)$  is compared against three limits:

- Steady state limit,  $(F_Q^{RTP}/P) * K(Z)$ ,
- Limiting condition LOCA limit,  $BQDES(X,Y,Z)$ , and
- Limiting condition centerline fuel melt limit,  $BCDES(X,Y,Z)$ .

$F_Q(X,Y,Z)$  is approximated by  $F_Q^C(X, Y, Z)$  for the steady state limit on  $F_Q(X,Y,Z)$ . An  $F_Q^C(X, Y, Z)$  evaluation requires using the moveable incore detectors to obtain a power distribution map in MODE 1. From the incore flux map results we obtain the measured value ( $F_Q^M(X,Y,Z)$ ) of  $F_Q(X,Y,Z)$ . Then,

$$F_Q^C(X, Y, Z) = \text{overall measured } F_Q(X,Y,Z) * 1.05 * 1.03$$

where, 1.05 is the measurement reliability factor that accounts for flux map measurement uncertainty (Reference 5) and 1.03 is the local engineering heat flux hot channel factor to account for fuel rod manufacturing tolerance (Reference 4).

$BQDES(X,Y,Z)$  and  $BCDES(X,Y,Z)$  are cycle dependent design limits to ensure the  $F_Q(X,Y,Z)$  limit is met during transients. An evaluation of these limits requires obtaining an incore flux map in MODE 1. From the incore flux map results we obtain the assembly nodal measured value ( $F_Q^M(X, Y, Z)$ ) of  $F_Q(X,Y,Z)$ .  $F_Q^M(X, Y, Z)$  is compared directly to the limits  $BQDES(X,Y,Z)$  and  $BCDES(X,Y,Z)$ . This is appropriate since  $BQDES(X,Y,Z)$  and  $BCDES(X,Y,Z)$  have been adjusted for uncertainties.

The expression for  $BQDES(X,Y,Z)$  is:  $BQDES(X,Y,Z) = P^d(X,Y,Z) * MQ(X,Y,Z) * NRF * F1 / MRF$

## BASES

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### LCO (continued)

where:

- BQDES(X,Y,Z) is the cycle dependent maximum allowable design peaking factor for fuel assembly X,Y at axial location Z. BQDES(X,Y,Z) ensures that the LOCA limit will be preserved for operation within the LCO limits, including allowances for calculational and measurement uncertainties;
- P<sup>d</sup>(X,Y,Z) is the design power distribution for fuel assembly X,Y at axial location Z, including the operational flexibility factor;
- MQ(X,Y,Z) is the minimum available margin ratio for the LOCA limit at assembly X,Y and axial location Z;
- NRF is the nuclear reliability factor;
- F1 is the spacer grid factor;
- MRF is measurement reliability factor.

The expression for BCDES(X,Y,Z) is:  $BCDES(X,Y,Z) = P^d(X,Y,Z) * MC(X,Y,Z) * NRF * F1 / MRF$

where:

- BCDES(X,Y,Z) is the cycle dependent maximum allowable design peaking factor for fuel assembly X,Y, at axial location Z. BCDES(X,Y,Z) ensures that the centerline fuel melt limit will be preserved for operation within the LCO limits, including allowances for calculational and measurement uncertainties;
- MC(X,Y,Z) is the minimum available margin ratio for the centerline fuel melt limit at assembly X,Y and axial location Z;

The reactor core is operating as designed if the measured steady state core power distribution agrees with prediction within statistical variation. This guarantees that the operating limits will preserve the thermal criteria in the applicable safety analyses. The core is operating as designed if the following relationship is satisfied:

$$F_Q^M(X, Y, Z) \leq BQNOM(X, Y, Z)$$

where:

- BQNOM(X,Y,Z) is the nominal design peaking factor for assembly X,Y at axial location Z increased by an allowance for the expected deviation between the measured and predicted design power distribution.

BASES

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LCO (continued)

The F<sub>Q</sub>(X,Y,Z) limits define limiting values for core power peaking that precludes peak cladding temperatures above 2200°F during either a large or small break LOCA.

BQNOM (X,Y,Z), BQDES(X,Y,Z), and BCDES(X,Y,Z) Data bases are provided for the plant power distribution analysis computer codes on a cycle specific basis and are determined using the methodology for core limit generation described in the references in the COLR.

This LCO requires operation within the bounds assumed in the safety analyses. Calculations are performed in the core design process to confirm that the core can be controlled in such a manner during operation that it can stay within the LOCA F<sub>Q</sub>(X,Y,Z) limits. If F<sub>Q</sub><sup>C</sup>(X,Y,Z) cannot be maintained within the LCO limits, reduction of the core power is required. Note that sufficient reduction of the AFD limits will also result in a reduction of the core power.

Violating the LCO limits for F<sub>Q</sub>(X,Y,Z) produces unacceptable consequences if a design basis event occurs while F<sub>Q</sub>(X,Y,Z) is outside its specified limits.

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APPLICABILITY

The F<sub>Q</sub>(X,Y,Z) limits must be maintained in MODE 1 to prevent core power distributions from exceeding the limits assumed in the safety analyses. Applicability in other MODES is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the reactor coolant to require a limit on the distribution of core power.

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ACTIONS

A.1

Reducing THERMAL POWER by ≥ 1% RTP for each 1% by which F<sub>Q</sub><sup>C</sup>(X,Y,Z) exceeds its limit, maintains an acceptable absolute power density. F<sub>Q</sub><sup>C</sup>(X,Y,Z) is F<sub>Q</sub><sup>M</sup>(X,Y,Z) multiplied by a factor accounting for manufacturing tolerances and measurement uncertainties. F<sub>Q</sub><sup>M</sup>(X,Y,Z) is the measured value of F<sub>Q</sub>(X,Y,Z). The Completion Time of 15 minutes provides an acceptable time to reduce power in an orderly manner and without allowing the plant to remain in an unacceptable condition for an extended period of time. The maximum allowable power level initially determined by Required Action A.1 may be affected by subsequent

BASES

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## ACTIONS (continued)

determinations of  $F_Q^C(X,Y,Z)$  and would require power reductions within 15 minutes of the  $F_Q^C(X,Y,Z)$  determination, if necessary to comply with the decreased maximum allowable power level. Decreases in  $F_Q^C(X,Y,Z)$  would allow increasing the maximum allowable power level and increasing power up to this revised limit.

A.2

Required Action A.2 requires an administrative reduction of the AFD limits by  $\geq 1\%$  for each 1% by which  $F_Q^C(X,Y,Z)$  exceeds the steady state limit. The allowed Completion Time of 2 hours, restricts the axial flux distribution such that even if a transient occurred, core peaking factor limits are not exceeded. The maximum allowable AFD limits initially determined by Required Action A.2 may be affected by subsequent determinations of  $F_Q^C(X,Y,Z)$  and would require further AFD limit reductions within 2 hours of the  $F_Q^C(X,Y,Z)$  determination, if necessary to comply with the decreased maximum allowable AFD limits. Decreases in  $F_Q^C(X,Y,Z)$  would allow increasing the maximum allowable AFD limits.

A.3

Reduction in the Overpower  $\Delta T$  trip setpoints (value of  $K_4$ ) by  $\geq 1\%$  in  $\Delta T$  span for each 1% by which  $F_Q^C(X,Y,Z)$  exceeds its limit, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 48 hours is sufficient considering the small likelihood of a severe transient in this time period, and the preceding prompt reduction in THERMAL POWER in accordance with Required Action A.1. The maximum allowable Overpower  $\Delta T$  trip setpoints initially determined by Required Action A.3 may be affected by subsequent determinations of  $F_Q^C(X,Y,Z)$  and would require Overpower  $\Delta T$  trip setpoint reductions within 48 hours of the  $F_Q^C(X,Y,Z)$  determination, if necessary to comply with the decreased maximum allowable Overpower  $\Delta T$  trip setpoints. Decreases in  $F_Q^C(X,Y,Z)$  would allow increasing the maximum allowable Overpower  $\Delta T$  trip setpoints.

BASES

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## ACTIONS (continued)

A.4

A reduction of the Power Range Neutron Flux - High trip setpoints by  $\geq 1\%$  for each 1% by which  $F_Q^C(X,Y,Z)$  exceeds its limit, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 72 hours is sufficient considering the small likelihood of a severe transient in this time period and the preceding prompt reduction in THERMAL POWER in accordance with Required Action A.1. The maximum allowable Power Range Neutron Flux - High trip setpoints initially determined by Required Action A.4 may be affected by subsequent determinations of  $F_Q^C(X,Y,Z)$  and would require Power Range Neutron Flux - High trip setpoint reductions within 72 hours of the  $F_Q^C(X,Y,Z)$  determination, if necessary to comply with the decreased maximum allowable Power Range Neutron Flux - High trip setpoints. Decreases in  $F_Q^C(X,Y,Z)$  would allow increasing the maximum allowable Power Range Neutron Flux - High trip setpoints.

A.5

Verification that  $F_Q^C(X,Y,Z)$  has been restored to within its steady state and transient limit, by performing SR 3.2.1.1, SR 3.2.1.2 and SR 3.2.1.3 prior to increasing THERMAL POWER above the limit imposed by Required Action A.1, ensures that core conditions during operation at higher power levels and future operation are consistent with safety analyses assumptions.

Since  $F_Q^C(X,Y,Z)$  exceeds the steady state limit, the limiting condition operational limit (BQDES) and the limiting condition Reactor Protection System limit (BCDES) may also be exceeded. By performing SR 3.2.1.2 and SR 3.2.1.3, appropriate actions with respect to reductions in AFD limits and OPΔT trip setpoints will be performed, ensuring that core conditions during operational and Condition II transients are maintained within the bounds of the safety analysis.

Condition A is modified by a Note that requires Required Action A.5 to be performed whenever the Condition is entered. This ensures that SR 3.2.1.1, SR 3.2.1.2 and SR 3.2.1.3 will be performed prior to increasing THERMAL POWER above the limit of Required Action A.1, even when Condition A is exited prior to performing Required Action A.5. Performance of SR 3.2.1.1, SR 3.2.1.2 and SR 3.2.1.3 are necessary to assure  $F_Q^C(X,Y,Z)$  is properly evaluated prior to increasing THERMAL POWER.

BASES

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ACTIONS (continued)

B.1 and B.2

The F<sub>Q</sub>(X,Y,Z) margin supporting AFD operational limits (AFD margin) during transient operations is based on the relationship between F<sub>Q</sub><sup>M</sup>(X,Y,Z) and the limiting condition operational limit, BQDES (X,Y,Z), as follows:

$$\%AFD \text{ margin} = \left( 1 - \frac{F_Q^M(X,Y,Z)}{BQDES(X,Y,Z)} \right) * 100\%$$

The AFD min margin = minimum % margin value of all locations examined. If the reactor core is operating as designed, then F<sub>Q</sub><sup>M</sup>(X,Y,Z) is less than BQDES (X,Y,Z) and calculation of %AFD margin is not required. If the AFD margin is less than zero, then F<sub>Q</sub><sup>M</sup>(X,Y,Z) is greater than BQDES (X,Y,Z) and the AFD limits may not be adequate to prevent exceeding the peaking criteria for a LOCA if a normal operational transient occurs.

Required Actions B.1 and B.2 require reducing the AFD limit lines as follows. The AFD limit reduction is from the full power AFD limits. The adjusted AFD limits must be used until a new measurement shows that a smaller adjustment can be made to the AFD limits, or that no adjustment is necessary:

APL = PAFDL – Absolute Value of (PSLOPE<sup>AFD</sup> \* % AFD Margin)  
ANL = NAFDL + Absolute Value of (NSLOPE<sup>AFD</sup> \* % AFD Margin)

where,

- APL is the adjusted positive AFD limit.
- ANL is the adjusted negative AFD limit.
- PAFDL is the positive AFD limit defined in the COLR.
- NAFDL is the negative AFD limit defined in the COLR.
- PSLOPE<sup>AFD</sup> is the adjustment to the positive AFD limit required to compensate for each 1% that F<sub>Q</sub><sup>M</sup>(X,Y,Z) exceeds BQDES (X,Y,Z) as defined in the COLR.
- NSLOPE<sup>AFD</sup> is the adjustment to the negative AFD limit required to compensate for each 1% that F<sub>Q</sub><sup>M</sup>(X,Y,Z) exceeds BQDES (X,Y,Z) as defined in the COLR.
- % AFD Margin is the most negative margin determined above.

BASES

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ACTIONS (continued)

Completing Required Actions B.1 and B.2 within the allowed Completion Time of 2 hours, restricts the axial flux distribution such that even if a transient occurred, core peaking factor limits are not exceeded.

C.1 and C.2

The F<sub>Q</sub>(X,Y,Z) margin supporting the Overpower ΔT f<sub>2</sub>(ΔI) breakpoints (f<sub>2</sub>(ΔI) margin) during transient operations is based on the relationship between F<sub>Q</sub><sup>M</sup>(X,Y,Z) and the limit, BCDES(X,Y,Z), as follows:

$$\% f_2(\Delta I) \text{ margin} = \left( 1 - \frac{F_Q^M(X,Y,Z)}{BCDES(X,Y,Z)} \right) * 100\%$$

The f<sub>2</sub>(ΔI) min margin = minimum % margin value of all locations examined. If the reactor core is operating as designed, then F<sub>Q</sub><sup>M</sup>(X,Y,Z) is less than BCDES(X,Y,Z) and calculation of % f<sub>2</sub>(ΔI) margin is not required. If the f<sub>2</sub>(ΔI) margin is less than zero, then F<sub>Q</sub><sup>M</sup>(X,Y,Z) is greater than BCDES(X,Y,Z) and there is a potential that the f<sub>2</sub>(ΔI) limits are insufficient to preclude centerline fuel melt during certain transients.

Required Actions C.1 and C.2 require reducing the f<sub>2</sub>(ΔI) breakpoint limits as follows. The f<sub>2</sub>(ΔI) breakpoint limit reduction is always from the full power f<sub>2</sub>(ΔI) breakpoint limits. The adjusted f<sub>2</sub>(ΔI) breakpoint limits must be used until a new measurement shows that a smaller adjustment can be made to the f<sub>2</sub>(ΔI) breakpoint limits, or that no adjustment is necessary.

$$\text{Pos}f_2(\Delta I)_{\text{Adjusted}} = \text{Pos}f_2(\Delta I)^{\text{Limit}} - \text{Absolute Value of (PSLOPE}^{f_2(\Delta I)} * \% f_2(\Delta I) \text{ Margin)}$$

$$\text{Neg}f_2(\Delta I)_{\text{Adjusted}} = \text{Neg}f_2(\Delta I)^{\text{Limit}} + \text{Absolute Value of (NSLOPE}^{f_2(\Delta I)} * \% f_2(\Delta I) \text{ Margin)}$$

where:

- Posf<sub>2</sub>(ΔI)<sub>Adjusted</sub> is the adjusted OPΔT positive f<sub>2</sub>(ΔI) breakpoint limit.
- Negf<sub>2</sub>(ΔI)<sub>Adjusted</sub> is the adjusted OPΔT negative f<sub>2</sub>(ΔI) breakpoint limit.
- Posf<sub>2</sub>(ΔI)<sup>Limit</sup> is the OPΔT positive f<sub>2</sub>(ΔI) breakpoint limit defined in the COLR.
- Negf<sub>2</sub>(ΔI)<sup>Limit</sup> is the OPΔT negative f<sub>2</sub>(ΔI) breakpoint limit defined in the COLR.

BASES

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ACTIONS (continued)

- PSLOPE<sup>f<sub>2</sub>(ΔI)</sup> is the adjustment to the positive OPΔT f<sub>2</sub>(ΔI) limit required to compensate for each 1% that F<sub>Q</sub><sup>M</sup>(X,Y,Z) exceeds BCDES(X,Y,Z) as defined in the COLR.
- NSLOPE<sup>f<sub>2</sub>(ΔI)</sup> is the adjustment to the negative OPΔT f<sub>2</sub>(ΔI) limit required to compensate for each 1% that F<sub>Q</sub><sup>M</sup>(X,Y,Z) exceeds BCDES(X,Y,Z) as defined in the COLR.
- % f<sub>2</sub>(ΔI) Margin is the most negative margin determined above.

Completing Required Actions C.1 and C.2 is a conservative action for protection against the consequences of transients since this adjustment limits the peak transient power level which can be achieved during an anticipated operational occurrence. Completing Required Actions C.1 and C.2 within the allowed Completion Time of 48 hours is sufficient considering the small likelihood of a limiting transient in this time period.

D.1

If Required Actions A.1 through A.5, B.1, B.2, C.1 or C.2 are not met within their associated Completion Times, the plant must be placed in a mode or condition in which the LCO requirements are not applicable. This is done by placing the plant in at least MODE 2 within 6 hours.

This allowed Completion Time is reasonable based on operating experience regarding the amount of time it takes to reach MODE 2 from full power operation in an orderly manner and without challenging plant systems.

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SURVEILLANCE  
REQUIREMENTS

SR 3.2.1.1, SR 3.2.1.2 and SR 3.2.1.3 are modified by a Note. It states that Surveillance performance is not required until 12 hours after an equilibrium power level has been achieved at which a power distribution map can be obtained. This allowance is modified, however, by one of the Frequency conditions that requires verification that F<sub>Q</sub><sup>C</sup>(X,Y,Z) and F<sub>Q</sub><sup>M</sup>(X,Y,Z) are within their specified limits after a power rise of more than 10% RTP over the THERMAL POWER at which they were last verified to be within specified limits. Because F<sub>Q</sub><sup>C</sup>(X,Y,Z) and F<sub>Q</sub><sup>M</sup>(X,Y,Z) could not have previously been measured in this reload core, there is a second Frequency condition, applicable only for reload cores, that requires determination of these parameters before exceeding 75% RTP. This ensures that some determination of F<sub>Q</sub><sup>C</sup>(X,Y,Z) and F<sub>Q</sub><sup>M</sup>(X,Y,Z) are made at a lower power level at which adequate margin is available before going to 100% RTP. Also, this Frequency condition, together with the Frequency condition requiring verification of F<sub>Q</sub><sup>C</sup>(X,Y,Z) and F<sub>Q</sub><sup>M</sup>(X,Y,Z) following a power increase of more than 10%, ensures that

## BASES

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### SURVEILLANCE REQUIREMENTS (continued)

they are verified as soon as RTP (or any other level for extended operation) is achieved. In the absence of these Frequency conditions, it is possible to increase power to RTP and operate for 31 days without verification of  $F_Q^C(X,Y,Z)$  and  $F_Q^M(X,Y,Z)$ . The Frequency condition is not intended to require verification of these parameters after every 10% increase in power level above the last verification. It only requires verification after a power level is achieved for extended operation that is 10% higher than that power at which  $F_Q(X,Y,Z)$  was last measured.

#### SR 3.2.1.1

Direct verification that  $F_Q^C(X,Y,Z)$  is within its specified limits involves increasing the overall measured  $F_Q(X,Y,Z)$  to allow for manufacturing tolerance and measurement uncertainties in order to obtain  $F_Q^C(X,Y,Z)$ .  $F_Q^C(X,Y,Z)$  is then compared to its specified limits.

The limit with which  $F_Q^C(X,Y,Z)$  is compared varies inversely with power above 50% RTP and directly with a function called  $K(Z)$  provided in the COLR.

The surveillance has been modified by a Note providing an allowance to not perform SR 3.2.1.1 if the Surveillance has been determined to be met based on the performance results of both SR 3.2.1.2 and SR 3.2.1.3. If both the AFD min margin and the  $f_2(\Delta I)$  min margin are  $\geq 0$ , then the steady state limit is met because these margins represent bounding limiting conditions. However, if AFD min margin or  $f_2(\Delta I)$  min margin is negative then a direct evaluation of the steady state limit is required to satisfy this surveillance requirement.

Performing this Surveillance in MODE 1 prior to exceeding 75% RTP ensures that the  $F_Q^C(X,Y,Z)$  limit is met when RTP is achieved, because peaking factors generally decrease as power level is increased.

If THERMAL POWER has been increased by  $\geq 10\%$  RTP since the last determination of  $F_Q^C(X,Y,Z)$ , another evaluation of this factor is required 12 hours after achieving equilibrium conditions at this higher power level (to ensure that  $F_Q^C(X,Y,Z)$  values are being reduced sufficiently with power increase to stay within the LCO limits).

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.2.1.2 and SR 3.2.1.3

The nuclear design process includes calculations performed to determine that the core can be operated within the  $F_Q(X,Y,Z)$  limits. Because flux maps are taken in steady state conditions, the variations in power distribution resulting from normal operational maneuvers are not present in the flux map data. These variations are, however, conservatively calculated by considering a wide range of unit maneuvers in normal operation. The maximum peaking factor increase over steady state values, calculated as a function of both assembly and axial location  $(X,Y,Z)$ , has been included in the cycle specific limits  $BQDES(X,Y,Z)$  and  $BCDES(X,Y,Z)$  using margin factors  $MQ(X,Y,Z)$  and  $MC(X,Y,Z)$ , respectively (Reference 5).

No uncertainties are applied to  $F_Q^M(X,Y,Z)$  because the limits,  $BQDES(X,Y,Z)$  and  $BCDES(X,Y,Z)$ , have been adjusted for uncertainties.

$BQDES(X,Y,Z)$  and  $BCDES(X,Y,Z)$  limits are not applicable for the following axial core regions, measured in percent of core height:

- a. Lower core region, from 0 to 15% inclusive and
- b. Upper core region, from 85 to 100% inclusive.

The top and bottom 15% of the core are excluded from the evaluation because of the low probability that these regions would be more limiting in the safety analyses and because of the difficulty of making a precise measurement in these regions.

This Surveillance has been modified by a Note that may require that more frequent surveillances be performed based on future projections. If  $F_Q^M(X,Y,Z)$  is evaluated and found to be within the applicable limiting condition limits, an evaluation is required to account for any increase to  $F_Q^M(X,Y,Z)$  that may occur and cause the  $F_Q(X,Y,Z)$  limit to be exceeded before the next required  $F_Q(X,Y,Z)$  evaluation.

In addition to ensuring via surveillance that the heat flux hot channel factor is within its limits when a measurement is taken, there are also requirements to extrapolate trends in  $F_Q^M(X,Y,Z)$  for the last two measurements out to 31 EFPD beyond the most recent measurement. If the extrapolation yields an  $F_Q^M(X,Y,Z) > BQNOM(X,Y,Z)$ , further consideration is required.

## BASES

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### SURVEILLANCE REQUIREMENTS (continued)

The implications of these extrapolations are considered separately for both the operational and RPS heat flux hot channel factor limits. If the extrapolations of  $F_Q^M(X,Y,Z)$  are unfavorable, additional actions must be taken. These actions are to meet the  $F_Q(X,Y,Z)$  limit with the last  $F_Q^M(X,Y,Z)$  increased by the appropriate factor specified in the COLR or to evaluate  $F_Q^M(X,Y,Z)$  prior to the projected point in time when the extrapolated values are expected to exceed the extrapolated limits. These alternative requirements prevent  $F_Q(X,Y,Z)$  from exceeding its limit for any significant period of time without detection using the best available data.

Extrapolation is not required for the initial flux map taken after reaching equilibrium conditions following a refueling outage since the initial flux map establishes the baseline measurement for future trending.

$F_Q(X,Y,Z)$  is verified at power levels  $\geq 10\%$  RTP above the THERMAL POWER of its last verification within 12 hours after achieving equilibrium conditions to ensure that  $F_Q(X,Y,Z)$  is within its limit at higher power levels.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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### REFERENCES

1. 10 CFR 50.46, 1974.
  2. Regulatory Guide 1.77, Rev. 0, May 1974.
  3. 10 CFR 50, Appendix A, GDC 26.
  4. WCAP-7308-L-P-A, "Evaluation of Nuclear Hot Channel Factor Uncertainties," June 1988.
  5. BAW-10163PA "Core Operating Limit Methodology for Westinghouse-Designed PWRs" June 1989.
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## B 3.2 POWER DISTRIBUTION LIMITS

### B 3.2.2 Nuclear Enthalpy Rise Hot Channel Factor (F<sub>ΔH</sub>(X,Y))

#### BASES

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**BACKGROUND** The purpose of this LCO is to establish limits on the power density at any point in the core so that the fuel design criteria are not exceeded and the accident analysis assumptions remain valid. The design limits on local (pellet) and integrated fuel rod peak power density are expressed in terms of hot channel factors. Control of the core power distribution with respect to these factors ensures that local conditions in the fuel rods and coolant channels do not challenge core integrity at any location during either normal operation or a postulated accident analyzed in the safety analyses.

F<sub>ΔH</sub>(X,Y) is defined as the ratio of the integral of the linear power along the fuel rod with the highest integrated power to the average integrated fuel rod power. Therefore, F<sub>ΔH</sub>(X,Y) is a measure of the maximum total power produced in a fuel rod.

F<sub>ΔH</sub>(X,Y) is sensitive to fuel loading patterns, bank insertion, and fuel burnup. F<sub>ΔH</sub>(X,Y) typically increases with control bank insertion and typically decreases with fuel burnup.

F<sub>ΔH</sub>(X,Y) is not directly measurable but is inferred from a power distribution map obtained with the movable incore detector system. Specifically, the results of the three dimensional power distribution map are analyzed by a computer to determine F<sub>ΔH</sub>(X,Y). An F<sub>ΔH</sub>(X,Y) evaluation requires obtaining an incore flux map in MODE 1. The incore flux map results provide the measured value ( F<sub>ΔH</sub><sup>M</sup>(X, Y) ) of F<sub>ΔH</sub>(X,Y) for each assembly location (X,Y). The F<sub>ΔH</sub> ratio (FDHR) is used in order to determine the F<sub>ΔH</sub> limit for the measured and design power distributions (Ref. 4). Then,

$$F_{\Delta H}^M(X,Y) = \frac{F_{\Delta H}^M(X, Y)}{MAP^M / AXIAL^M(X, Y)}$$

where MAP<sup>M</sup> is the maximum allowable peak from the COLR for the measured assembly power distribution at assembly location (X,Y) which accounts for calculational and measurement uncertainties, and AXIAL<sup>M</sup>(X, Y) is the measured ratio of the peak-to-average axial power at assembly location (X,Y).

BHDES(X,Y) is a cycle dependent design limit to preserve Departure from Nucleate Boiling(DNB) assumed for initial conditions at the time of limiting transients such as a Loss of Flow Accident (LOFA). BRDES(X,Y) is a

BASES

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BACKGROUND (continued)

cycle dependent design limit to preserve reactor protection system safety limits for DNB requirements (Ref. 4).

The expression for BHDES(X,Y) is:

$$\text{BHDES}(X,Y) = F\Delta\text{HR}^d(X,Y) * \text{MH}(X,Y)$$

where:  $F\Delta\text{HR}^d(X,Y) = \frac{F_{\Delta H}^d(X,Y)}{\text{MAP}^d / \text{AXIAL}^d(X,Y)}$

- $\text{MAP}^d$  is the maximum allowable peak from the COLR for the design assembly power distribution at assembly location (X,Y) which accounts for calculational and measurement uncertainties,
- $\text{AXIAL}^d(X,Y)$  is the design ratio of the peak-to-average axial power at assembly location (X,Y),
- $F_{\Delta H}^d(X,Y)$  is the design  $F_{\Delta H}$  assembly location (X, Y), and
- $\text{MH}(X,Y)$  is the minimum available margin ratio for initial condition DNB at the limiting conditions at assembly location (X,Y).

The expression for BRDES(X,Y) is:

$$\text{BRDES}(X,Y) = F\Delta\text{HR}^d(X,Y) * \text{MH}^s(X,Y)$$

where:  $\text{MH}^s(X,Y)$  is the minimum available margin ratio for steady state DNB at the limiting conditions at assembly location (X,Y).

The reactor core is "operating as designed" if the measured steady state core power distribution agrees with prediction within statistical variation. This guarantees that the operating limits will preserve the thermal criteria in the applicable safety analyses. The core is "operating as designed" if the following relationship is satisfied:

$$F\Delta\text{HR}^M(X,Y) \leq \text{BHNOM}(X,Y)$$

where:  $\text{BHNOM}(X,Y)$  is the nominal design radial peaking factor for an assembly at core location (X,Y) increased by an allowance for the expected deviation between the measured and predicted design power distribution. This factor is calculated at least every 31 EFPD. However, during power operation, the global power distribution is monitored by LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," which address directly and continuously measured process variables.

BASES

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BACKGROUND (continued)

The COLR provides peaking factor limits that ensure that the design basis value of the DNB is met for normal operation, operational transients, and any transient condition arising from events of moderate frequency. The DNB design basis precludes DNB and is met by limiting the minimum local DNB heat flux ratio to the design limit value using an NRC approved critical heat flux correlation. All DNB limited transient events are assumed to begin with an  $F_{\Delta H}(X,Y)$  value that satisfies the LCO requirements.

Operation outside the LCO limits may produce unacceptable consequences if a DNB limiting event occurs. The DNB design basis ensures that there is no overheating of the fuel that results in possible cladding perforation with the release of fission products to the reactor coolant.

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APPLICABLE  
SAFETY  
ANALYSES

Limits on  $F_{\Delta H}(X,Y)$  preclude core power distributions that exceed the following fuel design limits:

- a. There must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hottest fuel rod in the core does not experience a DNB condition,
- b. During a large break loss of coolant accident (LOCA), peak cladding temperature (PCT) must not exceed 2200°F,
- c. During an ejected rod accident, the energy deposition to the fuel must not exceed 280 cal/gm (Ref. 1), and
- d. Fuel design limits required by GDC 26 (Ref. 2) for the condition when control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn.

For transients that may be DNB limited, the Reactor Coolant System flow and  $F_{\Delta H}(X,Y)$  are the core parameters of most importance. The limits on  $F_{\Delta H}(X,Y)$  ensure that the DNB design basis is met for normal operation, operational transients, and any transients arising from events of moderate frequency. The DNB design basis is met by limiting the minimum local DNB heat flux ratio to the design limit value using an NRC approved critical heat flux correlation. This value provides a high degree of assurance that the hottest fuel rod in the core does not experience a DNB.

The allowable  $F_{\Delta H}(X,Y)$ ,  $F_{\Delta H}$  min margin and  $f_1(\Delta I)$  min margin, increase with decreasing power level. This functionality in  $F_{\Delta H}(X,Y)$  is

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BASES

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APPLICABLE SAFETY ANALYSES (continued)

included in the analyses that provide the Reactor Core Safety Limits (SLs) of SL 2.1.1. Therefore, any DNB events in which the calculation of the core limits is modeled implicitly use this variable value of F<sub>ΔH</sub>(X,Y) in the analyses. Likewise, all transients that may be DNB limited are assumed to begin with F<sub>ΔH</sub> min margin and f<sub>1</sub> (ΔI) min margin.

The LOCA safety analysis indirectly models F<sub>ΔH</sub>(X,Y) as an input parameter. The Nuclear Heat Flux Hot Channel Factor (F<sub>Q</sub>(X,Y,Z)) and the axial peaking factors are inserted directly into the LOCA safety analyses that verify the acceptability of the resulting peak cladding temperature (Ref. 3).

The fuel is protected in part by Technical Specifications, which ensure that the initial conditions assumed in the safety and accident analyses remain valid. The following LCOs ensure this: LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," LCO 3.1.6, "Control Bank Insertion Limits," LCO 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor F<sub>ΔH</sub>(X,Y)," and LCO 3.2.1, "Heat Flux Hot Channel Factor (F<sub>Q</sub>(X,Y,Z))."

F<sub>ΔH</sub>(X,Y) and F<sub>Q</sub>(X,Y,Z) are indirectly measured periodically using the movable incore detector system. Measurements are generally taken with the core at, or near, steady state conditions. Core monitoring and control under transient conditions (Condition 1 events) are accomplished by operating the core within the limits of the LCOs on AFD, QPTR, and Bank Insertion Limits.

F<sub>ΔH</sub>(X,Y) satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

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LCO	The LCO states that F <sub>ΔH</sub> (X,Y) shall be less than the limits provided in the COLR. This LCO relationship must be satisfied even if the core is operating at limiting conditions. This requires adjustment to the measured F <sub>ΔH</sub> (X,Y) to account for limiting conditions and the differences between design and measured conditions. The adjustments are accounted for by comparing FΔHR <sup>M</sup> (X,Y) to the limits BHDES(X,Y) and BRDES(X,Y). Therefore, if the F <sub>ΔH</sub> min margin is ≥ 0 and f <sub>1</sub> (ΔI) min margin ≥ 0 the LCO is satisfied.
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APPLICABILITY	The F <sub>ΔH</sub> (X,Y) limits must be maintained in MODE 1 to preclude core power distributions from exceeding the fuel design limits for DNBR and PCT. Applicability in other modes is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the coolant to require a limit on the distribution of core power. Specifically, the design bases events that are sensitive to
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BASES

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APPLICABILITY (continued)

F<sub>ΔH</sub>(X,Y) in other modes (MODES 2 through 5) have significant margin to DNB, and therefore, there is no need to restrict F<sub>ΔH</sub>(X,Y) in these modes.

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ACTIONS

The % F<sub>ΔH</sub> margin is based on the relationship between F<sub>ΔHR</sub><sup>M</sup>(X,Y) and the limit, BHDES (X,Y), as follows:

$$\% F_{\Delta H} \text{ Margin} = \left( 1 - \frac{F_{\Delta HR}^M(X,Y)}{BHDES(X,Y)} \right) \times 100\%$$

If the reactor core is "operating as designed", then F<sub>ΔHR</sub><sup>M</sup>(X,Y) is less than BHDES (X,Y) and calculation of %F<sub>ΔH</sub> margin is not required. If the %F<sub>ΔH</sub> margin is less than zero, then F<sub>ΔHR</sub><sup>M</sup>(X,Y) is greater than BHDES (X, Y) and the F<sub>ΔH</sub>(X,Y) limits may not be adequate to prevent exceeding the initial DNB conditions assumed for transients such as a LOFA. BHDES (X,Y) represents the maximum allowable design radial peaking factors which ensures that the initial condition DNB will be preserved for operation within the LCO limits, and includes allowances for calculational and measurement uncertainties. The F<sub>ΔH</sub> min margin is the minimum for all core locations examined.

Condition A is modified by a Note that requires that Required Actions A.3 and A.5 must be completed whenever Condition A is entered. If F<sub>ΔH</sub> min margin < 0 is restored to within limits prior to completion of the THERMAL POWER reduction in Required Action A.1, compliance with Required Actions A.3 and A.5 must be met.

However, if power is reduced below 50% RTP, Required Action A.5 requires that another determination of F<sub>ΔH</sub> min margin must be verified prior to exceeding 50% RTP, prior to exceeding 75% RTP, and within 24 hours after reaching or exceeding 95% RTP.

A.1 and A.2

If the value of F<sub>ΔH</sub> min margin is not restored to within its specified limit either by adjusting a misaligned rod or by reducing THERMAL POWER, the alternative option is to reduce allowable THERMAL POWER from RTP by at least RRH% (where RRH = Thermal power reduction required to compensate for each 1% that F<sub>ΔH</sub>(X,Y) exceeds its limit) multiplied by the F<sub>ΔH</sub> min margin in accordance with Required Action A.1 and reduce the Power Range Neutron Flux - High trip setpoints, as specified in TS Table 3.3.1-1 by ≥ RRH% multiplied times the F<sub>ΔH</sub> min margin in accordance with Required Action A.2. Reducing allowable RTP by at least RRH% multiplied by the F<sub>ΔH</sub> min margin increases the DNB

BASES

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## ACTIONS (continued)

margin and does not likely cause the DNBR limit to be violated in steady state operation. The reduction in trip setpoints ensures that continuing operation remains at an acceptable low power level with adequate DNBR margin. The allowed Completion Time of 2 hours for Required Action A.1 provides an acceptable time to reach the required power level from full power operation without allowing the plant to remain in an unacceptable condition for an extended period of time.

The allowed Completion Time of 72 hours to reset the trip setpoints per Required Action A.2 recognizes that, once power is reduced, the safety analysis assumptions are satisfied and there is no urgent need to reduce the trip setpoints. This is a sensitive operation that may inadvertently trip the Reactor Protection System.

A.3

Once the allowable power level has been reduced by at least RRH% multiplied by the  $F_{\Delta H}$  min margin per Required Action A.1, an incore flux map (SR 3.2.2.1) must be obtained and the  $F_{\Delta H}$  min margin is verified  $\geq 0$  at the lower power level. The unit is provided 22 additional hours to perform this task over and above the 2 hours allowed by Action A.1. The Completion Time of 24 hours is acceptable because of the increase in the DNB margin, which is obtained at lower power levels, and the low probability of having a DNB limiting event within this 24 hour period. Additionally, operating experience has indicated that this Completion Time is sufficient to obtain the incore flux map, perform the required calculations, and evaluate  $F_{\Delta H}$  min margin.

A.4

If the value of  $F_{\Delta HR}^M(X,Y)$  is not restored to within its specified limit, Overtemperature  $\Delta T$  K1 (OT $\Delta T$  K1) term is required to be reduced by at least TRH multiplied by the  $F_{\Delta H}$  min margin. The value of TRH is provided in the COLR. Completing Required Action A.4 ensures protection against the consequences of transients since this adjustment limits the peak transient power level which can be achieved during an anticipated operational occurrence. Also, completing Required Action A.4 within the allowed Completion Time of 48 hours is sufficient considering the small likelihood of a limiting transient in this time period.

BASES

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ACTIONS (continued)

A.5

Verification that F<sub>ΔH</sub> min margin is ≥ 0 after an out of limit occurrence ensures that the cause that led to the F<sub>ΔH</sub> min margin exceeding its limit is corrected, and that subsequent operation proceeds within the LCO limit. This Action demonstrates that the F<sub>ΔH</sub> min margin limit is ≥ 0 prior to exceeding 50% RTP, again prior to exceeding 75% RTP, and within 24 hours after THERMAL POWER is ≥ 95% RTP.

This Required Action is modified by a Note that states that THERMAL POWER does not have to be reduced prior to performing this Action.

B.1

The %f<sub>1</sub>(ΔI) margin is based on the relationship between FΔHR<sup>M</sup>(X,Y) and the limit, BRDES (X,Y), as follows:

$$\% f_1(\Delta I)Margin = \left( 1 - \frac{F\Delta HR^M(X,Y)}{BRDES(X,Y)} \right) \times 100\%$$

If the reactor core is "operating as designed", then FΔHR<sup>M</sup>(X,Y) is less than BRDES (X,Y) and calculation of %f<sub>1</sub>(ΔI) margin is not required. If the %f<sub>1</sub>(ΔI) margin is less than zero, then FΔHR<sup>M</sup>(X,Y) is greater than BRDES (X, Y) and the OTΔT setpoint limits may not be adequate to prevent exceeding DNB requirements.

BRDES (X,Y) represents the maximum allowable design radial peaking factors which ensure that the steady state DNBR limit will be preserved for operation within the LCO limits, including allowances for calculational and measurement uncertainties.

Required Action B.1 requires the reduction of the OTΔT K1 term by at least TRH multiplied by the f<sub>1</sub>(ΔI) min margin. TRH is the amount of OTΔT K1 setpoint reduction required to compensate for each 1% that F<sub>ΔH</sub>(X,Y) exceeds the limit provided in the COLR. Completing Required Action B.1 within the allowed Completion Time of 48 hours, restricts F<sub>ΔH</sub>(X,Y) such that even if a transient occurred, DNB requirements are met. The f<sub>1</sub>(ΔI) min margin is the minimum % of f<sub>1</sub>(ΔI) margin for all core locations examined.

BASES

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ACTIONS (continued)

C.1

When Required Actions A.1 through A.5, and B.1, cannot be completed within their required Completion Times, the plant must be placed in a mode in which the LCO requirements are not applicable. This is done by placing the plant in at least MODE 2 within 6 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience regarding the time required to reach MODE 2 from full power conditions in an orderly manner and without challenging plant systems.

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SURVEILLANCE  
REQUIREMENTS

SR 3.2.2.1 and SR 3.2.2.2 are modified by a Note. It states that, "Not required to be performed until 12 hours after an equilibrium power level has been achieved at which a power distribution map can be obtained." SR 3.2.2.1 and SR 3.2.2.2 require using the incore detector system to provide the necessary data to create a power distribution map. To provide the necessary data, MODE 1 needs to be entered, power escalated, stabilized and equilibrium conditions established at some higher power level. These surveillances could not be satisfactorily performed if the requirement for performance of the Surveillances was included in MODE 2 prior to entering MODE 1.

In a reload core,  $F_{\Delta H}^M(X,Y)$  could not have previously been measured, therefore, there is a Frequency condition, applicable only for reload cores, that requires determination of these parameters before exceeding 75% RTP. This ensures that some determination of  $F_{\Delta H}^M(X,Y)$  is made at a lower power level at which adequate margin is available before going to 100% RTP.

SR 3.2.2.1 and SR 3.2.2.2

In addition to ensuring via Surveillance that the nuclear enthalpy rise hot channel factor is within its limits when a measurement is taken, there are also requirements to extrapolate trends in  $F_{\Delta H}^M(X,Y)$  for the last two measurements out to 31 EFPD beyond the most recent measurement. If the extrapolation yields an  $F_{\Delta HR}^M(X,Y) > BHNOM(X,Y)$ , further consideration is required.

The implications of these extrapolations are considered separately for BHDES(X,Y) and BRDES(X,Y) limits. If the extrapolations of  $F_{\Delta H}^M(X,Y)$  are unfavorable, additional actions must be taken. These actions are to meet the  $F_{\Delta H}(X,Y)$  limit with the last  $F_{\Delta H}^M(X,Y)$  increased by the appropriate factor specified in the COLR or to evaluate  $F_{\Delta H}^M(X,Y)$  prior to the projected

## BASES

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### SURVEILLANCE REQUIREMENTS (continued)

point in time when the extrapolated values are expected to exceed the extrapolated limits. These alternative requirements attempt to prevent  $F_{\Delta H}(X,Y)$  from exceeding its limit for any significant period of time without detection using the best available data.

Extrapolation is not required for the initial flux map taken after reaching equilibrium conditions following a refueling outage since the initial flux map establishes the baseline measurement for future trending.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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### REFERENCES

1. Regulatory Guide 1.77, Rev. 0, May 1974.
  2. 10 CFR 50, Appendix A, GDC 26.
  3. 10 CFR 50.46.
  4. BAW-10163P-A, Revision 0, "Core Operating Limit Methodology for Westinghouse-Designed PWRs," June 1989.
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## B 3.2 POWER DISTRIBUTION LIMITS

### B 3.2.3 AXIAL FLUX DIFFERENCE (AFD)

#### BASES

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**BACKGROUND** The purpose of this LCO is to establish limits on the values of the AFD in order to limit the amount of axial power distribution skewing to either the top or bottom of the core. By limiting the amount of power distribution skewing, core peaking factors are consistent with the assumptions used in the safety analyses. Limiting power distribution skewing over time also minimizes the xenon distribution skewing, which is a significant factor in axial power distribution control.

The AFD limits are selected by considering a range of axial xenon distributions that may occur as a result of large variations of the AFD. Subsequently, power peaking factors and power distributions are examined to ensure that the loss of coolant accident (LOCA), loss of flow accident, and anticipated transient limits are met. Violation of the AFD limits invalidate the conclusions of the accident and transient analyses with regard to fuel cladding integrity.

The AFD is monitored on an automatic basis using the unit process computer, which has an AFD monitor alarm. The computer determines the 1 minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for two or more OPERABLE excore channels is outside its specified limits.

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**APPLICABLE SAFETY ANALYSES** The AFD is a measure of the axial power distribution skewing to either the top or bottom half of the core. The AFD is sensitive to many core related parameters such as control bank positions, core power level, axial burnup, axial xenon distribution, and, to a lesser extent, reactor coolant temperature and boron concentration.

The allowed range of the AFD is used in the nuclear design process to confirm that operation within these limits produces core peaking factors and axial power distributions that meet safety analysis requirements (Ref.1).

The limits on the AFD ensure that the Heat Flux Hot Channel Factor ( $F_Q(X,Y,Z)$ ) is not exceeded during either normal operation or in the event of xenon redistribution following power changes. The limits on the AFD also restrict the range of power distributions that are used as initial conditions in the analyses of Condition 2, 3, or 4 events. A Condition 4 event significantly affected by the initial axial power distribution, as indicated by AFD, is the LOCA. A Condition 3 event significantly affected by AFD is the Complete Loss of RCS Flow event.

BASES

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APPLICABLE SAFETY ANALYSES (continued)

A Condition 2 event significantly affected by AFD is the Uncontrolled RCCA Bank Withdrawal at Power Event (Ref. 2). Condition 2 accidents, simulated to begin from within the AFD limits are used to confirm the adequacy of the Overpower  $\Delta T$  and Overtemperature  $\Delta T$  trip setpoints.

The limits on the AFD satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

---

LCO

The shape of the power profile in the axial (i.e., the vertical) direction is largely under the control of the operator through the manual operation of the control banks or automatic motion of control banks. The automatic motion of the control banks is in response to temperature deviations resulting from manual operation of the Chemical and Volume Control System to change boron concentration or from power level changes.

Signals are available to the operator from the Nuclear Instrumentation System (NIS) excore neutron detectors (Refs. 1 and 3). Separate signals are taken from the top and bottom detectors. The AFD is defined as the difference in normalized flux signals between the top and bottom excore detectors in each detector well. For convenience, this flux difference is converted to provide flux difference units expressed as a percentage and labeled as  $\% \Delta$  flux or  $\% \Delta I$ .

The AFD limits are provided in the COLR. The AFD limits resulting from analysis of core power distributions relative to the initial condition peaking limits comprise a power-dependent envelope of acceptable AFD values. During steady-state operation, the core normally is controlled to a target AFD within a narrow (approximately  $\pm 5\%$  AFD) band. However, the limiting AFD values may be somewhat greater than the extremes of the normal operating band.

Violating this LCO on the AFD could produce unacceptable consequences if a Condition 2, 3, or 4 event occurs while the AFD is outside its specified limits.

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APPLICABILITY

The AFD requirements are applicable in MODE 1 greater than or equal to 50% RTP when the combination of THERMAL POWER and core peaking factors are of primary importance in safety analysis.

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ACTIONS

A.1

As an alternative to restoring the AFD to within its specified limits, Required Action A.1 requires a THERMAL POWER reduction to  $< 50\%$  RTP. This places the core in a condition for which the value of the AFD is not important in the applicable safety analyses. A Completion

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BASES

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ACTIONS (continued)

Time of 30 minutes is reasonable, based on operating experience, to reach 50% RTP without challenging plant systems.

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SURVEILLANCE  
REQUIREMENTS

SR 3.2.3.1

This Surveillance verifies that the AFD, as indicated by the NIS excore channel, is within its specified limits.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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REFERENCES

1. BAW-10163P-A, Revision 0, "Core Operating Limit Methodology for Westinghouse-Designed PWRs," June 1989.
  2. UFSAR, Chapter 15.
  3. UFSAR, Section 4.3.2.
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## B 3.2 POWER DISTRIBUTION LIMITS

### B 3.2.4 QUADRANT POWER TILT RATIO (QPTR)

#### BASES

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**BACKGROUND** The QPTR limit ensures that the gross radial power distribution remains consistent with the design values used in the safety analyses. Precise radial power distribution measurements are made during startup testing, after refueling, and periodically during power operation.

The power density at any point in the core must be limited so that the fuel design criteria are maintained. Together, LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.2.4, and LCO 3.1.6, "Control Bank Insertion Limits," provide limits on process variables that characterize and control the three dimensional power distribution of the reactor core. Control of these variables ensures that the core operates within the fuel design criteria and that the power distribution remains within the bounds used in the safety analyses.

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**APPLICABLE SAFETY ANALYSES** This LCO precludes core power distributions that violate the following fuel design criteria:

- a. During a large break loss of coolant accident, the peak cladding temperature must not exceed 2200°F (Ref. 1),
- b. During a loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 departure from nucleate boiling (DNB) criterion) that the hot fuel rod in the core does not experience a DNB condition,
- c. During an ejected rod accident, the energy deposition to the fuel must not exceed 280 cal/gm (Ref. 2), and
- d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 3).

The LCO limits on the AFD, the QPTR, the Heat Flux Hot Channel Factor ( $F_Q(X,Y,Z)$ ), the Nuclear Enthalpy Rise Hot Channel Factor ( $F_{\Delta H}(X,Y)$ ) and control bank insertion are established to preclude core power distributions that exceed the safety analyses limits.

The QPTR limits ensure that  $F_{\Delta H}(X,Y)$  and  $F_Q(X,Y,Z)$  remain below their limiting values by preventing an undetected change in the gross radial power distribution.

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BASES

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APPLICABLE SAFETY ANALYSES (continued)

In MODE 1, the  $F_{\Delta H}(X,Y)$  and  $F_Q(X,Y,Z)$  limits must be maintained to preclude core power distributions from exceeding design limits assumed in the safety analyses.

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

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LCO

The QPTR limit of 1.02, at which corrective action is required, provides a margin of protection for both the DNB ratio and linear heat generation rate contributing to excessive power peaks resulting from X-Y plane power tilts. A limiting QPTR of 1.02 can be tolerated before the margin for uncertainty in  $F_Q(X,Y,Z)$  and  $F_{\Delta H}(X,Y)$  is possibly challenged.

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APPLICABILITY

The QPTR limit must be maintained in MODE 1 with THERMAL POWER > 50% RTP to prevent core power distributions from exceeding the design limits.

Applicability in MODE 1  $\leq$  50% RTP and in other MODES is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the reactor coolant to require the implementation of a QPTR limit on the distribution of core power. The QPTR limit in these conditions is, therefore, not important. Note that the  $F_{\Delta H}(X,Y)$  and  $F_Q(X,Y,Z)$  LCOs still apply, but allow progressively higher peaking factors at 50% RTP or lower.

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ACTIONS

A.1

With the QPTR exceeding its limit, a power level reduction of 3% RTP for each 1% by which the QPTR exceeds 1.02 is a conservative tradeoff of total core power with peak linear power. The Completion Time of 2 hours allows sufficient time to identify the cause and correct the tilt. Note that the power reduction itself may cause a change in the tilted condition.

The maximum allowable power level initially determined by Required Action A.1 may be affected by subsequent determinations of QPTR. Increases in QPTR would require power reduction within 2 hours of QPTR determination, if necessary to comply with the decreased maximum allowable power level. Decreases in QPTR would allow increasing the maximum allowable power level and increasing power up to this revised limit.

BASES

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ACTIONS (continued)

A.2

After completion of Required Action A.1, the QPTR alarm may still be in its alarmed state. As such, any additional changes in the QPTR are detected by requiring a check of the QPTR once per 12 hours thereafter. A 12 hour Completion Time is sufficient because any additional change in QPTR would be relatively slow.

A.3

The peaking factors  $F_Q(X,Y,Z)$  and  $F_{\Delta H}(X,Y)$  are of primary importance in ensuring that the power distribution remains consistent with the initial conditions used in the safety analyses. Performing SRs on  $F_{\Delta H}(X,Y)$  and  $F_Q(X,Y,Z)$  within the Completion Time of 24 hours after achieving equilibrium conditions from a Thermal Power reduction per Required Action A.1 ensures that these primary indicators of power distribution are within their respective limits. Equilibrium conditions are achieved when the core is sufficiently stable at intended operating conditions to support flux mapping. A Completion Time of 24 hours after achieving equilibrium conditions from Thermal Power reduction per Required Action A.1 takes into consideration the rate at which peaking factors are likely to change, and the time required to stabilize the plant and perform a flux map. If these peaking factors are not within their limits, the Required Actions of the applicable LCOs of these Surveillances provide an appropriate response for the abnormal condition. If the QPTR remains above its specified limit, the peaking factor surveillances are required each 7 days thereafter to evaluate  $F_{\Delta H}(X,Y)$  and  $F_Q(X,Y,Z)$  with changes in power distribution. Relatively small changes are expected due to either burnup and xenon redistribution or correction of the cause for exceeding the QPTR limit.

A.4

Although  $F_{\Delta H}(X,Y)$  and  $F_Q(X,Y,Z)$  are of primary importance as initial conditions in the safety analyses, other changes in the power distribution may occur as the QPTR limit is exceeded and may have an impact on the validity of the safety analysis. A change in the power distribution can affect such reactor parameters as bank worths and peaking factors for rod malfunction accidents. When the QPTR exceeds its limit, it does not

BASES

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## ACTIONS (continued)

necessarily mean a safety concern exists. It does mean that there is an indication of a change in the gross radial power distribution that requires an investigation and evaluation that is accomplished by examining the incore power distribution. Specifically, the core peaking factors and the quadrant tilt must be evaluated because they are the factors that best characterize the core power distribution. This re-evaluation is required to ensure that, before increasing THERMAL POWER to above the limit of Required Action A.1, the reactor core conditions are consistent with the assumptions in the safety analyses.

A.5

If the QPTR is still exceeding the 1.02 limit and a re-evaluation of the safety analysis is completed and shows that safety requirements are met, the excore detectors shall be normalized to restore QPTR to within limits prior to increasing THERMAL POWER to above the limit of Required Action A.1. Normalization is accomplished in such a manner that the indicated QPTR following normalization is near 1.02. This is done to detect any subsequent significant changes in QPTR.

Required Action A.5 is modified by two Notes. Note 1 states that the QPTR shall not be restored to within limits by excore detector normalization until after the re-evaluation of the safety analysis has determined that core conditions at RTP are within the safety analysis assumptions (i.e., Required Action A.4). Note 2 states that if Required Action A.5 is performed, then Required Action A.6 shall be performed. Required Action A.5 normalizes the excore detectors to restore QPTR to within limits, which restores compliance with LCO 3.2.4. Thus, Note 2 prevents exiting the Actions prior to completing flux mapping to verify peaking factors, per Required Action A.6. These Notes are intended to prevent any ambiguity about the required sequence of actions.

A.6

Once the flux tilt is restored to within limits (i.e., Required Action A.5 is performed), it is acceptable to return to full power operation. However, as an added check that the core power distribution is consistent with the safety analysis assumptions, Required Action A.6 requires verification

BASES

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ACTIONS (continued)

that  $F_Q(X,Y,Z)$  and  $F_{\Delta H}(X,Y)$  are within their specified limits within 24 hours of achieving equilibrium conditions at RTP. As an added precaution, if the core power does not reach equilibrium conditions at RTP within 24 hours, but is increased slowly, then the peaking factor surveillances must be performed within 48 hours after increasing THERMAL POWER above the limit of Required Action A.1. These Completion Times are intended to allow adequate time to increase THERMAL POWER to above the limit of Required Action A.1, while not permitting the core to remain with unconfirmed power distributions for extended periods of time.

Required Action A.6 is modified by a Note that states that the peaking factor surveillances may only be done after the excore detectors have been normalized to restore QPTR to within limits (i.e., Required Action A.5). The intent of this Note is to have the peaking factor surveillances performed at operating power levels, which can only be accomplished after the excore detectors are normalized to restore QPTR to within limits and the core returned to power.

B.1

If Required Actions A.1 through A.6 are not completed within their associated Completion Times, the unit must be brought to a MODE or condition in which the requirements do not apply. To achieve this status, THERMAL POWER must be reduced to  $\leq 50\%$  RTP within 4 hours. The allowed Completion Time of 4 hours is reasonable, based on operating experience regarding the amount of time required to reach the reduced power level without challenging plant systems.

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SURVEILLANCE  
REQUIREMENTS

SR 3.2.4.1

SR 3.2.4.1 is modified by two Notes. Note 1 allows QPTR to be calculated with three power range channels if THERMAL POWER is  $\leq 75\%$  RTP and the input from one Power Range Neutron Flux channel is inoperable. Note 2 allows performance of SR 3.2.4.2 in lieu of SR 3.2.4.1.

This Surveillance verifies that the QPTR, as indicated by the Nuclear Instrumentation System (NIS) excore channels, is within its limits.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

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SURVEILLANCE REQUIREMENTS (continued)

For those causes of QPTR that occur quickly (e.g., a dropped rod), there typically are other indications of abnormality that prompt a verification of core power tilt.

SR 3.2.4.2

This Surveillance is modified by a Note, which states that the surveillance is only required to be performed if input to QPTR from one or more Power Range Neutron Flux channels are inoperable with THERMAL POWER > 75% RTP.

With an NIS power range channel inoperable, tilt monitoring for a portion of the reactor core becomes degraded. Large tilts are likely detected with the remaining channels, but the capability for detection of small power tilts in some quadrants is decreased.

With input to QPTR from one or more Power Range Neutron Flux channels inoperable and with THERMAL POWER >75% RTP, the surveillance is initially performed within 12 hours. Thereafter, the Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

For purposes of monitoring the QPTR when one power range channel is inoperable, the moveable incore detectors are used to confirm that the normalized symmetric power distribution is consistent with the indicated QPTR and any previous data indicating a tilt. The incore detector monitoring is performed with a full incore flux map or two sets of four thimble locations with quarter core symmetry. The two sets of four symmetric thimbles is a set of eight unique detector locations. These locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11, and N-8.

The symmetric thimble flux map can be used to generate symmetric thimble "tilt." This can be compared to a reference symmetric thimble tilt, from the most recent full core flux map, to generate an incore QPTR. Therefore, incore monitoring of QPTR can be used to confirm that QPTR is within limits.

With one NIS channel inoperable, the indicated tilt may be changed from the value indicated with all four channels OPERABLE. To confirm that no change in tilt has actually occurred, which might cause the QPTR limit to be exceeded, the incore result may be compared against previous flux maps either using the symmetric thimbles as described above or a complete flux map. Nominally, quadrant tilt from the Surveillance should be within 2% of the tilt shown by the most recent flux map data.

BASES

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- REFERENCES
1. 10 CFR 50.46.
  2. Regulatory Guide 1.77, Rev. 0, May 1974.
  3. 10 CFR 50, Appendix A, GDC 26.
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## B 3.3 INSTRUMENTATION

### B 3.3.1 Reactor Trip System (RTS) Instrumentation

#### BASES

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**BACKGROUND** The RTS initiates a unit shutdown, based on the values of selected unit parameters, to protect against violating the core fuel design limits and Reactor Coolant System (RCS) pressure boundary during Anticipated Operational Occurrences (AOOs) and to assist the Engineered Safety Features (ESF) Systems in mitigating accidents.

The protection and monitoring systems have been designed to assure safe operation of the reactor. This is achieved by specifying Limiting Safety System Settings (LSSS) in terms of parameters directly monitored by the RTS, as well as specifying LCOs on other reactor system parameters and equipment performance.

Technical Specifications are required by 10 CFR 50.36 to include LSSS. LSSS are defined by the regulation as settings for automatic protective devices related to those variables having significant safety functions. The regulation also states, "Where a LSSS is specified for a variable on which a safety limit has been placed, the setting must be chosen so that automatic protective actions will correct the abnormal situation before a Safety Limit (SL) is exceeded." The Analytical Limit is the limit of the process variable at which a protective action is initiated, as established by the safety analysis, to ensure that a SL is not exceeded. Any automatic protection action that occurs on reaching the Analytical Limit therefore ensures that the SL is not exceeded. However, in practice, the actual settings for automatic protection channels must be chosen to be more conservative than the Analytical Limit to account for instrument loop uncertainties related to the setting at which the automatic protective action would actually occur.

The Nominal Trip Setpoint (NTSP) specified in Table 3.3.1-1 is a predetermined setting for a protection channel chosen to ensure automatic actuation prior to the process variable reaching the Analytical Limit and thus ensuring that the SL would not be exceeded. As such, the NTSP accounts for uncertainties in setting the channel (e.g., calibration), uncertainties in how the channel might actually perform (e.g., repeatability), changes in the point of action of the channel over time (e.g., drift during surveillance intervals), and any other factors which may influence its actual performance (e.g., harsh accident environments). In this manner, the NTSP ensures that SLs are not exceeded. Therefore, the NTSP meets the definition of an LSSS (Ref. 1).

## BASES

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### BACKGROUND (continued)

Technical Specifications contain values related to the OPERABILITY of equipment required for safe operation of the facility. OPERABLE is defined in Technical Specifications as "...being capable of performing its safety functions(s)." Relying solely on the NTSP to define OPERABILITY in Technical Specifications would be an overly restrictive requirement if it were applied as an OPERABILITY limit for the "as-found" value of a protection channel setting during a surveillance. This would result in Technical Specification compliance problems, as well as reports and corrective actions required by the rule which are not necessary to ensure safety. For example, an automatic protection channel with a setting that has been found to be different from the NTSP due to some drift of the setting may still be OPERABLE since drift is to be expected. This expected drift would have been specifically accounted for in the setpoint methodology for calculating the NTSP and thus the automatic protective action would still have ensured that the SL would not be exceeded with the "as-found" setting of the protection channel.

Therefore, the channel would still be OPERABLE since it would have performed its safety function and the only corrective action required would be to reset the channel within the established as-left tolerance around the NTSP to account for further drift during the next surveillance interval.

During AOOs, which are those events expected to occur one or more times during the unit life, the acceptable limits are:

1. The Departure from Nucleate Boiling Ratio (DNBR) shall be maintained above the Safety Limit (SL) value to prevent Departure from Nucleate Boiling (DNB),
2. Fuel centerline melt shall not occur, and
3. The RCS pressure SL of 2735 psig shall not be exceeded.

Operation within the SLs of Specification 2.0, "Safety Limits (SLs)," also maintains the above values and assures that offsite dose will be within the 10 CFR 50 and 10 CFR 100 criteria during AOOs.

Accidents are events that are analyzed even though they are not expected to occur during the unit life. The acceptable limit during accidents is that offsite dose shall be maintained within an acceptable fraction of 10 CFR 100 limits. Different accident categories are allowed a different fraction of these limits, based on probability of occurrence. Meeting the acceptable dose limit for an accident category is considered having acceptable consequences for that event.

## BASES

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### BACKGROUND (continued)

The RTS instrumentation is segmented into four distinct but interconnected modules as illustrated in Figure 7.2.2-2, UFSAR, Chapter 7 (Ref. 2), and as identified below:

1. Field transmitters or process sensors: provide a measurable electronic signal based upon the physical characteristics of the parameter being measured,
2. Signal Process Control and Protection System, including Process Protection System, Nuclear Instrumentation System (NIS), field contacts, and protection channel sets: provides analog to digital conversion (Digital Protection System), signal conditioning, setpoint comparison, process algorithm actuation, compatible electrical signal output to protection system channels, and control board/control room/miscellaneous indications,
3. Solid State Protection System (SSPS), including input, logic, and output bays: initiates proper unit shutdown and/or ESF actuation in accordance with the defined logic, which is based on the bistable, setpoint comparator, or contact outputs from the signal process control and protection system, and
4. Reactor trip switchgear, including reactor trip breakers and bypass breakers: provides the means to interrupt power to the Control Rod Drive Mechanisms (CRDMs) and allows the Rod Cluster Control Assemblies (RCCAs), or "rods," to fall into the core and shut down the reactor. The bypass breakers allow testing of the reactor trip breakers at power.

#### Field Transmitters or Sensors

To meet the design demands for redundancy and reliability, more than one, and often as many as four, field transmitters or sensors are used to measure unit parameters. To account for the calibration tolerances and instrument drift, which are assumed to occur between calibrations, statistical allowances are provided in the NTSP and Allowable Value. The OPERABILITY of each transmitter or sensor is determined by either "as-found" calibration data evaluated during the CHANNEL CALIBRATION or by qualitative assessment of field transmitter or sensor as related to the channel behavior observed during performance of the CHANNEL CHECK.

## BASES

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### BACKGROUND (continued)

#### Signal Process Control and Protection System

Generally, three or four channels of process control equipment are used for the signal processing of unit parameters measured by the field instruments. The process control equipment provides analog to digital conversion (Digital Protection System), signal conditioning, comparable output signals for instruments located on the main control board, and comparison of measured input signals with NTSPs derived from Analytical Limits established by the safety analyses. Analytical Limits are defined in UFSAR, Chapter 7 (Ref. 2), Chapter 6 (Ref. 3), and Chapter 15 (Ref. 4). If the measured value of a unit parameter exceeds the predetermined setpoint, an output from a bistable, setpoint comparator, or contact is forwarded to the SSPS for decision evaluation. Channel separation is maintained up to and through the input bays. However, not all unit parameters require four channels of sensor measurement and signal processing. Some unit parameters provide input only to the SSPS, while others provide input to the SSPS, the main control board, the unit computer, and one or more control systems.

Generally, if a parameter is used only for input to the protection circuits, three channels with a two-out-of-three logic are sufficient to provide the required reliability and redundancy. If one channel fails in a direction that would not result in a partial Function trip, the Function is still OPERABLE with a two-out-of-two logic. If one channel fails, such that a partial Function trip occurs, a trip will not occur and the Function is still OPERABLE with a one-out-of-two logic.

Generally, if a parameter is used for input to the SSPS and a control function, four channels with a two-out-of-four logic are sufficient to provide the required reliability and redundancy. The circuit must be able to withstand both an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation. Again, a single failure will neither cause nor prevent the protection function actuation. These requirements are described in IEEE-279-1971 (Ref. 5). The actual number of channels required for each unit parameter is specified in Reference 2.

Two logic channels are required to ensure no single random failure of a logic channel will disable the RTS. The logic channels are designed such that testing required while the reactor is at power may be accomplished without causing trip. Provisions to allow removing logic channels from service during maintenance are unnecessary because of the logic system's designed reliability.

BASES

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## BACKGROUND (continued)

Allowable Values and Nominal Trip Setpoints

The trip setpoints used in the bistables, setpoint comparators, or contacts are based on the Analytical Limits stated in Reference 4. The calculation of the NTSP specified in Table 3.3.1-1 is such that adequate protection is provided when all sensor and processing time delays are taken into account. To allow for calibration tolerances, instrumentation uncertainties, instrument drift, and severe environment errors for those RTS channels that must function in harsh environments as defined by 10 CFR 50.49 (Ref. 6), the Allowable Values specified in Table 3.3.1-1 in the accompanying LCO are conservative with respect to the Analytical Limits. A detailed description of the methodology used to calculate the Allowable Values and NTSP, including their explicit uncertainties, is provided in the plant specific setpoint methodology study (Ref. 7) which incorporates all of the known uncertainties applicable to each channel. The as-left tolerance and as-found tolerance band methodology is provided in UFSAR, Section 7.1.2. The magnitudes of these uncertainties are factored into the determination of each NTSP and corresponding Allowable Value. The trip setpoint entered into the bistable or setpoint comparator is more conservative than that specified by the Allowable Value to account for measurement errors detectable by the CHANNEL OPERATIONAL TEST (COT). The Allowable Value serves as the as-found Technical Specification OPERABILITY limit for the purpose of the COT.

The NTSP is the value at which the bistable or setpoint comparator is set and is the expected value to be achieved during calibration. The NTSP value is the LSSS and ensures the safety analysis limits are met for the surveillance interval selected when a channel is adjusted based on stated channel uncertainties. Any bistable or setpoint comparator is considered to be properly adjusted when the "as-left" NTSP value is within the as-left tolerance band for CHANNEL CALIBRATION uncertainty allowance (i.e.,  $\pm$  rack calibration and comparator setting uncertainties). The NTSP value is therefore considered a "nominal" value (i.e., expressed as a value without inequalities) for the purposes of COT and CHANNEL CALIBRATION.

NTSPs, in conjunction with the use of as-found and as-left tolerances, together with the requirements of the Allowable Value ensure that SLs are not violated during AOOs (and that the consequences of DBAs will be acceptable, providing the unit is operated from within the LCOs at the onset of the AOO or DBA and the equipment functions as designed).

## BASES

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### BACKGROUND (continued)

Note that the Allowable Values listed in Table 3.3.1-1 are the least conservative value of the as-found setpoint that a channel can have during a periodic CHANNEL CALIBRATION, COTs, or a TRIP ACTUATING DEVICE OPERATIONAL TEST that requires trip setpoint verification.

Each channel of the process control equipment can be tested online to verify that the signal or setpoint accuracy is within the specified allowance requirements of Reference 3. Once a designated channel is taken out of service for testing, a simulated signal is injected in place of the field instrument signal. The process equipment for the channel in test is then tested, verified, and calibrated. SRs for the channels are specified in the SRs section.

#### Solid State Protection System

The SSPS equipment is used for the decision logic processing of outputs from the signal processing equipment bistables, setpoint comparators, or contacts. To meet the redundancy requirements, two trains of SSPS, each performing the same functions, are provided. If one train is taken out of service for maintenance or test purposes, the second train will provide reactor trip and/or ESF actuation for the unit. If both trains are taken out of service or placed in test, a reactor trip will result. Each train is packaged in its own cabinet for physical and electrical separation to satisfy separation and independence requirements. The system has been designed to trip in the event of a loss of power, directing the unit to a safe shutdown condition.

The SSPS performs the decision logic for actuating a reactor trip or ESF actuation, generates the electrical output signal that will initiate the required trip or actuation, and provides the status, permissive, and annunciator output signals to the main control room of the unit.

The bistable, setpoint comparator, or contact outputs from the signal processing equipment are sensed by the SSPS equipment and combined into logic matrices that represent combinations indicative of various unit upset and accident transients. If a required logic matrix combination is completed, the system will initiate a reactor trip or send actuation signals via master and slave relays to those components whose aggregate Function best serves to alleviate the condition and restore the unit to a safe condition. Examples are given in the Applicable Safety Analyses, LCO, and Applicability sections of this Bases.

## BASES

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### BACKGROUND (continued)

#### Reactor Trip Switchgear

The reactor trip breakers are in the electrical power supply line from the control rod drive motor generator set power supply to the CRDMs. Opening of the reactor trip breakers interrupts power to the CRDMs, which allows the shutdown rods and control rods to fall into the core by gravity. Each reactor trip breaker is equipped with a bypass breaker to allow testing of the reactor trip breaker while the unit is at power.

During normal operation the output from the SSPS is a voltage signal that energizes the undervoltage coils in the reactor trip breakers and bypass breakers, if in use. When the required logic matrix combination is completed, the SSPS output voltage signal is removed, the undervoltage coils are de-energized, the breaker trip lever is actuated by the de-energized undervoltage coil, and the reactor trip breakers and bypass breakers are tripped open. This allows the shutdown rods and control rods to fall into the core. In addition to the de-energization of the undervoltage coils, each breaker is also equipped with a shunt trip device that is energized to trip the breaker open upon receipt of a reactor trip signal from the SSPS. Either the undervoltage coil or the shunt trip mechanism is sufficient by itself, thus providing a diverse trip mechanism.

The decision logic matrix Functions are described in the functional diagrams included in Reference 2. In addition to the reactor trip or ESF, these diagrams also describe the various "permissive interlocks" that are associated with unit conditions. Each train has a built in testing device that can automatically test the decision logic matrix Functions and the actuation channels while the unit is at power. When any one train is taken out of service for testing, the other train is capable of providing unit monitoring and protection until the testing has been completed. The testing device is semiautomatic to minimize testing time.

## BASES

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APPLICABLE  
SAFETY  
ANALYSES, LCO,  
and APPLICABILITY

The RTS functions to preserve the SLs during all AOOs and mitigates the consequences of DBAs in all MODES in which the Rod Control System is capable of rod withdrawal or one or more rods are not fully inserted.

Each of the analyzed accidents and transients can be detected by one or more RTS Functions. The accident analysis described in Reference 4 takes credit for most RTS trip Functions. RTS trip Functions that are retained yet not specifically credited in the accident analysis are implicitly credited in the safety analysis and the NRC staff approved licensing basis for the unit. These RTS trip Functions may provide protection for conditions that do not require dynamic transient analysis to demonstrate Function performance. They may also serve as backups to RTS trip Functions that were credited in the accident analysis.

Permissive and interlock setpoints allow the blocking of trips during plant startups, and restoration of trips when the permissive conditions are not satisfied, but they are not explicitly modeled in the Safety Analyses. These permissives and interlocks ensure that the starting conditions are consistent with the safety analysis, before preventive or mitigating actions occur. Because these permissives or interlocks are only one of multiple conservative starting assumptions for the accident analysis, they are generally considered as nominal values without regard to measurement accuracy.

The LCO requires all instrumentation performing an RTS Function, listed in Table 3.3.1-1 to be OPERABLE. The Allowable Value specified in Table 3.3.1-1 is the least conservative value of the as-found setpoint that the channel can have when tested, such that a channel is OPERABLE if the as-found setpoint is within the as-found tolerance and is conservative with respect to the Allowable Value during a CHANNEL CALIBRATION or COT. As such, the Allowable Value differs from the NTSP by an amount greater than or equal to the expected instrument channel uncertainties, such as drift, during the surveillance interval. In this manner, the actual setting of the channel NTSP will ensure that a SL is not exceeded at any given point of time as long as the channel has not drifted beyond expected tolerances during the surveillance interval. Note that, although the channel is OPERABLE under these circumstances, the trip setpoint must be left adjusted to a value within the as-left tolerance, in accordance with uncertainty assumptions stated in the referenced setpoint methodology (as-left criteria), and confirmed to be operating within the statistical allowances of the uncertainty terms assigned (as-found criteria).

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### APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

If the actual setting of the channel is found to be conservative with respect to the Allowable Value but is beyond the as-found tolerance band, the channel is OPERABLE but degraded. The degraded condition of the channel will be further evaluated during performance of the SR. This evaluation will consist of resetting the channel setpoint to the NTSP (within the allowed tolerance), and evaluating the channel's response. If the channel is functioning as required and is expected to pass the next surveillance, then the channel is OPERABLE and can be restored to service at the completion of the surveillance. After the surveillance is completed, the channel as-found condition will be entered into the Corrective Action Program for further evaluation.

A trip setpoint may be set more conservative than the NTSP as necessary in response to plant conditions. However, in this case, the OPERABILITY of this instrument must be verified based on the field setting and not the NTSP. Failure of any instrument renders the affected channel(s) inoperable and reduces the reliability of the affected Functions.

The LCO generally requires OPERABILITY of four or three channels in each instrumentation Function, two channels of Manual Reactor Trip in each logic Function, and two trains in each Automatic Trip Logic Function. Four OPERABLE instrumentation channels in a two-out-of-four configuration are required when one RTS channel is also used as a control system input. This configuration accounts for the possibility of the shared channel failing in such a manner that it creates a transient that requires RTS action. In this case, the RTS will still provide protection, even with random failure of one of the other three protection channels. Three OPERABLE instrumentation channels in a two-out-of-three configuration are generally required when there is no potential for control system and protection system interaction that could simultaneously create a need for RTS trip and disable one RTS channel. The two-out-of-three and two-out-of-four configurations allow one channel to be tripped during maintenance or testing without causing a reactor trip. Specific exceptions to the above general philosophy exist and are discussed below.

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### APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

#### Reactor Trip System Functions

The safety analyses and OPERABILITY requirements applicable to each RTS Function are discussed below:

##### 1. Manual Reactor Trip

The Manual Reactor Trip ensures that the control room operator can initiate a reactor trip at any time by using either of two reactor trip switches in the control room. A Manual Reactor Trip accomplishes the same results as any one of the automatic trip Functions. It is used by the reactor operator to shut down the reactor whenever any parameter is rapidly trending toward its Trip Setpoint.

There are two Manual Reactor Trip channels arranged in a one-out-of-two logic. The LCO requires two Manual Reactor Trip channels to be OPERABLE. Each channel is controlled by a manual reactor trip switch. Each channel activates the reactor trip breaker in both trains. Two independent channels are required to be OPERABLE so that no single random failure will disable the Manual Reactor Trip Function.

In MODE 1 or 2, manual initiation of a reactor trip must be OPERABLE. These are the MODES in which the reactor is critical. In MODE 3, 4, or 5, the manual initiation Function must also be OPERABLE if one or more shutdown rods or control rods are withdrawn or the Rod Control System is capable of withdrawing the shutdown rods or the control rods. In this condition, inadvertent control rod withdrawal is possible. In MODE 3, 4, or 5, manual initiation of a reactor trip does not have to be OPERABLE if the Rod Control System is not capable of withdrawing the shutdown rods or control rods and if all rods are fully inserted. If the rods cannot be withdrawn from the core, or all of the rods are inserted, there is no need to be able to trip the reactor. In MODE 6, neither the shutdown rods nor the control rods are permitted to be withdrawn and the CRDMs are disconnected from the control rods and shutdown rods. Therefore, the manual initiation Function is not required.

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### APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

#### 2. Power Range Neutron Flux

The NIS power range detectors are located external to the reactor vessel and measure neutrons leaking from the core. The NIS power range detectors provide input to the Rod Control System. Therefore, the actuation logic must be able to withstand an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation. Note that this Function also provides a signal to prevent automatic and manual rod withdrawal prior to initiating a reactor trip. Limiting further rod withdrawal may terminate the transient and eliminate the need to trip the reactor.

##### a. Power Range Neutron Flux - High

The Power Range Neutron Flux - High trip Function ensures that protection is provided, from all power levels, against a positive reactivity excursion leading to DNB during power operations. These can be caused by rod withdrawal or reductions in RCS temperature.

There are four Power Range Neutron Flux – High channels arranged in a two-out-of-four logic. The LCO requires all four of the Power Range Neutron Flux - High channels to be OPERABLE.

In MODE 1 or 2, when a positive reactivity excursion could occur, the Power Range Neutron Flux - High trip must be OPERABLE. This Function will terminate the reactivity excursion and shut down the reactor prior to reaching a power level that could damage the fuel. In MODE 3, 4, 5, or 6, the NIS power range detectors cannot detect neutron levels in this range. In these MODES, the Power Range Neutron Flux - High does not have to be OPERABLE because the reactor is shut down and reactivity excursions into the power range are extremely unlikely. Other RTS Functions and administrative controls provide protection against reactivity additions when in MODE 3, 4, 5, or 6.

##### b. Power Range Neutron Flux - Low

The LCO requirement for the Power Range Neutron Flux - Low trip Function ensures that protection is provided against a positive reactivity excursion from low power or subcritical conditions.

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### APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

There are four Power Range Neutron Flux – Low channels arranged in a two-out-of-four logic. The LCO requires all four of the Power Range Neutron Flux - Low channels to be OPERABLE.

In MODE 1, below the Power Range Neutron Flux (P-10 setpoint), and in MODE 2, the Power Range Neutron Flux - Low trip must be OPERABLE. This Function may be manually blocked by the operator when two out of four power range channels are greater than approximately 10% RTP (P-10 setpoint). This Function is automatically unblocked when three out of four power range channels are below the P-10 setpoint. Above the P-10 setpoint, positive reactivity additions are mitigated by the Power Range Neutron Flux - High trip Function.

In MODE 3, 4, 5, or 6, the Power Range Neutron Flux - Low trip Function does not have to be OPERABLE because the reactor is shut down and the NIS power range detectors cannot detect neutron levels in this range. Other RTS trip Functions and administrative controls provide protection against positive reactivity additions or power excursions in MODE 3, 4, 5, or 6.

#### 3. Power Range Neutron Flux Rate

The Power Range Neutron Flux Rate trips use the same channels as discussed for Function 2 above.

##### a. Power Range Neutron Flux - High Positive Rate

The Power Range Neutron Flux - High Positive Rate trip Function ensures that protection is provided against rapid increases in neutron flux that are characteristic of an RCCA drive rod housing rupture and the accompanying ejection of the RCCA. This Function compliments the Power Range Neutron Flux - High and Low Setpoint trip Functions to ensure that the criteria are met for a rod ejection from the power range.

There are four Power Range Neutron Flux – High Positive Rate channels arranged in a two-out-of-four logic. The LCO requires all four of the Power Range Neutron Flux - High Positive Rate channels to be OPERABLE.

In MODE 1 or 2, when there is a potential to add a large amount of positive reactivity from a Rod Ejection Accident (REA), the Power Range Neutron Flux - High Positive Rate trip must be OPERABLE. In MODE 3, 4, 5, with Rod Control System capable

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### APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

of rod withdrawal or one or more rods not fully inserted, or in MODE 6, the Power Range Neutron Flux - High Positive Rate trip Function does not have to be OPERABLE because other RTS trip Functions and administrative controls will provide protection against positive reactivity additions. Also, in MODE 3, 4, or 5, with Rod Control System incapable of rod withdrawal and all rods fully inserted, there is a sufficient degree of SDM in the event of an REA. In MODE 6, no rods are withdrawn and the SDM is increased during refueling operations. The reactor vessel head is also removed or the closure bolts are detensioned preventing any pressure buildup. In addition, the NIS power range detectors cannot detect neutron levels present in this mode.

b. Power Range Neutron Flux - High Negative Rate

The Power Range Neutron Flux - High Negative Rate trip Function ensures that protection is provided for multiple rod drop accidents. At high power levels, a multiple rod drop accident could cause local flux peaking that would result in a nonconservative local DNBR. DNBR is defined as the ratio of the heat flux required to cause a DNB at a particular location in the core to the local heat flux. The DNBR is indicative of the margin to DNB. No credit is taken for the operation of this Function for those rod drop accidents in which the local DNBRs will be greater than the limit.

There are four Power Range Neutron Flux – High Negative Rate channels arranged in a two-out-of-four logic. The LCO requires all four Power Range Neutron Flux - High Negative Rate channels to be OPERABLE.

In MODE 1 or 2, when there is potential for a multiple rod drop accident to occur, the Power Range Neutron Flux - High Negative Rate trip must be OPERABLE. In MODE 3, 4, 5, or 6, the Power Range Neutron Flux - High Negative Rate trip Function does not have to be OPERABLE because the core is not critical and DNB is not a concern. In MODE 6, no rods are withdrawn and the required SDM is increased during refueling operations. In addition, the NIS power range detectors cannot detect neutron levels present in this MODE.

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### APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

#### 4. Intermediate Range Neutron Flux

The Intermediate Range Neutron Flux trip Function ensures that protection is provided against an uncontrolled RCCA bank rod withdrawal accident from a subcritical condition during startup. This trip Function provides redundant protection to the Power Range Neutron Flux - Low Setpoint trip Function. The NIS intermediate range detectors are located external to the reactor vessel and measure neutrons leaking from the core. The NIS intermediate range detectors do not provide any input to control systems. Note that this Function also provides a signal to prevent automatic and manual rod withdrawal prior to initiating a reactor trip. Limiting further rod withdrawal may terminate the transient and eliminate the need to trip the reactor.

There are two Intermediate Range Neutron Flux channels arranged in a one-out-of-two logic. The LCO requires two channels of Intermediate Range Neutron Flux to be OPERABLE. Two OPERABLE channels are sufficient to ensure no single random failure will disable this trip Function.

Because this trip Function is important only during startup, there is generally no need to disable channels for testing while the Function is required to be OPERABLE. Therefore, a third channel is unnecessary.

In MODE 1 below the P-10 setpoint, and in MODE 2 above the P-6 setpoint, when there is a potential for an uncontrolled RCCA bank rod withdrawal accident during reactor startup, the Intermediate Range Neutron Flux trip must be OPERABLE. Above the P-10 setpoint, the Power Range Neutron Flux - High Setpoint trip and the Power Range Neutron Flux - High Positive Rate trip provide core protection for a rod withdrawal accident. In MODE 2 below the P-6 setpoint, the Source Range Neutron Flux Trip provides the core protection for reactivity accidents. In MODE 3, 4, or 5, the Intermediate Range Neutron Flux trip does not have to be OPERABLE because the Rod Control System is not capable of rod withdrawal or the Source Range Neutron Flux function is required to be OPERABLE, providing protection. In MODE 6, all rods are fully inserted and the core has a required increased SDM. Also, the NIS intermediate range detectors cannot detect neutron levels present in this MODE.

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

5. Source Range Neutron Flux

The LCO requirement for the Source Range Neutron Flux trip Function ensures that protection is provided against an uncontrolled RCCA bank rod withdrawal accident from a subcritical condition during startup. This trip Function provides redundant protection to the Power Range Neutron Flux - Low trip Function. In MODES 3, 4, and 5, administrative controls also prevent the uncontrolled withdrawal of rods. The NIS source range detectors are located external to the reactor vessel and measure neutrons leaking from the core. The NIS source range detectors do not provide any inputs to control systems. The source range trip is the only RTS automatic protection function required in MODES 3, 4, and 5 when rods are capable of withdrawal or one or more rods are not fully inserted. Therefore, the functional capability at the specified Trip Setpoint is assumed to be available.

The Source Range Neutron Flux Function provides protection for control rod withdrawal from subcritical and boron dilution events.

There are two Source Range Neutron Flux channels arranged in a one-out-of-two logic. In MODE 2 when below the P-6 setpoint and in MODES 3, 4, and 5 when there is a potential for an uncontrolled RCCA bank rod withdrawal accident, the Source Range Neutron Flux trip must be OPERABLE. Two OPERABLE channels are sufficient to ensure no single random failure will disable this trip Function. Above the P-6 setpoint, the Intermediate Range Neutron Flux trip and the Power Range Neutron Flux - Low trip will provide core protection for reactivity accidents. Above the P-6 setpoint, the NIS source range drawer input is shorted out, driving the output of the drawer to zero.

In MODES 3, 4, and 5 with all rods fully inserted and the Rod Control System not capable of rod withdrawal, and in MODE 6, the outputs of the Function to RTS logic are not required OPERABLE. The requirements for the NIS source range detectors to monitor core neutron levels and provide indication of reactivity changes that may occur as a result of events like a boron dilution are addressed in LCO 3.3.9 "Boron Dilution Monitoring Instrumentation (BDMI)," for MODE 3, 4, or 5 and LCO 3.9.3, "Nuclear Instrumentation," for MODE 6.

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

6. Overtemperature  $\Delta T$

The Overtemperature  $\Delta T$  trip Function is provided to ensure that the design limit DNBR is met. This trip Function also limits the range over which the Overpower  $\Delta T$  trip Function must provide protection. The inputs to the Overtemperature  $\Delta T$  trip include pressurizer pressure, coolant temperature, axial power distribution, and reactor power as indicated by loop  $\Delta T$  assuming full reactor coolant flow. Protection from violating the DNBR limit is assured for those transients that are slow with respect to delays from the core to the measurement system. The Function monitors both variation in power and flow since a decrease in flow has the same effect on  $\Delta T$  as a power increase. The Overtemperature  $\Delta T$  trip Function uses each loop's  $\Delta T$  as a measure of reactor power and is compared with a setpoint that is automatically varied with the following parameters:

- reactor coolant average temperature - the Trip Setpoint is varied to correct for changes in coolant density and specific heat capacity with changes in coolant temperature,
- pressurizer pressure - the Trip Setpoint is varied to correct for changes in system pressure, and
- axial power distribution -  $f(\Delta I)$ , the Trip Setpoint is varied to account for imbalances in the axial power distribution as detected by the NIS upper and lower power range detectors. If axial peaks are greater than the design limit, as indicated by the difference between the upper and lower NIS power range detectors, the Trip Setpoint is reduced in accordance with Note 1 of Table 3.3.1-1.

Dynamic compensation is included for system piping delays from the core to the temperature measurement system.

The Overtemperature  $\Delta T$  trip Function is calculated for each loop as described in Note 1 of Table 3.3.1-1. Trip occurs if Overtemperature  $\Delta T$  is indicated in two loops. The pressure and temperature signals are used for other control functions. Therefore, the actuation logic must be able to withstand an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation. Note that this Function also provides a signal to generate a turbine runback prior to reaching the Trip Setpoint. A turbine runback will reduce turbine power and reactor power. A reduction in power will normally alleviate the Overtemperature  $\Delta T$  condition and may prevent a reactor trip.

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

There are four Overtemperature  $\Delta T$  channels arranged in a two-out-of-four logic. The LCO requires all four channels of the Overtemperature  $\Delta T$  trip Function to be OPERABLE. Note that the Overtemperature  $\Delta T$  Function receives input from channels shared with other RTS Functions. Failures that affect multiple Functions require entry into the Conditions applicable to all affected Functions.

In MODE 1 or 2, the Overtemperature  $\Delta T$  trip must be OPERABLE to prevent DNB. In MODE 3, 4, 5, or 6, this trip Function does not have to be OPERABLE because the reactor is not operating and there is insufficient heat production to be concerned about DNB.

7. Overpower  $\Delta T$

The Overpower  $\Delta T$  trip Function ensures that protection is provided to ensure the integrity of the fuel (i.e., no fuel pellet melting and less than 1% cladding strain) under all possible overpower conditions. This trip Function also limits the required range of the Overtemperature  $\Delta T$  trip Function and provides a backup to the Power Range Neutron Flux - High Setpoint trip. The Overpower  $\Delta T$  trip Function ensures that the allowable heat generation rate (kW/ft) of the fuel is not exceeded. It uses the  $\Delta T$  of each loop as a measure of reactor power with a setpoint that is automatically varied with the following parameters:

- reactor coolant average temperature - the Trip Setpoint is varied to correct for changes in coolant density and specific heat capacity with changes in coolant temperature,
- rate of change of reactor coolant average temperature - including dynamic compensation for the delays between the core and the temperature measurement system, and
- axial power distribution -  $f(\Delta I)$ , the Trip Setpoint is varied to account for imbalances in the axial power distribution as detected by the NIS upper and lower power range detectors. If axial peaks are greater than the design limit, as indicated by the difference between the upper and lower NIS power range detectors, the Trip Setpoint is reduced in accordance with Note 2 of Table 3.3.1-1.

The Overpower  $\Delta T$  trip Function is calculated for each loop as per Note 2 of Table 3.3.1-1. Trip occurs if Overpower  $\Delta T$  is indicated in two loops. The temperature signals are used for other control functions. Therefore, the actuation logic must be able to withstand an input failure to the control system, which may then require the

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### APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

protection function actuation and a single failure in the remaining channels providing the protection function actuation. Note that this Function also provides a signal to generate a turbine runback prior to reaching the Allowable Value. A turbine runback will reduce turbine power and reactor power. A reduction in power will normally alleviate the Overpower  $\Delta T$  condition and may prevent a reactor trip.

There are four Overpower  $\Delta T$  channels arranged in a two-out-of-four logic. The LCO requires four channels of the Overpower  $\Delta T$  trip Function to be OPERABLE. Note that the Overpower  $\Delta T$  trip Function receives input from channels shared with other RTS Functions. Failures that affect multiple Functions require entry into the Conditions applicable to all affected Functions.

In MODE 1 or 2, the Overpower  $\Delta T$  trip Function must be OPERABLE. These are the only times that enough heat is generated in the fuel to be concerned about the heat generation rates and overheating of the fuel. In MODE 3, 4, 5, or 6, this trip Function does not have to be OPERABLE because the reactor is not operating and there is insufficient heat production to be concerned about fuel overheating and fuel damage.

#### 8. Pressurizer Pressure

The same sensors provide input to the Pressurizer Pressure - High and - Low trips and the Overtemperature  $\Delta T$  trip. The Pressurizer Pressure channels are also used to provide input to the Pressurizer Pressure Control System. The actuation logic must be able to withstand an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation.

##### a. Pressurizer Pressure - Low

The Pressurizer Pressure - Low trip Function ensures that protection is provided against violating the DNBR limit due to low pressure.

There are four Pressurizer Pressure - Low channels arranged in a two-out-of-four logic. The LCO requires four channels of Pressurizer Pressure - Low to be OPERABLE.

In MODE 1, when DNB is a major concern, the Pressurizer Pressure - Low trip must be OPERABLE. This trip Function is automatically enabled on increasing power by the P-7 interlock (NIS power range P-10 or turbine impulse pressure greater than

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### APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

approximately 10% of full power equivalent (P-13)). On decreasing power, this trip Function is automatically blocked below P-7.

b. Pressurizer Pressure - High

The Pressurizer Pressure - High trip Function ensures that protection is provided against overpressurizing the RCS. This trip Function operates in conjunction with the pressurizer relief and safety valves to prevent RCS overpressure conditions.

There are four Pressurizer Pressure - High channels arranged in a two-out-of-four logic. The LCO requires four channels of the Pressurizer Pressure - High to be OPERABLE.

The Pressurizer Pressure - High LSSS is selected to be below the pressurizer safety valve actuation pressure and above the Power Operated Relief Valve (PORV) setting. This setting minimizes challenges to safety valves while avoiding unnecessary reactor trip for those pressure increases that can be controlled by the PORVs.

In MODE 1 or 2, the Pressurizer Pressure - High trip must be OPERABLE to help prevent RCS overpressurization and minimize challenges to the relief and safety valves. In MODE 3, 4, 5, or 6, the Pressurizer Pressure - High trip Function does not have to be OPERABLE because transients that could cause an overpressure condition will be slow to occur. Therefore, the operator will have sufficient time to evaluate unit conditions and take corrective actions. Additionally, low temperature overpressure protection systems provide overpressure protection when below MODE 4.

9. Pressurizer Water Level - High

The Pressurizer Water Level - High trip Function provides a backup signal for the Pressurizer Pressure - High trip and also provides protection against water relief through the pressurizer safety valves. These valves are designed to pass steam in order to achieve their design energy removal rate. A reactor trip is actuated prior to the pressurizer becoming water solid. The LCO requires three channels of Pressurizer Water Level - High to be OPERABLE. The pressurizer level channels are used as input to the Pressurizer Level Control System. A fourth channel is not required to address control/protection interaction concerns. The level channels do not

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### APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

actuate the safety valves, and the high pressure reactor trip is set below the safety valve setting. Therefore, with the slow rate of charging available, pressure overshoot due to level channel failure cannot cause the safety valve to lift before reactor high pressure trip.

There are three Pressurizer Level - High channels arranged in a two-out-of-three logic. In MODE 1, when there is a potential for overfilling the pressurizer, the Pressurizer Water Level - High trip must be OPERABLE. This trip Function is automatically enabled on increasing power by the P-7 interlock. On decreasing power, this trip Function is automatically blocked below P-7.

#### 10. Reactor Coolant Flow - Low

The Reactor Coolant Flow - Low trip Function ensures that protection is provided against violating the DNBR limit due to low flow in one or more RCS loops, while avoiding reactor trips due to normal variations in loop flow. Above the P-7 setpoint, the reactor trip on low flow in two or more RCS loops is automatically enabled. Above the P-8 setpoint, which is approximately 35% RTP, a loss of flow in any RCS loop will actuate a reactor trip. Each RCS loop has three flow detectors to monitor flow. There are three per loop Reactor Coolant Flow - Low channels using these detectors and are arranged in a two-out-of-three logic for each loop. The flow signals are not used for any control system input.

Design flow is 94,600 (91,400 X 1.035) gpm per loop (Reference 14). UFSAR Table 5.1-1 lists this value as the Full Power Operability Flow, gpm/loop.

The LCO requires three Reactor Coolant Flow - Low channels per loop to be OPERABLE in MODE 1 above P-7.

In MODE 1 above the P-8 setpoint, a loss of flow in one RCS loop could result in DNB conditions in the core because of the higher power level. In MODE 1 below the P-8 setpoint and above the P-7 setpoint, a loss of flow in two or more loops is required to actuate a reactor trip because of the lower power level and the greater margin to the design limit DNBR. Below the P-7 setpoint, all reactor trips on low flow are automatically blocked.

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

11. Undervoltage Reactor Coolant Pumps (RCPs)

The Undervoltage RCPs reactor trip Function ensures that protection is provided against violating the DNBR limit due to a loss of flow in two or more RCS loops. The voltage to each RCP is monitored. Above the P-7 setpoint, a loss of voltage detected on two or more RCP buses will initiate a reactor trip. This trip Function will generate a reactor trip before the Reactor Coolant Flow - Low Trip Setpoint is reached. Time delays are incorporated into the Undervoltage RCPs channels to prevent reactor trips due to momentary electrical power transients.

There are four (one per bus) Undervoltage RCP channels arranged in a two-out-of-four logic. The LCO requires four Undervoltage RCPs channels (one per bus) to be OPERABLE.

In MODE 1 above the P-7 setpoint, the Undervoltage RCP trip must be OPERABLE. Below the P-7 setpoint, all reactor trips on loss of flow are automatically blocked since no conceivable power distributions could occur that would cause a DNB concern at this low power level. Above the P-7 setpoint, the reactor trip on loss of flow in two or more RCS loops is automatically enabled.

12. Underfrequency Reactor Coolant Pumps

The Underfrequency RCPs reactor trip Function ensures that protection is provided against violating the DNBR limit due to a loss of flow in two or more RCS loops from a major network frequency disturbance. An underfrequency condition will slow down the pumps, thereby reducing their coastdown time following a pump trip. The proper coastdown time is required so that reactor heat can be removed immediately after reactor trip. The frequency of each RCP bus is monitored. Above the P-7 setpoint, a loss of frequency detected on two or more RCP buses will initiate a reactor trip. This trip Function will generate a reactor trip before the Reactor Coolant Flow - Low Trip Setpoint is reached. Time delays are incorporated into the Underfrequency RCPs channels to prevent reactor trips due to momentary electrical power transients.

There are four (one per bus) Underfrequency RCP channels arranged in a two-out-of-four logic. The LCO requires four Underfrequency RCPs channels (one per bus) to be OPERABLE.

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### APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

In MODE 1 above the P-7 setpoint, the Underfrequency RCPs trip must be OPERABLE. Below the P-7 setpoint, all reactor trips on loss of flow are automatically blocked since no conceivable power distributions could occur that would cause a DNB concern at this low power level. Above the P-7 setpoint, the reactor trip on loss of flow in two or more RCS loops is automatically enabled. Note that this Function also provides a signal to trip all four reactor coolant pumps.

#### 13. Steam Generator Water Level - Low Low

The Steam Generator Water Level Low-Low trip protects the reactor from loss of heat sink in the event of a sustained steam/feedwater flow mismatch resulting from loss of normal feedwater or a feedwater system pipe break outside of containment. This function also provides input to the steam generator level control system. IEEE-279 requirements are satisfied by 2/3 logic for protection function actuation, thus allowing for a single failure of a channel and still performing the protection function.

Control/protection interaction is addressed by the use of the Median Signal Selector that prevents a single failure of a channel providing input to the control system requiring protection function action. That is, a single failure of a channel providing input to the control system does not result in the control system initiating a condition requiring protection function action. The Median Signal Selector performs this by not selecting the channels indicating the highest or lowest steam generator levels as input to the control system.

With the transmitters located inside containment and thus possibly experiencing adverse environmental conditions (due to a feedline break), the Environmental Allowance Modifier (EAM) was devised. The EAM function (Containment Pressure (EAM) with a setpoint of < 0.5 psig) senses the presence of adverse containment conditions (elevated pressure) and enables the Steam Generator Water Level - Low-Low trip setpoint (Adverse) which reflects the increased transmitter uncertainties due to this environment. The EAM allows the use of a lower Steam Generator Water Level - Low-Low (EAM) trip setpoint when these conditions are not present, thus allowing more margin to trip for normal operating conditions.

The Trip Time Delay (TTD) creates additional operational margin when the plant needs it most, during early escalation to power, by allowing the operator time to recover level when the primary side load is sufficiently small to allow such action. The TTD is based on continuous monitoring of primary side power through the use of RCS loop  $\Delta T$ .

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Two time delays are calculated, based on the number of steam generators indicating less than the Low-Low Level trip setpoint and the primary side power level. The magnitude of the delays decreases with increasing primary side power level, up to 50% RTP. Above 50% RTP there are no time delays for the Low-Low level trips.

In the event of failure of a Steam Generator Water Level channel, it is placed in the trip condition as input to the Solid State Protection System and does not affect either the EAM or TTD setpoint calculations for the remaining operable channels. Failure of the Containment Pressure (EAM) channel to a protection set also does not affect the EAM setpoint calculations. It is then necessary for the operator to force the use of the shorter TTD by adjustment of the single steam generator time delay calculation ( $T_S$ ) to match the multiple steam generator time delay calculation ( $T_M$ ) for the affected protection set, through the Eagle-21 System Man-Machine-Interface (MMI) test cart. Failure of the RCS loop  $\Delta T$  channel input (failure of more than one  $T_H$  RTD or failure of a  $T_C$  RTD) does not affect the TTD calculation for a protection set. Although not affecting the TTD calculation, this results in the requirement that the operator adjust the threshold power level for zero seconds time delay from 50% RTP to 0% RTP, through the MMI, or place the affected protection sets Steam Generator Water Level - Low-Low channel in trip.

This Function also performs the ESFAS function of starting the Auxiliary Feedwater (AFW) pumps on low low SG level.

There are three Steam Generator Water Level Low-Low channels per steam generator arranged in a two-out-of-three logic. These channels are arranged in four protection sets with each channel of the Containment Pressure (EAM) and RCS Loop  $\Delta T$  inputting into its associated protection set. The LCO requires three channels of SG Water Level - Low Low per SG to be OPERABLE.

In MODE 1 or 2, when the reactor requires a heat sink, the SG Water Level - Low Low trip must be OPERABLE. The normal source of water for the SGs is the Main Feedwater (MFW) System (not safety related). The MFW System is only in operation in MODE 1 or 2. The AFW System is the safety related backup source of water to ensure that the SGs remain the heat sink for the reactor. During normal startups and shutdowns, the AFW System provides feedwater to maintain SG level. In MODE 3, 4, 5, or 6, the SG Water Level - Low Low Function does not have to be OPERABLE because the MFW System is not in operation and the reactor is not operating or even critical. Decay heat removal is accomplished by the AFW System

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

in MODE 3 and by the Residual Heat Removal (RHR) System in MODE 4, 5, or 6.

14. Turbine Trip

a. Turbine Trip - Low Fluid Oil Pressure

The Turbine Trip - Low Fluid Oil Pressure trip Function anticipates the loss of heat removal capabilities of the secondary system following a turbine trip. This trip Function acts to minimize the pressure/temperature transient on the reactor. Any turbine trip from a power level below the P-9 setpoint, approximately 50% power, will not actuate a reactor trip. Three pressure switches monitor the auto stop oil pressure in the Turbine Electrohydraulic Control System. A low pressure condition sensed by two-out-of-three pressure switches will actuate a reactor trip. These pressure switches do not provide any input to the control system. The unit is designed to withstand a complete loss of load and not sustain core damage or challenge the RCS pressure limitations. Core protection is provided by the Pressurizer Pressure - High trip Function and RCS integrity is ensured by the pressurizer safety valves.

The LCO requires three channels of Turbine Trip - Low Fluid Oil Pressure to be OPERABLE in MODE 1 above P-9.

Below the P-9 setpoint, a turbine trip does not actuate a reactor trip. In MODE 2, 3, 4, 5, or 6, there is no potential for a turbine trip, and the Turbine Trip - Low Fluid Oil Pressure trip Function does not need to be OPERABLE.

b. Turbine Trip - Turbine Stop Valve Closure

The Turbine Trip - Turbine Stop Valve Closure trip Function anticipates the loss of heat removal capabilities of the secondary system following a turbine trip from a power level above the P-9 setpoint, approximately 50% power. This action will actuate a reactor trip. The trip Function anticipates the loss of secondary heat removal capability that occurs when the stop valves close. Tripping the reactor in anticipation of loss of secondary heat removal acts to minimize the pressure and temperature transient on the reactor. This trip Function will not and is not required to operate in the presence of a single channel failure. The unit is designed to withstand a complete loss of load and not sustain core damage or challenge the RCS pressure limitations.

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### APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Core protection is provided by the Pressurizer Pressure - High trip Function, and RCS integrity is ensured by the pressurizer safety valves. This trip Function is diverse to the Turbine Trip - Low Fluid Oil Pressure trip Function. Each turbine stop valve is equipped with one limit switch that inputs to the RTS. If all four limit switches indicate that the stop valves are all closed, a reactor trip is initiated.

The LSSS for this Function is set to assure channel trip occurs when the associated stop valve is completely closed.

The LCO requires four Turbine Trip - Turbine Stop Valve Closure channels, one per valve, to be OPERABLE in MODE 1 above P-9. All four channels must trip to cause reactor trip.

Below the P-9 setpoint, a load rejection can be accommodated by the Steam Dump System. In MODE 2, 3, 4, 5, or 6, there is no potential for a load rejection, and the Turbine Trip - Stop Valve Closure trip Function does not need to be OPERABLE.

#### 15. Safety Injection Input from Engineered Safety Feature Actuation System

The SI Input from ESFAS ensures that if a reactor trip has not already been generated by the RTS, the ESFAS automatic actuation logic will initiate a reactor trip upon any signal that initiates SI. This is not a condition of acceptability for the LOCA. However, other transients and accidents take credit for varying levels of ESF performance and rely upon rod insertion, except for the most reactive rod that is assumed to be fully withdrawn, to ensure reactor shutdown. Therefore, a reactor trip is initiated every time an SI signal is present.

Trip Setpoint and Allowable Values are not applicable to this Function. The SI Input is provided by solid state logic in the ESFAS. Therefore, there is no measurement signal with which to associate an LSSS.

There are two trains of SI input from ESFAS arranged in a one-out-of-two logic. The LCO requires two trains of SI Input from ESFAS to be OPERABLE in MODE 1 or 2.

A reactor trip is initiated every time an SI signal is present. Therefore, this trip Function must be OPERABLE in MODE 1 or 2, when the reactor is critical, and must be shut down in the event of an accident. In MODE 3, 4, 5, or 6, the reactor is not critical,

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### APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

and this trip Function does not need to be OPERABLE.

#### 16. Reactor Trip System Interlocks

Reactor protection interlocks are provided to ensure reactor trips are in the correct configuration for the current unit status. They back up operator actions to ensure protection system Functions are not bypassed during unit conditions under which the safety analysis assumes the Functions are not bypassed. Therefore, the interlock Functions do not need to be OPERABLE when the associated reactor trip functions are outside the applicable MODES. These are:

##### a. Intermediate Range Neutron Flux, P-6

The Intermediate Range Neutron Flux, P-6 interlock is actuated when any NIS intermediate range channel goes approximately four decades above the minimum channel reading. If both channels drop below the setpoint, the permissive will automatically be defeated. The LCO requirement for the P-6 interlock ensures that the following Functions are performed:

- on increasing power, the P-6 interlock allows the manual block of the NIS Source Range, Neutron Flux reactor trip. This prevents a premature block of the source range trip and allows the operator to ensure that the intermediate range is OPERABLE prior to leaving the source range. When the source range trip is blocked, the input to the SR drawer is shorted out driving the output of drawer to zero, and
- on decreasing power, the P-6 interlock automatically energizes the NIS source range detectors and enables the NIS Source Range Neutron Flux reactor trip.

There are two Intermediate Range Neutron Flux, P-6 channels arranged in a one-out-of-two logic. The LCO requires two channels of Intermediate Range Neutron Flux, P-6 interlock to be OPERABLE in MODE 2 when below the P-6 interlock setpoint.

Above the P-6 interlock setpoint, the NIS Source Range Neutron Flux reactor trip will be blocked, and this Function will no longer be necessary.

In MODE 3, 4, 5, or 6, the P-6 interlock does not have to be OPERABLE because the NIS Source Range is providing core protection.

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

b. Low Power Reactor Trips Block, P-7

The Low Power Reactor Trips Block, P-7 interlock is actuated by input from either the Power Range Neutron Flux, P-10, or the Turbine Impulse Pressure, P-13 interlock. The LCO requirement for the P-7 interlock ensures that the following Functions are performed:

(1) on increasing power, the P-7 interlock automatically enables reactor trips on the following Functions:

- Pressurizer Pressure - Low,
- Pressurizer Water Level - High,
- Reactor Coolant Flow - Low (low flow in two or more RCS loops),
- Undervoltage RCPs, and
- Underfrequency RCPs.

These reactor trips are only required when operating above the P-7 setpoint (approximately 10% power). The reactor trips provide protection against violating the DNBR limit. Below the P-7 setpoint, the RCS is capable of providing sufficient natural circulation without any RCP running.

(2) on decreasing power, the P-7 interlock automatically blocks reactor trips on the following Functions:

- Pressurizer Pressure - Low,
- Pressurizer Water Level - High,
- Reactor Coolant Flow - Low (low flow in two or more RCS loops),
- Undervoltage RCPs, and
- Underfrequency RCPs.

Trip Setpoint and Allowable Value are not applicable to the P-7 interlock because it is a logic Function and thus has no parameter with which to associate an LSSS.

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### APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The P-7 interlock is a logic Function with train and not channel identity. Therefore, the LCO requires one channel per train of Low Power Reactor Trips Block, P-7 interlock to be OPERABLE in MODE 1.

The low power trips are blocked below the P-7 setpoint and unblocked above the P-7 setpoint. In MODE 2, 3, 4, 5, or 6, this Function does not have to be OPERABLE because the interlock performs its Function when power level drops below 10% power, which is in MODE 1.

c. Power Range Neutron Flux, P-8

The Power Range Neutron Flux, P-8 interlock is actuated at approximately 35% power as determined by two-out-of-four NIS power range detectors. The P-8 interlock automatically enables the Reactor Coolant Flow - Low reactor trip on low flow in one or more RCS loops on increasing power. The LCO requirement for this trip Function ensures that protection is provided against a loss of flow in any RCS loop that could result in DNB conditions in the core when greater than approximately 35% power. On decreasing power, the reactor trip on low flow in any loop is automatically blocked.

The LCO requires four channels of Power Range Neutron Flux, P-8 interlock to be OPERABLE in MODE 1.

In MODE 1, a loss of flow in one RCS loop could result in DNB conditions, so the Power Range Neutron Flux, P-8 interlock must be OPERABLE. In MODE 2, 3, 4, 5, or 6, this Function does not have to be OPERABLE because the core is not producing sufficient power to be concerned about DNB conditions.

d. Power Range Neutron Flux, P-9

The Power Range Neutron Flux, P-9 interlock is actuated at approximately 50% power as determined by two-out-of-four NIS power range detectors. The LCO requirement for this Function ensures that the Turbine Trip - Low Fluid Oil Pressure and Turbine Trip - Turbine Stop Valve Closure reactor trips are enabled above the P-9 setpoint. Above the P-9 setpoint, a turbine trip will cause a load rejection beyond the capacity of the Steam Dump System. A reactor trip is automatically initiated on a turbine trip when it is above the P-9 setpoint, to minimize the transient on the reactor.

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### APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The LCO requires four channels of Power Range Neutron Flux, P-9 interlock to be OPERABLE in MODE 1.

In MODE 1, a turbine trip could cause a load rejection beyond the capacity of the Steam Dump System, so the Power Range Neutron Flux interlock must be OPERABLE. In MODE 2, 3, 4, 5, or 6, this Function does not have to be OPERABLE because the reactor is not at a power level sufficient to have a load rejection beyond the capacity of the Steam Dump System.

e. Power Range Neutron Flux, P-10

The Power Range Neutron Flux, P-10 interlock is actuated at approximately 10% power, as determined by two-out-of-four NIS power range detectors. If power level falls below 10% RTP on 3 of 4 channels, the nuclear instrument trips will be automatically unblocked. The LCO requirement for the P-10 interlock ensures that the following Functions are performed:

- on increasing power, the P-10 interlock allows the operator to manually block the Intermediate Range Neutron Flux reactor trip. Note that blocking the reactor trip also blocks the signal to prevent automatic and manual rod withdrawal,
- on increasing power, the P-10 interlock allows the operator to manually block the Power Range Neutron Flux - Low reactor trip,
- on increasing power, the P-10 interlock automatically provides a backup signal to block the Source Range Neutron Flux reactor trip, and also shorts out the input to the SR drawer, driving the output of drawer to zero,
- the P-10 interlock provides one of the two inputs to the P-7 interlock, and
- on decreasing power, the P-10 interlock automatically enables the Power Range Neutron Flux - Low reactor trip and the Intermediate Range Neutron Flux reactor trip (and rod stop).

The LCO requires four channels of Power Range Neutron Flux, P-10 interlock to be OPERABLE in MODE 1 or 2.

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### APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

OPERABILITY in MODE 1 ensures the Function is available to perform its decreasing power Functions in the event of a reactor shutdown. This Function must be OPERABLE in MODE 2 to ensure that core protection is provided during a startup or shutdown by the Power Range Neutron Flux - Low and Intermediate Range Neutron Flux reactor trips. In MODE 3, 4, 5, or 6, this Function does not have to be OPERABLE because the reactor is not at power and the Source Range Neutron Flux reactor trip provides core protection.

f. Turbine Impulse Pressure, P-13

The Turbine Impulse Pressure, P-13 interlock is actuated when the pressure in the first stage of the high pressure turbine is greater than approximately 10% of the rated full power pressure. This is determined by one-out-of-two pressure detectors. The LCO requirement for this Function ensures that one of the inputs to the P-7 interlock is available.

The LCO requires two channels of Turbine Impulse Pressure, P-13 interlock to be OPERABLE in MODE 1.

The Turbine Impulse Chamber Pressure, P-13 interlock must be OPERABLE when the turbine generator is operating. The interlock Function is not required OPERABLE in MODE 2, 3, 4, 5, or 6 because the turbine generator is not operating.

17. Reactor Trip Breakers

This trip Function applies to the reactor trip breakers exclusive of individual trip mechanisms. There are two Reactor Trip Breakers arranged in a one-out-of-two logic. The LCO requires two OPERABLE trains of trip breakers. A trip breaker train consists of all trip breakers associated with a single RTS logic train that are racked in, closed, and capable of supplying power to the Rod Control System. Thus, the train may consist of the main breaker, bypass breaker, or main breaker and bypass breaker, depending upon the system configuration. Two OPERABLE trains ensure no single random failure can disable the RTS trip capability.

These trip Functions must be OPERABLE in MODE 1 or 2 when the reactor is critical. In MODE 3, 4, or 5, these RTS trip Functions must be OPERABLE when the Rod Control System is capable of rod withdrawal or one or more rods are not fully inserted.

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

18. Reactor Trip Breaker Undervoltage and Shunt Trip Mechanisms

The LCO requires both the Undervoltage and Shunt Trip Mechanisms to be OPERABLE for each reactor trip breaker that is in service. The trip mechanisms are not required to be OPERABLE for trip breakers that are open, racked out, incapable of supplying power to the Rod Control System, or declared inoperable under Function 17 above. OPERABILITY of both trip mechanisms on each breaker ensures that no single trip mechanism failure will prevent opening any breaker on a valid signal.

These trip Functions must be OPERABLE in MODE 1 or 2 when the reactor is critical. In MODE 3, 4, or 5, these RTS trip Functions must be OPERABLE when the Rod Control System is capable of rod withdrawal or one or more rods are not fully inserted.

19. Automatic Trip Logic

The LCO requirement for the reactor trip breakers (Functions 17 and 18) and Automatic Trip Logic (Function 19) ensures that means are provided to interrupt the power to allow the rods to fall into the reactor core. Each reactor trip breaker is equipped with an undervoltage coil and a shunt trip coil to trip the breaker open when needed. Each reactor trip breaker is equipped with a bypass breaker to allow testing of the trip breaker while the unit is at power. The reactor trip signals generated by the RTS Automatic Trip Logic cause the reactor trip breakers and associated bypass breakers to open and shut down the reactor.

There are two RTS Automatic Trip Logic trains arranged in a one-out-of-two logic. The LCO requires two trains of RTS Automatic Trip Logic to be OPERABLE. Having two OPERABLE channels ensures that random failure of a single logic channel will not prevent reactor trip.

These trip Functions must be OPERABLE in MODE 1 or 2 when the reactor is critical. In MODE 3, 4, or 5, these RTS trip Functions must be OPERABLE when the Rod Control System is capable of rod withdrawal or one or more rods are not fully inserted.

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

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### ACTIONS

A Note has been added to the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed in Table 3.3.1-1.

In the event a channel's trip setting is found nonconservative with respect to the Allowable Value, or the channel is not functioning as required, or the transmitter, instrument loop, signal processing electronics, setpoint comparator trip output, contact output, or bistable is found inoperable, then all affected Functions provided by that channel must be declared inoperable and the LCO Condition(s) entered for the protection Function(s) affected. When the Required Channels in Table 3.3.1-1 are specified on a "per" basis (e.g., on a per steam line loop, per SG, etc., basis), then the Condition may be entered separately for each steam line, loop, SG, etc., as appropriate.

When the number of inoperable channels in a trip Function exceed those specified in one or other related Conditions associated with a trip Function, then the unit is outside the safety analysis. Therefore, LCO 3.0.3 must be immediately entered if applicable in the current MODE of operation.

#### A.1

Condition A applies to all RTS protection Functions. Condition A addresses the situation where one or more required channels or trains for one or more Functions are inoperable at the same time. The Required Action is to refer to Table 3.3.1-1 and to take the Required Actions for the protection functions affected. The Completion Times are those from the referenced Conditions and Required Actions.

#### B.1 and B.2

Condition B applies to the Manual Reactor Trip in MODE 1 or 2. This action addresses the train orientation of the SSPS for this Function. With one channel inoperable, the inoperable channel must be restored to OPERABLE status within 48 hours. In this Condition, the remaining OPERABLE channel is adequate to perform the safety function.

The Completion Time of 48 hours is reasonable considering that there are two automatic actuation trains and another manual initiation channel OPERABLE, and the low probability of an event occurring during this interval.

If the Manual Reactor Trip Function cannot be restored to OPERABLE status within the allowed 48 hour Completion Time, the unit must be brought to a MODE in which the requirement does not apply. To achieve

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### ACTIONS (continued)

this status, the unit must be brought to at least MODE 3 within 6 additional hours (54 hours total time). The 6 additional hours to reach MODE 3 is reasonable, based on operating experience, to reach MODE 3 from full power operation in an orderly manner and without challenging unit systems. With the unit in MODE 3, ACTION C would apply to any inoperable Manual Reactor Trip Function if the Rod Control System is capable of rod withdrawal or one or more rods are not fully inserted.

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

Condition C applies to the following reactor trip Functions in MODE 3, 4, or 5 with the Rod Control System capable of rod withdrawal or one or more rods not fully inserted:

- Manual Reactor Trip,
- reactor trip breakers,
- reactor trip breaker Undervoltage and Shunt Trip Mechanisms, and
- Automatic Trip Logic.

This action addresses the train orientation of the SSPS for these Functions. With one channel or train inoperable, the inoperable channel or train must be restored to OPERABLE status within 48 hours. If the affected Function(s) cannot be restored to OPERABLE status within the allowed 48 hour Completion Time, the unit must be placed in a MODE in which the requirement does not apply. To achieve this status, action must be initiated within the same 48 hours to ensure that all rods are fully inserted, and the Rod Control System must be placed in a condition incapable of rod withdrawal within the next hour. The additional hour provides sufficient time to accomplish the action in an orderly manner. With rods fully inserted and the Rod Control System incapable of rod withdrawal, these Functions are no longer required.

The Completion Time is reasonable considering that in this Condition, the remaining OPERABLE train is adequate to perform the safety function, and given the low probability of an event occurring during this interval.

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### ACTIONS (continued)

#### D.1 and D.2

Condition D applies to the Power Range Neutron Flux - High Function.

The NIS power range detectors provide input to the Rod Control System and therefore, have a two-out-of-four trip logic. A known inoperable channel must be placed in the tripped condition. This results in a partial trip condition requiring only one-out-of-three logic for actuation. The 72 hours allowed by Required Action D.1 to place the inoperable channel in the tripped condition is justified in WCAP-14333-P-A (Ref. 8).

If Required Action D.1 cannot be met within the specified Completion Time, the plant must be placed in a MODE where this Function is no longer required OPERABLE. Seventy-eight hours are allowed to place the plant in MODE 3. The 78 hour Completion Time includes 6 hours for the MODE reduction as required by Required Action D.2. This is a reasonable time, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging plant systems.

The Required Actions have been modified by two Notes. Note 1 allows the inoperable channel to be placed in the bypassed condition for up to 12 hours while performing routine surveillance testing of other channels. With one channel inoperable, the Note also allows routine surveillance testing of another channel with the inoperable channel in bypass. The Note also allows placing the inoperable channel in the bypass condition to allow setpoint adjustments of other channels when required to reduce the Power Range Neutron Flux-High setpoint in accordance with other Technical Specifications. The 12 hour time limit is justified in Reference 8.

Note 2 states to perform SR 3.2.4.2 if input to QPTR from one or more Power Range Neutron Flux channels are inoperable with THERMAL POWER >75% RTP.

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### ACTIONS (continued)

#### E.1 and E.2

Condition E applies to the following reactor trip Functions:

- Power Range Neutron Flux - Low,
- Overtemperature  $\Delta T$ ,
- Overpower  $\Delta T$ ,
- Power Range Neutron Flux - High Positive Rate,
- Power Range Neutron Flux - High Negative Rate, and
- Pressurizer Pressure - High.

A known inoperable channel must be placed in the tripped condition within 72 hours. Placing the channel in the tripped condition results in a partial trip condition requiring only one-out-of-two logic for actuation of the two-out-of-three trips and one-out-of-three logic for actuation of the two-out-of-four trips. The 72 hours allowed to place the inoperable channel in the tripped condition is justified in Reference 8.

If the inoperable channel cannot be placed in the trip condition within the specified Completion Time, the unit must be placed in a MODE where these Functions are not required OPERABLE. An additional 6 hours is allowed to place the unit in MODE 3. Six hours is a reasonable time, based on operating experience, to place the unit in MODE 3 from full power in an orderly manner and without challenging unit systems.

The Required Actions have been modified by a Note that allows placing the inoperable channel in the bypassed condition for up to 12 hours while performing routine surveillance testing of the other channels. The 12 hour time limit is justified in Reference 8.

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ACTIONS (continued)

F.1 and F.2

Condition F applies to the Intermediate Range Neutron Flux trip when THERMAL POWER is above the P-6 setpoint and below the P-10 setpoint and one channel is inoperable. Above the P-6 setpoint and below the P-10 setpoint, the NIS intermediate range detector performs the monitoring Functions. If THERMAL POWER is greater than the P-6 setpoint but less than the P-10 setpoint, 24 hours is allowed to reduce THERMAL POWER below the P-6 setpoint or to increase THERMAL POWER above the P-10 setpoint. The NIS Intermediate Range Neutron Flux channels must be OPERABLE when the power level is above the capability of the source range, P-6, and below the capability of the power range, P-10. If THERMAL POWER is greater than the P-10 setpoint, the NIS power range detectors perform the monitoring and protection functions and the intermediate range is not required. The Completion Times allow for a slow and controlled power adjustment above P-10 or below P-6 and take into account the redundant capability afforded by the redundant OPERABLE channel, and the low probability of its failure during this period. This action does not require the inoperable channel to be tripped because the Function uses one-out-of-two logic. Tripping one channel would trip the reactor. Thus, the Required Actions specified in this Condition are only applicable when channel failure does not result in reactor trip.

G.1 and G.2

Condition G applies to two inoperable Intermediate Range Neutron Flux trip channels in MODE 2 when THERMAL POWER is above the P-6 setpoint and below the P-10 setpoint. Required Actions specified in this Condition are only applicable when channel failures do not result in reactor trip. Above the P-6 setpoint and below the P-10 setpoint, the NIS intermediate range detector performs the monitoring Functions. With no intermediate range channels OPERABLE, the Required Actions are to suspend operations involving positive reactivity additions immediately. This will preclude any power level increase since there are no OPERABLE Intermediate Range Neutron Flux channels. The operator must also reduce THERMAL POWER below the P-6 setpoint within two hours. Below P-6, the Source Range Neutron Flux channels will be able to monitor the core power level. The Completion Time of 2 hours will allow a slow and controlled power reduction to less than the P-6 setpoint and takes into account the low probability of occurrence of an event during this period that may require the protection afforded by the NIS Intermediate Range Neutron Flux trip.

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### ACTIONS (continued)

Required Action G.1 is modified by a Note to indicate that normal plant control operations that individually add limited positive reactivity (e.g., temperature or boron fluctuations associated with RCS inventory management or temperature control) are not precluded by this Action, provided they are accounted for in the calculated SDM.

#### H.1

Condition H applies to one inoperable Source Range Neutron Flux trip channel when in MODE 2, below the P-6 setpoint, and performing a reactor startup. With the unit in this Condition, below P-6, the NIS source range performs the monitoring and protection functions. With one of the two channels inoperable, operations involving positive reactivity additions shall be suspended immediately.

This will preclude any power escalation. With only one source range channel OPERABLE, core protection is severely reduced and any actions that add positive reactivity to the core must be suspended immediately.

Required Action H.1 is modified by a Note to indicate that normal plant control operations that individually add limited positive reactivity (e.g., temperature or boron fluctuations associated with RCS inventory management or temperature control) are not precluded by this Action, provided they are accounted for in the calculated SDM.

#### I.1

Condition I applies to two inoperable Source Range Neutron Flux trip channels when in MODE 2, below the P-6 setpoint, and in MODE 3, 4, or 5 with the Rod Control System capable of rod withdrawal or one or more rods not fully inserted. With the unit in this Condition, below P-6, the NIS source range performs the monitoring and protection functions. With both source range channels inoperable, the reactor trip breakers must be opened immediately. With the reactor trip breakers open, the core is in a more stable condition.

#### J.1, J.2.1, and J.2.2

Condition J applies to one inoperable source range channel in MODE 3, 4, or 5 with the Rod Control System capable of rod withdrawal or one or more rods not fully inserted. With the unit in this Condition, below P-6, the NIS source range performs the monitoring and protection functions. With one of the source range channels inoperable, 48 hours is allowed to

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### ACTIONS (continued)

restore it to an OPERABLE status. If the channel cannot be returned to an OPERABLE status, action must be initiated within the same 48 hours to ensure that all rods are fully inserted, and the Rod Control System must be placed in a condition incapable of rod withdrawal within the next hour.

#### K.1 and K.2

Condition K applies to the following reactor trip Functions:

- Pressurizer Pressure - Low,
- Pressurizer Water Level - High,
- Reactor Coolant Flow – Low,
- Undervoltage RCPs, and
- Underfrequency RCPs.

With one channel inoperable, the inoperable channel must be placed in the tripped condition within 72 hours (Ref. 8). For the Pressurizer Pressure - Low, Pressurizer Water Level - High, Undervoltage RCPs, and Underfrequency RCPs trip Functions, placing the channel in the tripped condition when above the P-7 setpoint results in a partial trip condition requiring only one additional channel to initiate a reactor trip. For the Reactor Coolant Flow - Low trip Function, placing the channel in the tripped condition when above the P-8 setpoint results in a partial trip condition requiring only one additional channel in the same loop to initiate a reactor trip. For the latter trip Function, two tripped channels in two RCS loops are required to initiate a reactor trip when below the P-8 setpoint and above the P-7 setpoint. These Functions do not have to be OPERABLE below the P-7 setpoint because there are no loss of flow trips below the P-7 setpoint. There is insufficient heat production to generate DNB conditions below the P-7 setpoint. The 72 hours allowed to place the channel in the tripped condition is justified in Reference 8. An additional 6 hours is allowed to reduce THERMAL POWER to below P-7 if the inoperable channel cannot be restored to OPERABLE status or placed in trip within the specified Completion Time.

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### ACTIONS (continued)

Allowance of this time interval takes into consideration the redundant capability provided by the remaining redundant OPERABLE channel, and the low probability of occurrence of an event during this period that may require the protection afforded by the Functions associated with Condition K.

The Required Actions have been modified by a Note that allows placing the inoperable channel in the bypassed condition for up to 12 hours while performing routine surveillance testing of the other channels. The 12 hour time limit is justified in Reference 8.

#### L.1 and L.2

Condition L applies to Turbine Trip on Low Fluid Oil Pressure or on Turbine Stop Valve Closure. With one channel inoperable, the inoperable channel must be placed in the trip condition within 72 hours. If placed in the tripped condition, this results in a partial trip condition requiring only one additional Low Fluid Oil Pressure channel or three additional Turbine Stop Valve Closure channels to initiate a reactor trip. If the channel cannot be restored to OPERABLE status or placed in the trip condition, then power must be reduced below the P-9 setpoint within the next 4 hours. The 72 hours allowed to place the inoperable channel in the tripped condition is justified in Reference 8. Four hours is allowed for reducing power.

The Required Actions have been modified by a Note that allows placing the inoperable channel in the bypassed condition for up to 12 hours while performing routine surveillance testing of the other channels. The 12 hour time limit is justified in Reference 8.

#### M.1 and M.2

Condition M applies to the SI Input from ESFAS reactor trip and the RTS Automatic Trip Logic in MODES 1 and 2. These actions address the train orientation of the RTS for these Functions. With one train inoperable, 24 hours are allowed to restore the train to OPERABLE status (Required Action M.1) or the unit must be placed in MODE 3 within the next 6 hours. The Completion Time of 24 hours (Required Action M.1) is reasonable considering that in this Condition, the remaining OPERABLE train is adequate to perform the safety function and given the low probability of an event during this interval. The 24 hours allowed to restore the inoperable RTS Automatic Trip Logic train to OPERABLE status is justified in Reference 8. An additional 6 hours is allowed to place the unit in MODE 3. Six hours is reasonable, based on operating experience,

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### ACTIONS (continued)

to reach MODE 3 from full power in an orderly manner and without challenging unit systems.

The Required Actions have been modified by a Note that allows bypassing one train up to 4 hours for surveillance testing, provided the other train is OPERABLE. The 4 hour time limit for testing the RTS Automatic Trip logic train may include testing the reactor trip breaker also, if both the Logic test and reactor trip breaker test are conducted within the 4 hour time limit. The 4 hour time limit is justified in Reference 8.

#### N.1 and N.2

Condition N applies to the reactor trip breakers in MODES 1 and 2. These actions address the train orientation of the RTS for the reactor trip breakers. With one train inoperable, 24 hours is allowed for train corrective maintenance to restore the train to OPERABLE status or the unit must be placed in MODE 3 within the next 6 hours. The 24 hour Completion Time is justified in Reference 12. An additional 6 hours is allowed to place the unit in MODE 3. Six hours is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging unit systems.

Placing the unit in MODE 3 results in Condition C entry while a reactor trip breaker is inoperable.

The Required Actions have been modified by a Note. The Note allows one train to be bypassed for up to 4 hours for surveillance testing, provided the other train is OPERABLE. The 4 hour time limit is justified in Reference 12.

#### O.1 and O.2

Condition O applies to the P-6 and P-10 interlocks. With one or more channels inoperable for one-out-of-two or two-out-of-four coincidence logic, the associated interlock must be verified to be in its required state for the existing unit condition within 1 hour or the unit must be placed in MODE 3 within the next 6 hours. Verifying the interlock status manually accomplishes the interlock's Function. The Completion Time of 1 hour is based on operating experience and the minimum amount of time allowed for manual operator actions. An additional 6 hours is allowed to place the unit in MODE 3. Six hours is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging unit systems. The 1 hour and 6 hour Completion

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### ACTIONS (continued)

Times are equal to the time allowed by LCO 3.0.3 for shutdown actions in the event of a complete loss of RTS Function.

#### P.1 and P.2

Condition P applies to the P-7, P-8, P-9, and P-13 interlocks. With one or more channels inoperable for one-out-of-two or two-out-of-four coincidence logic, the associated interlock must be verified to be in its required state for the existing unit condition within 1 hour or the unit must be placed in MODE 2 within the next 6 hours. These actions are conservative for the case where power level is being raised. Verifying the interlock status manually accomplishes the interlock's Function. The Completion Time of 1 hour is based on operating experience and the minimum amount of time allowed for manual operator actions. An additional 6 hours is allowed to place the unit in MODE 2. Six hours is reasonable, based on operating experience, to reach MODE 2 from full power in an orderly manner and without challenging unit systems.

#### Q.1 and Q.2

Condition Q applies to the reactor trip breaker Undervoltage and Shunt Trip Mechanisms, or diverse trip features, in MODES 1 and 2. With one of the diverse trip features inoperable, it must be restored to an OPERABLE status within 48 hours or the unit must be placed in a MODE where the requirement does not apply. This is accomplished by placing the unit in MODE 3 within the next 6 hours (54 hours total time). Six hours is a reasonable time, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging unit systems. With the unit in MODE 3, ACTION C would apply to any inoperable reactor trip breaker trip mechanism. The Required Actions have been modified by a Note. The Note states that the affected reactor trip breaker shall not be bypassed while one of the diverse features is inoperable except for up to 4 hours to perform maintenance to one of the diverse features. The allowable time for performing maintenance of the diverse features is 4 hours for the reasons stated under Condition N.

The Completion Time of 48 hours for Required Action Q.1 is reasonable considering that in this Condition there is one remaining diverse feature for the affected reactor trip breaker, and one OPERABLE reactor trip breaker capable of performing the safety function and given the low probability of an event occurring during this interval.

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### ACTIONS (continued)

#### R.1 and R.2

Condition R applies to the following reactor trip Functions:

- Steam Generator Water Level--Low-Low (Adverse), and
- Steam Generator Water Level--Low-Low (EAM)

A known inoperable channel must be placed in the tripped condition within 6 hours. Placing the channel in the tripped condition results in a partial trip condition requiring only one-out-of-two logic for actuation of the two-out-of-three trips.

In addition to placing the channel in the tripped condition it is also necessary to force the use of the shorter TTD by adjustment of the single steam generator time delay calculation ( $T_S$ ) to match the multiple steam generator time delay calculation ( $T_M$ ) for the affected protection set within 4 hours.

The Required Actions have been modified by a Note that allows placing the inoperable channel in the bypassed condition for up to 4 hours while performing routine surveillance testing of the other channels.

#### S.1, S.2, and S.3

Condition S applies to the Containment Pressure (EAM) coincident with Steam Generator Water Level--Low-Low (Adverse) reactor trip.

Failure of the Containment Pressure (EAM) channel to a protection set does not affect the EAM setpoint calculations. A known inoperable Containment Pressure channel results in the requirement to adjust the Steam Generator Water Level - Low-Low (EAM) channels trip setpoints for the affected protection set to the same value as Steam Generator Water Level - Low-Low (Adverse) within 6 hours.

An alternative to adjusting the affected Steam Generator Water Level - Low-Low (EAM) trip setpoints to the same value as the Steam Generator Water Level - Low-Low (Adverse) trip setpoints is to place the associated protection set's SG Water Level Low-Low channels in the tripped condition within 6 hours

If neither of the above Required Actions are completed within their associated Completion Time, then the unit must be placed in a MODE where these Functions are not required OPERABLE. This requires the

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### ACTIONS (continued)

unit be placed in MODE 3 within 12 hours. The allowed Completion Times are reasonable to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. In MODE 3, these Functions are no longer required OPERABLE.

#### T.1, T.2, and T.3

Condition T applies to the RCS Loop  $\Delta T$  coincident with SG Water Level -  
- Low Low reactor trips.

Failure of the RCS loop  $\Delta T$  channel input (failure of more than one  $T_H$  RTD or failure of a  $T_C$  RTD) does not affect the TTD calculation for a protection set. This results in the requirement that the operator adjust the threshold power level for zero seconds time delay from 50% RTP to 0% RTP within 6 hours. With the trip time delay adjusted to zero seconds the additional operational margin that allows the operator time to recover SG level is removed.

An alternative to adjusting the threshold power level for zero seconds time delay is to place the affected protection set's SG Water Level Low-Low level channels in the tripped condition within 6 hours.

If neither of the above Required Actions can be completed within their associated Completion Times then the unit must be placed in a MODE where these Functions are not required OPERABLE. This requires the unit be placed in MODE 3 within 12 hours. The allowed Completion Times are reasonable to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. In MODE 3, these Functions are no longer required OPERABLE.

#### U.1

If the Required Action is not met within the specified Completion Time of Condition R, the unit must be placed in a MODE where this Function is not required OPERABLE. Six hours is allowed to place the unit in MODE 3. Six hours is a reasonable time, based on operating experience, to place the unit in MODE 3 from full power in an orderly manner and without challenging unit systems.

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### SURVEILLANCE REQUIREMENTS

The SRs for each RTS Function are identified by the SRs column of Table 3.3.1-1 for that Function.

A Note has been added to the SR Table stating that Table 3.3.1-1 determines which SRs apply to which RTS Functions.

Note that each channel of process protection supplies both trains of the RTS. When testing Channel I, Train A and Train B must be examined. Similarly, Train A and Train B must be examined when testing Channel II, Channel III, and Channel IV (if applicable). The CHANNEL CALIBRATION and COTs are performed in a manner that is consistent with the assumptions used in analytically calculating the required channel accuracies.

#### SR 3.3.1.1

Performance of the CHANNEL CHECK ensures that gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the unit staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.2

SR 3.3.1.2 compares the calorimetric heat balance calculation to the power range channel output. If the absolute difference is greater than 2 percent, the power range channel is not declared inoperable, but must be adjusted. The power range channel output shall be adjusted consistent with the calorimetric heat balance calculation results if the absolute difference is greater than 2 percent. If the power range channel output cannot be properly adjusted, the channel is declared inoperable.

The Note clarifies that this Surveillance is required only if reactor power is  $\geq 15\%$  RTP and that 12 hours are allowed for performing the first Surveillance after reaching 15% RTP. A power level of 15% RTP is chosen based on plant stability, i.e., automatic rod control capability and turbine generator synchronized to the grid.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.3.1.3

SR 3.3.1.3 compares the incore system to the NIS channel output. If the absolute difference is  $\geq 3\%$ , the NIS channel is still OPERABLE, but must be readjusted. The excore NIS channel shall be adjusted if the absolute difference between the incore and excore AFD is  $\geq 3\%$ .

If the NIS channel cannot be properly readjusted, the channel is declared inoperable. This Surveillance is performed to verify the  $f(\Delta I)$  input to the overtemperature  $\Delta T$  and overpower  $\Delta T$  Functions.

A Note clarifies that the Surveillance is required only if reactor power is  $\geq 15\%$  RTP and that 96 hours is allowed for performing the first Surveillance after reaching 15% RTP.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.4

SR 3.3.1.4 is the performance of a TADOT. This test shall verify OPERABILITY by actuation of the end devices. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable TADOT of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions.

The reactor trip breaker test shall include separate verification of the undervoltage and shunt trip mechanisms. Independent verification of reactor trip breaker undervoltage and shunt trip Function is not required for the bypass breakers. No capability is provided for performing such a test at power. The independent test for bypass breakers is included in SR 3.3.1.12. The bypass breaker test shall include a local shunt trip. A Note has been added to indicate that this test must be performed on the bypass breaker prior to placing it in service.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.3.1.5

SR 3.3.1.5 is the performance of an ACTUATION LOGIC TEST. The SSPS is tested using the semiautomatic tester. The train being tested is placed in the bypass condition, thus preventing inadvertent actuation. Through the semiautomatic tester, all possible logic combinations, with and without applicable permissives, are tested for each protection function, including operation of the P-7 permissive which is a logic function only.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.3.1.6

SR 3.3.1.6 is a calibration of the excore channels to the incore channels. If the measurements do not agree, the excore channels are not declared inoperable but must be calibrated to agree with the incore detector measurements. If the excore channels cannot be adjusted, the channels are declared inoperable. This Surveillance is performed to verify the  $f(\Delta I)$  input to the overtemperature  $\Delta T$  and overpower  $\Delta T$  Functions.

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### SURVEILLANCE REQUIREMENTS (continued)

A Note modifies SR 3.3.1.6. The Note states that this Surveillance is required only if reactor power is > 50% RTP and that 24 hours is allowed for performing the first surveillance after reaching 50% RTP.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

#### SR 3.3.1.7

SR 3.3.1.7 is the performance of a COT.

A COT is performed on each required channel to ensure the entire channel will perform the intended Function. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable COT of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions.

Setpoints must be conservative with respect to the Allowable Values specified in Table 3.3.1-1.

The difference between the current "as found" values and the previous test "as left" values must be consistent with the drift allowance used in the setpoint methodology. The setpoint shall be left set consistent with the assumptions of the current unit specific setpoint methodology.

The "as-found" and "as-left" values must also be recorded and reviewed for consistency with the assumptions of Reference 9.

SR 3.3.1.7 is modified by a Note that provides a 4 hour delay in the requirement to perform this Surveillance for source range instrumentation when entering MODE 3 from MODE 2. This Note allows a normal shutdown to proceed without a delay for testing in MODE 2 and for a short time in MODE 3 until the reactor trip breakers are open and SR 3.3.1.7 is no longer required to be performed. If the unit is to be in MODE 3 with the reactor trip breakers closed for > 4 hours this Surveillance must be performed prior to 4 hours after entry into MODE 3.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.7 is modified by two Notes as identified in Table 3.3.1-1. The first Note requires evaluation of channel performance for the condition where the as-found setting for the channel setpoint is outside its as-found tolerance but conservative with respect to the Allowable Value. Evaluation of channel performance will verify that the channel will continue to behave in accordance with safety analysis assumptions and the channel performance assumptions in the setpoint methodology. The purpose of the assessment is to ensure confidence in the channel performance prior to returning the channel to service. For channels determined to be OPERABLE but degraded, after returning the channel to service the channels will be evaluated under the plant Corrective Action Program. Entry into the Corrective Action Program will ensure required review and documentation of the condition. The second Note requires that the as-left setting for the channel be returned to within the as-left tolerance of the NTSP. Where a setpoint more conservative than the NTSP is used in the plant surveillance procedures (field setting), the as-left and as-found tolerances, as applicable, will be applied to the surveillance procedure setpoint. This will ensure that sufficient margin to the Safety Limit and/or Analytical Limit is maintained. If the as-left channel setting cannot be returned to a setting within the as-left tolerance of the NTSP, then the channel shall be declared inoperable.

The second Note also requires that the methodologies for calculating the as-left and the as-found tolerances be in UFSAR, Section 7.1.2.

SR 3.3.1.8

SR 3.3.1.8 is the performance of a COT as described in SR 3.3.1.7, except it is modified by a Note stating that this test shall include verification that the P-6 and P-10 interlocks are in their required state for the existing unit condition. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable COT of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. The Frequency is modified by a Note that allows this surveillance to be satisfied if it has been performed within the Frequency specified in the Surveillance Frequency Control Program or of the Frequencies prior to reactor startup and after reducing power below P-10 and P-6. The Frequency of "prior to startup" ensures this surveillance is performed prior to critical operations and applies to the source, intermediate and power range low instrument channels. The Frequency of 12 hours after reducing power below P-10 (applicable to intermediate and power range low channels) and 4 hours after reducing

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## SURVEILLANCE REQUIREMENTS (continued)

power below P-6 (applicable to source range channels) allows a normal shutdown to be completed and the unit removed from the MODE of Applicability for this surveillance without a delay to perform the testing required by this surveillance. The Frequency thereafter applies if the plant remains in the MODE of Applicability after the initial performances of prior to reactor startup and 12 and four hours after reducing power below P-10 or P-6, respectively. The MODE of Applicability for this surveillance is < P-10 for the power range low and intermediate range channels and < P-6 for the source range channels. Once the unit is in MODE 3, this surveillance is no longer required. If power is to be maintained < P-10 for more than 12 hours or < P-6 for more than 4 hours, then the testing required by this surveillance must be performed prior to the expiration of the time limit. Twelve hours and four hours are reasonable times to complete the required testing or place the unit in a MODE where this surveillance is no longer required. This test ensures that the NIS source, intermediate, and power range low channels are OPERABLE prior to taking the reactor critical and after reducing power into the applicable MODE (< P-10 or < P-6) for periods > 12 and 4 hours, respectively.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.3.1.8 is modified by two Notes as identified in Table 3.3.1-1. The first Note requires evaluation of channel performance for the condition where the as-found setting for the channel setpoint is outside its as-found tolerance but conservative with respect to the Allowable Value. Evaluation of channel performance will verify that the channel will continue to behave in accordance with safety analysis assumptions and the channel performance assumptions in the setpoint methodology. The purpose of the assessment is to ensure confidence in the channel performance prior to returning the channel to service. For channels determined to be OPERABLE but degraded, after returning the channel to service the performance of these channels will be evaluated under the plant Corrective Action Program. Entry into the Corrective Action Program will ensure required review and documentation of the condition. The second Note requires that the as-left setting for the channel be returned to within the as-left tolerance of the NTSP. Where a setpoint more conservative than the NTSP is used in the plant surveillance procedures (field setting), the as-left and as-found tolerances, as applicable, will be applied to the surveillance procedure setpoint. This will ensure that sufficient margin to the Safety Limit and/or Analytical Limit is maintained. If the as-left channel setting cannot be returned to a setting within the as-left tolerance of the NTSP, then the channel shall be declared inoperable.

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### SURVEILLANCE REQUIREMENTS (continued)

The second Note also requires that the methodologies for calculating the as-left and the as-found tolerances be in UFSAR, Section 7.1.2.

#### SR 3.3.1.9

SR 3.3.1.9 is the performance of a TADOT.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable TADOT of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions.

The SR is modified by a Note that excludes verification of setpoints from the TADOT. Since this SR applies to RCP undervoltage and underfrequency relays, setpoint verification requires elaborate bench calibration and is accomplished during the CHANNEL CALIBRATION.

#### SR 3.3.1.10

CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy.

CHANNEL CALIBRATIONS must be performed consistent with the assumptions of the unit specific setpoint methodology. The difference between the current "as-found" values and NTSP must be consistent with the drift allowance used in the setpoint methodology.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.3.1.10 is modified by a Note stating that this test shall include verification that the time constants are adjusted to the prescribed values where applicable.

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## SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.10 is modified by two Notes as identified in Table 3.3.1-1. The first Note requires evaluation of channel performance for the condition where the as-found setting for the channel setpoint is outside its as-found tolerance but conservative with respect to the Allowable Value. Evaluation of channel performance will verify that the channel will continue to behave in accordance with safety analysis assumptions and the channel performance assumptions in the setpoint methodology. The purpose of the assessment is to ensure confidence in the channel performance prior to returning the channel to service. For channels determined to be OPERABLE but degraded, after returning the channel to service the performance of these channels will be evaluated under the plant Corrective Action Program. Entry into the Corrective Action Program will ensure required review and documentation of the condition. The second Note requires that the as-left setting for the channel be returned to within the as-left tolerance of the NTSP. Where a setpoint more conservative than the NTSP is used in the plant surveillance procedures (field setting), the as-left and as-found tolerances, as applicable, will be applied to the surveillance procedure setpoint. This will ensure that sufficient margin to the Safety Limit and/or Analytical Limit is maintained. If the as-left channel setting cannot be returned to a setting within the as-left tolerance of the NTSP, then the channel shall be declared inoperable.

The second Note also requires that the methodologies for calculating the as-left and the as-found tolerances be in UFSAR, Section 7.1.2.

SR 3.3.1.11

SR 3.3.1.11 is the performance of a CHANNEL CALIBRATION, as described in SR 3.3.1.10. This SR is modified by a Note stating that neutron detectors are excluded from the CHANNEL CALIBRATION. The CHANNEL CALIBRATION for the power range neutron detectors consists of a normalization of the detectors based on a power calorimetric and flux map performed above 15% RTP. The CHANNEL CALIBRATION for the source range consists of checking the discriminator voltage and adjusting if necessary. The CHANNEL CALIBRATION for the intermediate range neutron detectors consists of comparing the output of the intermediate range drawer to the secondary side calorimetric and adjusting if necessary. This Surveillance is not required for the NIS power range detectors for entry into MODE 2 or 1, and is not required for the NIS intermediate range detectors for entry into MODE 2, because the unit must be in at least MODE 2 to perform the test for the intermediate range detectors and MODE 1 for the power range detectors.

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### SURVEILLANCE REQUIREMENTS (continued)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.3.1.11 is modified by two Notes as identified in Table 3.3.1-1. The first Note requires evaluation of channel performance for the condition where the as-found setting for the channel setpoint is outside its as-found tolerance but conservative with respect to the Allowable Value. Evaluation of channel performance will verify that the channel will continue to behave in accordance with safety analysis assumptions and the channel performance assumptions in the setpoint methodology. The purpose of the assessment is to ensure confidence in the channel performance prior to returning the channel to service. For channels determined to be OPERABLE but degraded, after returning the channel to service the performance of these channels will be evaluated under the plant Corrective Action Program. Entry into the Corrective Action Program will ensure required review and documentation of the condition. The second Note requires that the as-left setting for the channel be returned to within the as-left tolerance of the NTSP. Where a setpoint more conservative than the NTSP is used in the plant surveillance procedures (field setting), the as-left and as-found tolerances, as applicable, will be applied to the surveillance procedure setpoint. This will ensure that sufficient margin to the Safety Limit and/or Analytical Limit is maintained. If the as-left channel setting cannot be returned to a setting within the as-left tolerance of the NTSP, then the channel shall be declared inoperable.

The second Note also requires that the methodologies for calculating the as-left and the as-found tolerances be in UFSAR, Section 7.1.2.

#### SR 3.3.1.12

SR 3.3.1.12 is the performance of a TADOT of the Manual Reactor Trip and the SI Input from ESFAS. The test shall independently verify the OPERABILITY of the undervoltage and shunt trip mechanisms for the Manual Reactor Trip Function for the Reactor Trip Breakers and Reactor Trip Bypass Breakers. The Reactor Trip Bypass Breaker test shall include testing of the automatic and manual undervoltage trip.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

The SR is modified by a Note that excludes verification of setpoints from the TADOT. The Functions affected have no setpoints associated with them.

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## SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.13

SR 3.3.1.13 is the performance of a TADOT of Turbine Trip Functions. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable TADOT of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. This TADOT is as described in SR 3.3.1.4, except that this test is performed prior to exceeding the P-9 interlock whenever the unit has been in MODE 3. This Surveillance is not required if it has been performed within the previous 31 days. Verification of the Trip Setpoint does not have to be performed for this Surveillance. Performance of this test will ensure that the turbine trip Function is OPERABLE prior to exceeding the P-9 interlock.

SR 3.3.1.14

SR 3.3.1.14 verifies that the individual channel/train actuation response times are less than or equal to the maximum values assumed in the accident analysis. Response time testing acceptance criteria are included in UFSAR Table 7.2.1-5. Individual component response times are not modeled in the analyses.

The analyses model the overall or total elapsed time, from the point at which the parameter exceeds the trip setpoint value at the sensor to the point at which the equipment reaches the required functional state (i.e., control and shutdown rods fully inserted in the reactor core).

For channels that include dynamic transfer Functions (e.g., lag, lead/lag, rate/lag, etc.), the response time test may be performed with the transfer Function set to one, with the resulting measured response time compared to the appropriate UFSAR response time. Alternately, the response time test can be performed with the time constants set to their nominal value, provided the required response time is analytically calculated assuming the time constants are set at their nominal values. The response time may be measured by a series of overlapping tests such that the entire response time is measured.

Response time may be verified by actual response time tests in any series of sequential, overlapping or total channel measurements, or by the summation of allocated sensor, signal processing and actuation logic response times with actual response time tests on the remainder of the

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### SURVEILLANCE REQUIREMENTS (continued)

channel. Allocations for sensor response times may be obtained from: (1) historical records based on acceptable response time tests (hydraulic, noise, or power interrupt tests), (2) in place, onsite, or offsite (e.g., vendor) test measurements, or (3) utilizing vendor engineering specifications. WCAP-13632-P-A, Revision 2, "Elimination of Pressure Sensor Response Time Testing Requirements," (Ref. 10) provides the basis and methodology for using allocated sensor response times in the overall verification of the channel response time for specific sensors identified in the WCAP. Response time verification for other sensor types must be demonstrated by test.

WCAP-14036-P, Revision 1, "Elimination of Periodic Protection Channel Response Time Tests," (Ref. 13) provides the basis and methodology for using allocated signal processing and actuation logic response times in the overall verification of the protection system channel response time. The allocations for sensor, signal conditioning, and actuation logic response times must be verified prior to placing the component in operational service and re-verified following maintenance that may adversely affect response time. In general, electrical repair work does not impact response time provided the parts used for repair are of the same type and value. Specific components identified in the WCAP may be replaced without verification testing. One example where response time could be affected is replacing the sensing assembly of a transmitter.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.3.1.14 is modified by a Note stating that neutron detectors are excluded from RTS RESPONSE TIME testing. This Note is necessary because of the difficulty in generating an appropriate detector input signal. Excluding the detectors is acceptable because the principles of detector operation ensure a virtually instantaneous response.

BASES

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- REFERENCES
1. Regulatory Guide 1.105, Revision 3, "Setpoints for Safety Related Instrumentation."
  2. UFSAR, Chapter 7.
  3. UFSAR, Chapter 6.
  4. UFSAR, Chapter 15.
  5. IEEE-279-1971.
  6. 10 CFR 50.49.
  7. Calculation SQN-EEB-PL&S, Precautions, Limitations, and Setpoints for NSSS.
  8. WCAP-14333-P-A, Rev. 1, October 1998.
  9. WCAP-10271-P-A, Supplement 1, May 1986.
  10. WCAP-13632-P-A, Revision 2, "Elimination of Pressure Sensor Response Time Testing Requirements," January 1996.
  11. WCAP-10271-P-A, Supplement 2, June 1990.
  12. WCAP-15376, Rev. 0, October 2000.
  13. WCAP-14036-P, Revision 1, "Elimination of Periodic Protection Channel Response Time Tests," December 1995.
  14. Letter from Siva P. Lingam (NRC) to Joseph W. Shea (TVA), "Sequoyah Nuclear Plant, Units 1 and 2 - Issuance of Amendments to Revise the Technical Specification to allow use of Areva Advanced W17 High Performance Fuel (TS-SQN-2011-07) (TAC NOS. ME6538 and ME6539)," dated September 26, 2012.
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## B 3.3 INSTRUMENTATION

### B 3.3.2 Engineered Safety Feature Actuation System (ESFAS) Instrumentation

#### BASES

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**BACKGROUND** The ESFAS initiates necessary safety systems, based on the values of selected unit parameters, to protect against violating core design limits and the Reactor Coolant System (RCS) pressure boundary, and to mitigate accidents. This is achieved by specifying limiting safety system settings (LSSS) in terms of parameters directly monitored by the ESFAS, as well as specifying LCOs on other reactor system parameters and equipment performance.

Technical Specifications are required by 10 CFR 50.36 to include LSSS. LSSS are defined by the regulation as settings for automatic protective devices related to those variables having significant safety functions. The regulation also states, "Where a LSSS is specified for a variable on which a safety limit has been placed, the setting must be chosen so that automatic protective actions will correct the abnormal situation before a Safety Limit (SL) is exceeded." The Analytical Limit is the limit of the process variable at which a protective action is initiated, as established by the safety analysis, to ensure that a SL is not exceeded. Any automatic protection action that occurs on reaching the Analytical Limit therefore ensures that the SL is not exceeded. However, in practice, the actual settings for automatic protection channels must be chosen to be more conservative than the Analytical Limit to account for instrument loop uncertainties related to the setting at which the automatic protective action would actually occur.

The Nominal Trip Setpoint (NTSP) specified in Table 3.3.2-1 is a predetermined setting for a protection channel chosen to ensure automatic actuation prior to the process variable reaching the Analytical Limit and thus ensuring that the SL would not be exceeded. As such, the NTSP accounts for uncertainties in setting the channel (e.g., calibration), uncertainties in how the channel might actually perform (e.g., repeatability), changes in the point of action of the channel over time (e.g., drift during surveillance intervals), and any other factors which may influence its actual performance (e.g., harsh accident environments). In this manner, the NTSP ensures that SLs are not exceeded. Therefore, the NTSP meets the definition of an LSSS (Ref. 1).

Technical Specifications contain values related to the OPERABILITY of equipment required for safe operation of the facility. OPERABLE is defined in Technical Specifications as "...being capable of performing its safety functions(s)." Relying solely on the NTSP to define OPERABILITY in Technical Specifications would be an overly restrictive requirement if it were applied as an OPERABILITY limit for the "as-found" value of a protection channel setting during a surveillance. This would result in

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### BACKGROUND (continued)

Technical Specification compliance problems, as well as reports and corrective actions required by the rule which are not necessary to ensure safety. For example, an automatic protection channel with a setting that has been found to be different from the NTSP due to some drift of the setting may still be OPERABLE since drift is to be expected. This expected drift would have been specifically accounted for in the setpoint methodology for calculating the NTSP and thus the automatic protective action would still have ensured that the SL would not be exceeded with the "as-found" setting of the protection channel. Therefore, the channel would still be OPERABLE since it would have performed its safety function and the only corrective action required would be to reset the channel within the established as-left tolerance around the NTSP to account for further drift during the next surveillance interval.

During Anticipated Operational Occurrences (AOOs), which are those events expected to occur one or more times during the unit life, the acceptable limits are:

1. The Departure from Nucleate Boiling Ratio (DNBR) shall be maintained above the SL value to prevent departure from nucleate boiling (DNB),
2. Fuel centerline melt shall not occur, and
3. The RCS pressure SL of 2735 psig shall not be exceeded.

Operation within the SLs of Specification 2.0, "Safety Limits (SLs)," also maintains the above values and assures that offsite dose will be within the 10 CFR 50 and 10 CFR 100 criteria during AOOs.

Accidents are events that are analyzed even though they are not expected to occur during the unit life. The acceptable limit during accidents is that offsite dose shall be maintained within an acceptable fraction of 10 CFR 100 limits. Different accident categories are allowed a different fraction of these limits, based on probability of occurrence. Meeting the acceptable dose limit for an accident category is considered having acceptable consequences for that event.

The ESFAS instrumentation is segmented into three distinct but interconnected modules as identified below:

- Field transmitters or process sensors and instrumentation: provide a measurable electronic signal based on the physical characteristics of the parameter being measured,

## BASES

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### BACKGROUND (continued)

- Signal processing equipment including Process protection system, field contacts, and protection channel sets: provide signal conditioning, setpoint comparison, process algorithm actuation, compatible electrical signal output to protection system channels, and control board/control room/miscellaneous indications, and
- Solid State Protection System (SSPS) including input, logic, and output bays: initiates the proper unit shutdown or engineered safety feature (ESF) actuation in accordance with the defined logic and based on the bistable, setpoint comparator, or contact outputs from the signal process control and protection system.

#### Field Transmitters or Sensors

To meet the design demands for redundancy and reliability, more than one, and often as many as four, field transmitters or sensors are used to measure unit parameters. In many cases, field transmitters or sensors that input to the ESFAS are shared with the Reactor Trip System (RTS). In some cases, the same channels also provide control system inputs. To account for calibration tolerances and instrument drift, which are assumed to occur between calibrations, statistical allowances are provided in the NTSP and Allowable Value. The OPERABILITY of each transmitter or sensor is determined by either "as-found" calibration data evaluated during the CHANNEL CALIBRATION or by qualitative assessment of field transmitter or sensor, as related to the channel behavior observed during performance of the CHANNEL CHECK.

#### Signal Processing Equipment

Generally, three or four channels of process control equipment are used for the signal processing of unit parameters measured by the field instruments. The process control equipment provides analog to digital conversion (Digital Protection System), signal conditioning, comparable output signals for instruments located on the main control board, and comparison of measured input signals with NTSPs derived from Analytical Limits established by the safety analyses. Analytical Limits are defined in UFSAR, Chapter 6 (Ref. 2), Chapter 7 (Ref. 3), and Chapter 15 (Ref. 4). If the measured value of a unit parameter exceeds the predetermined setpoint, an output from a bistable, setpoint comparator, or contact is forwarded to the SSPS for decision evaluation. Channel separation is maintained up to and through the input bays. However, not all unit parameters require four channels of sensor measurement and signal processing. Some unit parameters provide input only to the SSPS, while others provide input to the SSPS, and one or more control systems.

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### BACKGROUND (continued)

Generally, if a parameter is used only for input to the protection circuits, three channels with a two-out-of-three logic are sufficient to provide the required reliability and redundancy. If one channel fails in a direction that would not result in a partial Function trip, the Function is still OPERABLE with a two-out-of-two logic. If one channel fails such that a partial Function trip occurs, a trip will not occur and the Function is still OPERABLE with a one-out-of-two logic.

Generally, if a parameter is used for input to the SSPS and a control function, four channels with a two-out-of-four logic are sufficient to provide the required reliability and redundancy. The circuit must be able to withstand both an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation. Again, a single failure will neither cause nor prevent the protection function actuation.

These requirements are described in IEEE-279-1971 (Ref. 5). The actual number of channels required for each unit parameter is specified in Reference 3.

#### NTSPs and ESFAS Setpoints Allowable Values

The trip setpoints used in the bistables, setpoint comparators, or contacts are based on the analytical limits stated in Reference 3. The calculation of the NTSPs specified in Table 3.3.2-1 is such that adequate protection is provided when all sensor and processing time delays are taken into account. To allow for calibration tolerances, instrumentation uncertainties, instrument drift, and severe environment errors for those ESFAS channels that must function in harsh environments as defined by 10 CFR 50.49 (Ref. 6), the Allowable Values specified in Table 3.3.2-1 in the accompanying LCO are conservative with respect to the analytical limits. A detailed description of the methodology used to calculate the Allowable Values and ESFAS NTSPs including their explicit uncertainties, is provided in the plant specific setpoint methodology study (Ref. 7) which incorporates all of the known uncertainties applicable to each channel. The as-left tolerance and as-found tolerance band methodology is provided in UFSAR, Section 7.1.2. The magnitudes of these uncertainties are factored into the determination of each ESFAS NTSP and corresponding Allowable Value. The nominal ESFAS setpoint entered is more conservative than that specified by the Allowable Value to account for measurement errors detectable by the CHANNEL OPERATIONAL TEST (COT). The Allowable Value serves as the as-found Technical Specification OPERABILITY limit for the purpose of the COT.

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### BACKGROUND (continued)

The NTSP is the value at which the bistables or setpoint comparators are set and is the expected value to be achieved during calibration. The NTSP value is the LSSS and ensures the safety analysis limits are met for the surveillance interval selected when a channel is adjusted based on stated channel uncertainties. Any bistable or setpoint comparator is considered to be properly adjusted when the "as-left" NTSP value is within the as-left tolerance for CHANNEL CALIBRATION uncertainty allowance (i.e., + rack calibration and comparator setting uncertainties). The NTSP value is therefore considered a "nominal value" (i.e., expressed as a value without inequalities) for the purposes of the COT and CHANNEL CALIBRATION.

Nominal Trip Setpoints, in conjunction with the use of as-found and as-left tolerances together with the requirements of the Allowable Value ensure that the consequences of Design Basis Accidents (DBAs) will be acceptable, providing the unit is operated from within the LCOs at the onset of the DBA and the equipment functions as designed.

Note that the Allowable Values listed in Table 3.3.2-1 are the least conservative value of the as-found setpoint that a channel can have during a periodic CHANNEL CALIBRATION, COT, or a TADOT.

Each channel can be tested on line to verify that the signal processing equipment and setpoint accuracy is within the specified allowance requirements of Reference 3. Once a designated channel is taken out of service for testing, a simulated signal is injected in place of the field instrument signal. The process equipment for the channel in test is then tested, verified, and calibrated. SRs for the channels are specified in the SR section.

#### Solid State Protection System

The SSPS equipment is used for the decision logic processing of outputs from the signal processing equipment bistables, setpoint comparators, or contacts. To meet the redundancy requirements, two trains of SSPS, each performing the same functions, are provided. If one train is taken out of service for maintenance or test purposes, the second train will provide ESF actuation for the unit. If both trains are taken out of service or placed in test, a reactor trip will result. Each train is packaged in its own cabinet for physical and electrical separation to satisfy separation and independence requirements.

The SSPS performs the decision logic for most ESF equipment actuation; generates the electrical output signals that initiate the required actuation; and provides the status, permissive, and annunciator output signals to the main control room of the unit.

## BASES

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### BACKGROUND (continued)

The bistable, setpoint comparator, or contact outputs from the signal processing equipment are sensed by the SSPS equipment and combined into logic matrices that represent combinations indicative of various transients. If a required logic matrix combination is completed, the system will send actuation signals via master and slave relays to those components whose aggregate Function best serves to alleviate the condition and restore the unit to a safe condition. Examples are given in the Applicable Safety Analyses, LCO, and Applicability sections of this Bases.

Each SSPS train has a built in testing device that can automatically test the decision logic matrix functions and the actuation channels while the unit is at power. When any one train is taken out of service for testing, the other train is capable of providing unit monitoring and protection until the testing has been completed. The testing device is semiautomatic to minimize testing time.

The actuation of ESF components is accomplished through master and slave relays. The SSPS energizes the master relays appropriate for the condition of the unit. Each master relay then energizes one or more slave relays, which then cause actuation of the end devices. The master and slave relays are routinely tested to ensure operation. The test of the master relays energizes the relay, which then operates the contacts and applies a low voltage to the associated slave relays. The low voltage is not sufficient to actuate the slave relays but only demonstrates signal path continuity. The SLAVE RELAY TEST actuates the devices if their operation will not interfere with continued unit operation. For the latter case, actual component operation is prevented by the SLAVE RELAY TEST circuit, and slave relay contact operation is verified by a continuity check of the circuit containing the slave relay.

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### APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

Each of the analyzed accidents can be detected by one or more ESFAS Functions. One of the ESFAS Functions is the primary actuation signal for that accident. An ESFAS Function may be the primary actuation signal for more than one type of accident. An ESFAS Function may also be a secondary, or backup, actuation signal for one or more other accidents. For example, Pressurizer Pressure - Low is a primary actuation signal for small loss of coolant accidents (LOCAs) and a backup actuation signal for steam line breaks (SLBs) outside containment.

Functions such as manual initiation, not specifically credited in the accident safety analysis, are implicitly credited in the safety analysis and the NRC staff approved licensing basis for the unit. These Functions may provide protection for conditions that do not require dynamic transient analysis to demonstrate Function performance. These Functions may

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### APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

also serve as backups to Functions that were credited in the accident analysis (Ref. 4).

Permissive and interlock setpoints allow the blocking of trips during plant startups, and restoration of trips when the permissive conditions are not satisfied, but they are not explicitly modeled in the Safety Analyses. These permissives and interlocks ensure that the starting conditions are consistent with the safety analysis, before preventive or mitigating actions occur. Because these permissives or interlocks are only one of multiple conservative starting assumptions for the accident analysis, they are generally considered as nominal values without regard to measurement accuracy.

The LCO requires all instrumentation performing an ESFAS Function, listed in Table 3.3.2-1 in the accompanying LCO, to be OPERABLE. The Allowable Value specified in Table 3.3.2-1 is the least conservative value of the as-found setpoint that the channel can have when tested, such that a channel is OPERABLE if the as-found setpoint is within the as-found tolerance and is conservative with respect to the Allowable Value during the CHANNEL CALIBRATION or COT. As such, the Allowable Value differs from the NTSP by an amount greater than or equal to the expected instrument channel uncertainties, such as drift, during the surveillance interval. In this manner, the actual setting of the channel NTSP will ensure that a SL is not exceeded at any given point of time as long as the channel has not drifted beyond expected tolerances during the surveillance interval. Note that, although the channel is OPERABLE under these circumstances, the trip setpoint must be left adjusted to a value within the as-left tolerance, in accordance with uncertainty assumptions stated in the referenced setpoint methodology (as-left criteria), and confirmed to be operating within the statistical allowances of the uncertainty terms assigned (as-found criteria).

If the actual setting of the channel is found to be conservative with respect to the Allowable Value but is beyond the as-found tolerance band, the channel is OPERABLE, but degraded. The degraded condition of the channel will be evaluated during performance of the SR. This evaluation will consist of resetting the channel setpoint to the NTSP (within the allowed tolerance) and evaluating the channel response. If the channel is functioning as required and expected to pass the next surveillance, then the channel can be restored to service at the completion of the surveillance.

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### APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

A trip setpoint may be set more conservative than the NTSP as necessary in response to plant conditions. However, in this case, the OPERABILITY of this instrument must be verified based on the field setting and not the NTSP. Failure of any instrument renders the affected channel(s) inoperable and reduces the reliability of the affected Functions.

The LCO generally requires OPERABILITY of four or three channels in each instrumentation function and two channels in each logic and manual initiation function. The two-out-of-three and the two-out-of-four configurations allow one channel to be tripped during maintenance or testing without causing an ESFAS initiation. Two logic or manual initiation channels are required to ensure no single random failure disables the ESFAS.

The required channels of ESFAS instrumentation provide unit protection in the event of any of the analyzed accidents. ESFAS protection functions are as follows:

#### 1. Safety Injection

Safety Injection (SI) provides two primary functions:

1. Primary side water addition to ensure maintenance or recovery of reactor vessel water level (coverage of the active fuel for heat removal, clad integrity, and for limiting peak clad temperature to < 2200°F), and
2. Boration to ensure recovery and maintenance of SDM ( $k_{\text{eff}} < 1.0$ ).

These functions are necessary to mitigate the effects of high energy line breaks (HELBs) both inside and outside of containment. The SI signal is also used to initiate other Functions such as:

- Phase A Isolation,
- Containment Ventilation Isolation,
- Reactor Trip,
- ERCW and CCS Pump Start and System Isolation,
- Turbine Trip,
- Feedwater Isolation,

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### APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

- Start of motor driven auxiliary feedwater (AFW) pumps,
- Control room ventilation isolation, and
- Enabling automatic switchover of Emergency Core Cooling Systems (ECCS) suction to containment sump.

These other functions ensure:

- Isolation of nonessential systems through containment penetrations,
- Trip of the turbine and reactor to limit power generation,
- Isolation of main feedwater (MFW) to limit secondary side mass losses,
- Start of AFW to ensure secondary side cooling capability,
- Isolation of the control room to ensure habitability, and
- Enabling ECCS suction from the refueling water storage tank (RWST) switchover on low low RWST level to ensure continued cooling via use of the containment sump.

a. Safety Injection - Manual Initiation

The LCO requires one channel per train to be OPERABLE. The operator can initiate SI at any time by using either of two switches in the control room. This action will cause actuation of all components in the same manner as any of the automatic actuation signals.

The LCO for the Manual Initiation Function ensures the proper amount of redundancy is maintained in the manual ESFAS actuation circuitry to ensure the operator has manual ESFAS initiation capability.

Each channel consists of one hand switch and the interconnecting wiring to the actuation logic cabinet. Each hand switch actuates both trains. This configuration does not allow testing at power.

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

b. Safety Injection - Automatic Actuation Logic and Actuation Relays

This LCO requires two trains to be OPERABLE. Actuation logic consists of all circuitry housed within the actuation subsystems, including the initiating relay contacts responsible for actuating the ESF equipment. The two trains are redundant such that only one is necessary to perform the ESFAS Function.

Manual and automatic initiation of SI must be OPERABLE in MODES 1, 2, and 3. In these MODES, there is sufficient energy in the primary and secondary systems to warrant automatic initiation of ESF systems. Manual Initiation is also required in MODE 4 even though automatic actuation is not required. In this MODE, adequate time is available to manually actuate required components in the event of an abnormal condition or accident, but because of the large number of components actuated on a SI, actuation is simplified by the use of the manual actuation hand switches. Automatic actuation logic and actuation relays must be OPERABLE in MODE 4 to support system level manual initiation.

These Functions are not required to be OPERABLE in MODES 5 and 6 because there is adequate time for the operator to evaluate unit conditions and respond by manually starting individual systems, pumps, and other equipment to mitigate the consequences of an abnormal condition or accident. Unit pressure and temperature are very low and many ESF components are administratively locked out or otherwise prevented from actuating to prevent inadvertent overpressurization of unit systems.

c. Safety Injection - Containment Pressure - High

This signal provides protection against the following accidents:

- SLB inside containment,
- LOCA, and
- Feed line break inside containment.

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Containment Pressure - High provides no input to any control functions. Thus, three OPERABLE channels are sufficient to satisfy protective requirements with a two-out-of-three logic. The transmitters and electronics are located inside the containment annulus, but outside containment, and experience more adverse environmental conditions than if they were located outside containment altogether. However, the environmental effects are less severe than if the transmitters were located inside containment. The NTSP reflects the inclusion of both steady state instrument uncertainties and slightly more adverse environmental instrument uncertainties.

Containment Pressure - High must be OPERABLE in MODES 1, 2, and 3 when there is sufficient energy in the primary and secondary systems to pressurize the containment following a pipe break. In MODES 4, 5, and 6, there is insufficient energy in the primary or secondary systems to pressurize the containment.

d. Safety Injection - Pressurizer Pressure – Low

This signal provides protection against the following accidents:

- Inadvertent opening of a steam generator (SG) relief or safety valve,
- SLB,
- A spectrum of rod cluster control assembly ejection accidents (rod ejection),
- Inadvertent opening of a pressurizer relief or safety valve,
- LOCAs, and
- SG Tube Rupture.

Three protection channels are necessary to satisfy the protective requirements with a two-out-of-three logic.

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### APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The transmitters are located inside containment, with the taps in the vapor space region of the pressurizer, and thus possibly experiencing adverse environmental conditions (LOCA, SLB inside containment, rod ejection). Therefore, the NTSP reflects the inclusion of both steady state and adverse environmental instrument uncertainties.

This Function must be OPERABLE in MODES 1, 2, and 3 (above P-11) to mitigate the consequences of an HELB inside containment. This signal may be manually blocked by the operator below the P-11 setpoint. Automatic SI actuation below this pressure setpoint is then performed by the Containment Pressure - High signal.

This Function is not required to be OPERABLE in MODE 3 below the P-11 setpoint. Other ESF functions are used to detect accident conditions and actuate the ESF systems in this MODE. In MODES 4, 5, and 6, this Function is not needed for accident detection and mitigation.

e. Safety Injection - Steam Line Pressure

(1) Steam Line Pressure – Low

Steam Line Pressure - Low provides protection against the following accidents:

- SLB,
- Feed line break, and
- Inadvertent opening of an SG relief or an SG safety valve.

Steam Line Pressure - Low provides no input to any control functions. Thus, three OPERABLE channels on each steam line are sufficient to satisfy the protective requirements with a two-out-of-three logic on each steam line.

With the transmitters typically located inside the steam valve vaults, it is possible for them to experience adverse environmental conditions during a secondary side break. Therefore, the NTSP reflects both steady state and adverse environmental instrument uncertainties.

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### APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

This Function is anticipatory in nature and has a lead/lag ratio of approximately 50/5.

Steam Line Pressure - Low must be OPERABLE in MODES 1, 2, and 3 (above P-11) when a secondary side break or stuck open valve could result in the rapid depressurization of the steam lines. This signal may be manually blocked by the operator below the P-11 setpoint. Below P-11, feed line break is not a concern. Inside containment SLB will be terminated by automatic SI actuation via Containment Pressure - High, and outside containment SLB will be terminated by the Steam Line Pressure - Negative Rate - High signal for steam line isolation. This Function is not required to be OPERABLE in MODE 4, 5, or 6 because there is insufficient energy in the secondary side of the unit to cause an accident.

#### 2. Containment Spray

Containment Spray functions to lower containment pressure and temperature after an HELB in containment.

This function is necessary to ensure the pressure boundary integrity of the containment structure.

The containment spray actuation signal starts the containment spray pumps and aligns the discharge of the pumps to the containment spray nozzle headers in the upper levels of containment. Water is initially drawn from the RWST by the containment spray pumps. When the RWST reaches the low low level setpoint, the spray pump suctions are shifted to the containment sump if continued containment spray is required. Containment spray is actuated manually or automatically by Containment Pressure – High - High.

##### a. Containment Spray - Manual Initiation

The operator can initiate containment spray at any time from the control room by simultaneously turning two Phase B & Containment Ventilation Isolation switches in the same train. Because an inadvertent actuation of containment spray could have such serious consequences, two switches must be turned simultaneously to initiate containment spray. There are two sets of two switches each in the control room. Simultaneously turning the two switches in either set will actuate containment spray in both trains in the same manner as the automatic actuation

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### APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

signal. Two Manual Initiation switches in each train are required to be OPERABLE to ensure no single failure disables the Manual Initiation Function. Note that Manual Initiation of containment spray also actuates Phase B containment isolation but does not close the Main Steam Isolation Valves.

b. Containment Spray - Automatic Actuation Logic and Actuation Relays

Automatic actuation logic and actuation relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b.

Manual and automatic initiation of containment spray must be OPERABLE in MODES 1, 2, and 3 when there is a potential for an accident to occur, and sufficient energy in the primary or secondary systems to pose a threat to containment integrity due to overpressure conditions. Manual initiation is also required in MODE 4, even though automatic actuation is not required. In this MODE, adequate time is available to manually actuate required components in the event of a DBA. However, because of the large number of components actuated on a containment spray, actuation is simplified by the use of the manual actuation switches. Automatic actuation logic and actuation relays must be OPERABLE in MODE 4 to support system level manual initiation. In MODES 5 and 6, there is insufficient energy in the primary and secondary systems to result in containment overpressure. In MODES 5 and 6, there is also adequate time for the operators to evaluate unit conditions and respond, to mitigate the consequences of abnormal conditions by manually starting individual components.

c. Containment Spray - Containment Pressure

This signal provides protection against a LOCA or a SLB inside containment. The transmitters (d/p cells) are located outside of containment with the sensing line (high pressure side of the transmitter) located inside containment. The transmitters and electronics are located inside the containment annulus, but outside containment, and experience more adverse environmental conditions than if they were located outside containment altogether. However, the environmental effects are less severe than if the transmitters were located inside containment. The NTSP reflects the inclusion of both steady

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### APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

state instrument uncertainties and slightly more adverse environmental instrument uncertainties.

This is one of the only Functions that requires the output to energize to perform its required action. It is not desirable to have a loss of power actuate containment spray, since the consequences of an inadvertent actuation of containment spray could be serious. Note that this Function also has the inoperable channel placed in bypass rather than trip to decrease the probability of an inadvertent actuation.

This function uses four channels in a two-out-of-four logic configuration. This arrangement exceeds the minimum redundancy requirements. Additional redundancy is warranted because this Function is energized to trip. Containment Pressure – High - High must be OPERABLE in MODES 1, 2, and 3 when there is sufficient energy in the primary and secondary sides to pressurize the containment following a pipe break. In MODES 4, 5, and 6, there is insufficient energy in the primary and secondary sides to pressurize the containment and reach the Containment Pressure - High - High setpoint.

#### 3. Containment Isolation

Containment Isolation provides isolation of the containment atmosphere, and all process systems that penetrate containment, from the environment. This Function is necessary to prevent or limit the release of radioactivity to the environment in the event of a large break LOCA.

There are two separate Containment Isolation signals, Phase A and Phase B. Phase A isolation isolates all automatically isolable process lines, except component cooling water, essential raw cooling water, and control air, at a relatively low containment pressure indicative of primary or secondary system leaks. For these types of events, forced circulation cooling using the reactor coolant pumps (RCPs) and SGs is the preferred (but not required) method of decay heat removal. Since component cooling water is required to support RCP operation, not isolating component cooling water on the Phase A signal enhances unit safety by allowing operators to use forced RCS circulation to cool the unit. Isolating component cooling water on the Phase A signal may force the use of feed and bleed cooling, which could prove more difficult to control.

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### APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Phase A containment isolation is actuated automatically by SI, or manually via the automatic actuation logic. All process lines penetrating containment, with the exception of component cooling water, essential raw cooling water, and control air, are isolated.

Component cooling water is not isolated at this time to permit continued operation of the RCPs with cooling water flow to the thermal barrier heat exchangers and air or oil coolers. All process lines not equipped with remote operated isolation valves are manually closed, or otherwise isolated, prior to reaching MODE 4.

Manual Phase A Containment Isolation is accomplished by either of two switches in the control room. Either switch actuates both trains. Note that manual actuation of Phase A Containment Isolation also actuates Containment Ventilation Isolation.

The Phase B signal isolates component cooling water, essential raw cooling water, and control air. This occurs at a relatively high containment pressure that is indicative of a large break LOCA or a SLB. For these events, forced circulation using the RCPs is no longer desirable. Isolating the component cooling water at the higher pressure does not pose a challenge to the containment boundary because the Component Cooling Water System is a closed loop inside containment. Although some system components do not meet all of the ASME Code requirements applied to the containment itself, the system is continuously pressurized to a pressure greater than the Phase B setpoint. Thus, routine operation demonstrates the integrity of the system pressure boundary for pressures exceeding the Phase B setpoint. Furthermore, because system pressure exceeds the Phase B setpoint, any system leakage prior to initiation of Phase B isolation would be into containment. Therefore, the combination of Component Cooling Water System design and Phase B isolation ensures the Component Cooling Water System is not a potential path for radioactive release from containment.

Phase B containment isolation is actuated by Containment Pressure – High - High, or manually, via the automatic actuation logic, as previously discussed. For containment pressure to reach a value high enough to actuate Containment Pressure – High - High, a large break LOCA or SLB must have occurred. RCP operation will no longer be required and component cooling water to the RCPs is, therefore, no longer necessary. The RCPs can be operated with seal injection flow alone and without component cooling water flow to the thermal barrier heat exchanger.

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### APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Manual Phase B Containment Isolation is accomplished by the same switches that actuate Containment Spray. When the two switches in either set are turned simultaneously, Phase B Containment Isolation and Containment Spray will be actuated in both trains.

a. Containment Isolation - Phase A Isolation

(1) Phase A Isolation - Manual Initiation

Manual Phase A Containment Isolation is actuated by either of two switches in the control room. Either switch actuates both trains. Note that manual initiation of Phase A Containment Isolation also actuates Containment Ventilation Isolation.

(2) Phase A Isolation - Automatic Actuation Logic and Actuation Relays

Automatic Actuation Logic and Actuation Relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b.

Manual and automatic initiation of Phase A Containment Isolation must be OPERABLE in MODES 1, 2, and 3, when there is a potential for an accident to occur. Manual initiation is also required in MODE 4 even though automatic actuation is not required. In this MODE, adequate time is available to manually actuate required components in the event of an accident, but because of the large number of components actuated on a Phase A Containment Isolation, actuation is simplified by the use of the manual actuation switches. Automatic actuation logic and actuation relays must be OPERABLE in MODE 4 to support system level manual initiation. In MODES 5 and 6, there is insufficient energy in the primary or secondary systems to pressurize the containment to require Phase A Containment Isolation. There also is adequate time for the operator to evaluate unit conditions and manually actuate individual isolation valves in response to abnormal or accident conditions.

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

(3) Phase A Isolation - Safety Injection

Phase A Containment Isolation is also initiated by all Functions that initiate SI. The Phase A Containment Isolation requirements for these Functions are the same as the requirements for their SI function. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead, Function 1, SI, is referenced for all initiating Functions and requirements.

b. Containment Isolation - Phase B Isolation

Phase B Containment Isolation is accomplished by Manual Initiation, Automatic Actuation Logic and Actuation Relays, and by Containment Pressure channels (the same channels that actuate Containment Spray, Function 2). The Containment Pressure trip of Phase B Containment Isolation is energized to trip in order to minimize the potential of spurious trips that may damage the RCPs.

(1) Phase B Isolation - Manual Initiation

The operator can initiate Phase B containment isolation at any time from the control room by simultaneously turning two Phase B & Containment Ventilation Isolation switches in the same train. There are two sets of two switches each in the control room. Simultaneously turning the two switches in either set will actuate Phase B containment isolation in both trains in the same manner as the automatic actuation signal. Two Manual Initiation switches in each train are required to be OPERABLE to ensure no single failure disables the Manual Initiation Function. Note that Manual Initiation of Phase B containment isolation also actuates containment spray and containment vent isolation.

(2) Phase B Isolation - Automatic Actuation Logic and Actuation Relays

Manual and automatic initiation of Phase B containment isolation must be OPERABLE in MODES 1, 2, and 3, when there is a potential for an accident to occur. Manual initiation is also required in MODE 4 even though automatic actuation is not required. In this MODE, adequate time is available to manually actuate required components in the event of an accident. However, because of the large number of components actuated on a Phase B containment

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

isolation, actuation is simplified by the use of the manual actuation hand switches. Automatic actuation logic and actuation relays must be OPERABLE in MODE 4 to support system level manual initiation. In MODES 5 and 6, there is insufficient energy in the primary or secondary systems to pressurize the containment to require Phase B containment isolation. There also is adequate time for the operator to evaluate unit conditions and manually actuate individual isolation valves in response to abnormal or accident conditions.

(3) Phase B Isolation - Containment Pressure

The basis for containment pressure MODE applicability is as discussed for ESFAS Function 2.c above.

4. Steam Line Isolation

Isolation of the main steam lines provides protection in the event of a SLB inside or outside containment. Rapid isolation of the steam lines will limit the steam break accident to the blowdown from one SG, at most. For a SLB upstream of the main steam isolation valves (MSIVs), inside or outside of containment, closure of the MSIVs limits the accident to the blowdown from only the affected SG. For a SLB downstream of the MSIVs, closure of the MSIVs terminates the accident as soon as the steam lines depressurize. Steam Line Isolation also mitigates the effects of a feed line break and ensures a source of steam for the turbine driven AFW pump during a feed line break.

a. Steam Line Isolation - Manual Initiation

Manual initiation of Steam Line Isolation can be accomplished from the control room. There are four switches in the control room and each switch initiates action to immediately close its associated MSIV. The LCO requires one channel per steam line to be OPERABLE.

b. Steam Line Isolation - Automatic Actuation Logic and Actuation Relays

Automatic actuation logic and actuation relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b.

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Manual and automatic initiation of steam line isolation must be OPERABLE in MODES 1, 2, and 3 when there is sufficient energy in the RCS and SGs to have a SLB or other accident. This could result in the release of significant quantities of energy and cause a cooldown of the primary system. The Steam Line Isolation Function is required in MODES 2 and 3 unless all MSIVs are closed. In MODES 4, 5, and 6, there is insufficient energy in the RCS and SGs to experience a SLB or other accident releasing significant quantities of energy.

c. Steam Line Isolation - Containment Pressure - High - High

This Function actuates closure of the MSIVs in the event of a LOCA or a SLB inside containment to maintain at least one unfaulted SG as a heat sink for the reactor, and to limit the mass and energy release to containment. The transmitters (d/p cells) are located outside containment with the sensing line (high pressure side of the transmitter) located inside containment. Containment Pressure – High - High provides no input to any control functions. Thus, three OPERABLE channels are sufficient to satisfy protective requirements with two-out-of-three logic. However, for enhanced reliability, this Function was designed with four channels and a two-out-of-four logic.

The transmitters and electronics are located inside the containment annulus, but outside containment, and experience more adverse environmental conditions than if they were located outside containment altogether. However, the environmental effects are less severe than if the transmitters were located inside containment. The NTSP reflects the inclusion of both steady state instrument uncertainties and slightly more adverse environmental instrument uncertainties.

Containment Pressure – High - High must be OPERABLE in MODES 1, 2, and 3, when there is sufficient energy in the primary and secondary side to pressurize the containment following a pipe break. This would cause a significant increase in the containment pressure, thus allowing detection and closure of the MSIVs. The Steam Line Isolation Function remains OPERABLE in MODES 2 and 3 unless all MSIVs are closed. In MODES 4, 5, and 6, there is not enough energy in the primary and secondary sides to pressurize the containment to the Containment Pressure – High - High setpoint.

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

d. Steam Line Isolation - Steam Line Pressure

(1) Steam Line Pressure – Low

Steam Line Pressure - Low provides closure of the MSIVs in the event of a SLB to maintain at least one unfaulted SG as a heat sink for the reactor, and to limit the mass and energy release to containment. This Function provides closure of the MSIVs in the event of a feed line break to ensure a supply of steam for the turbine driven AFW pump. Steam Line Pressure - Low was discussed previously under SI Function 1.e.1.

Steam Line Pressure - Low Function must be OPERABLE in MODES 1, 2, and 3 (when the Steam Line Isolation on Steam Line Pressure, Negative Rate-High is blocked), with any main steam valve open, when a secondary side break or stuck open valve could result in the rapid depressurization of the steam lines. This signal may be manually blocked by the operator below the P-11 setpoint. Below P-11, an inside containment SLB will be terminated by automatic actuation via Containment Pressure - High - High. Stuck valve transients and outside containment SLBs will be terminated by the Steam Line Pressure - Negative Rate - High signal for Steam Line Isolation below P-11 when SI has been manually blocked. The Steam Line Isolation Function is required in MODES 2 and 3 unless all MSIVs are closed. This Function is not required to be OPERABLE in MODES 4, 5, and 6 because there is insufficient energy in the secondary side of the unit to have an accident.

(2) Steam Line Pressure - Negative Rate – High

Steam Line Pressure - Negative Rate - High provides closure of the MSIVs for a SLB when less than the P-11 setpoint, to maintain at least one unfaulted SG as a heat sink for the reactor, and to limit the mass and energy release to containment. When the operator manually blocks the Steam Line Pressure - Low main steam isolation signal when less than the P-11 setpoint, the Steam Line Pressure - Negative Rate - High signal is automatically enabled. Steam Line Pressure - Negative Rate - High provides no input to any control functions. Thus, three OPERABLE channels are sufficient to satisfy requirements with a two-out-of-three logic on each steam line.

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### APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Steam Line Pressure - Negative Rate - High must be OPERABLE in MODE 3 when less than the P-11 setpoint, and the Steam Line Isolation on Steam Line Pressure, Low is blocked, when a secondary side break or stuck open valve could result in the rapid depressurization of the steam line(s). In MODES 1 and 2, and in MODE 3, when above the P-11 setpoint, this signal is automatically disabled and the Steam Line Pressure - Low signal is automatically enabled. The Steam Line Isolation Function is required to be OPERABLE in MODES 2 and 3 unless all MSIVs are closed. In MODES 4, 5, and 6, there is insufficient energy in the primary and secondary sides to have a SLB or other accident that would result in a release of significant enough quantities of energy to cause a cooldown of the RCS.

While the transmitters may experience elevated ambient temperatures due to a SLB, the trip function is based on rate of change, not the absolute accuracy of the indicated steam pressure. Therefore, the NTSP reflects only steady state instrument uncertainties.

#### 5. Turbine Trip and Feedwater Isolation

The primary functions of the Turbine Trip and Feedwater Isolation signals are to prevent damage to the turbine due to water in the steam lines, and to stop the excessive flow of feedwater into the SGs. These Functions are necessary to mitigate the effects of a high water level in the SGs, which could result in carryover of water into the steam lines and excessive cooldown of the primary system. The SG high water level is due to excessive feedwater flows.

The Function is actuated when the level in any SG exceeds the high high setpoint, and performs the following functions:

- Trips the main turbine,
- Trips the MFW pumps,
- Initiates feedwater isolation, and
- Shuts the MFW regulating valves and the bypass feedwater regulating valves.

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

This Function is actuated by SG Water Level - High High, or by a SI signal. The nominal trip setpoint and allowable value limits are a percentage of the narrow range instrument span for each steam generator. The RTS also initiates a turbine trip signal whenever a reactor trip (P-4) is generated. In the event of SI, the unit is taken off line and the turbine generator must be tripped. The MFW System is also taken out of operation and the AFW System is automatically started. The SI signal was discussed previously.

a. Turbine Trip and Feedwater Isolation - Automatic Actuation Logic and Actuation Relays

Automatic Actuation Logic and Actuation Relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b.

b. Turbine Trip and Feedwater Isolation - Steam Generator Water Level - High High (P-14)

This signal provides protection against excessive feedwater flow. The ESFAS SG water level instruments provide input to the SG Water Level Control System. Therefore, the actuation logic must be able to withstand both an input failure to the control system (which may then require the protection function actuation) and a single failure in the other channels providing the protection function actuation. Only three protection channels are necessary, with a two-out-of-three logic, to satisfy the protective requirements because a median signal selector is provided.

The transmitters (d/p cells) are located inside containment. However, the events that this Function protects against cannot cause a severe environment in containment. Therefore, the NTSP reflects only steady state instrument uncertainties.

c. Turbine Trip and Feedwater Isolation - Safety Injection

Turbine Trip and Feedwater Isolation is also initiated by all Functions that initiate SI. The Feedwater Isolation Function requirements for these Functions are the same as the requirements for their SI function. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead Function 1, SI, is referenced for all initiating functions and requirements.

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Turbine Trip and Feedwater Isolation Functions must be OPERABLE in MODES 1 and 2 and 3 except when all MFIVs, MFRVs, and associated MFRV bypass valves are closed or isolated by a closed manual valve when the MFW System is in operation and the turbine generator may be in operation. In MODES 4, 5, and 6, the MFW System and the turbine generator are not in service and this Function is not required to be OPERABLE.

6. Auxiliary Feedwater

The AFW System is designed to provide a secondary side heat sink for the reactor in the event that the MFW System is not available. The system has two motor driven pumps and a turbine driven pump, making it available during normal unit operation, during a loss of AC power, a loss of MFW, and during a Feedwater System pipe break. The normal source of water for the AFW System is the condensate storage tank (CST) (not safety related). A low pressure in the AFW suction line will automatically realign the pump suctions to the Essential Raw Cooling Water (ERCW) System (safety related).

a. Auxiliary Feedwater - Automatic Actuation Logic and Actuation Relays (Solid State Protection System)

Automatic actuation logic and actuation relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b.

b. Auxiliary Feedwater - Steam Generator Water Level - Low Low

SG Water Level - Low Low provides protection against a loss of heat sink due to a feed line break outside of containment, or a loss of MFW, which results in a loss of SG water level. SG Water Level - Low Low provides input to the SG Level Control System. Therefore, the actuation logic must be able to withstand both an input failure to the control system which may then require a protection function actuation and a single failure in the other channels providing the protection function actuation. Only three protection channels, with a two-out-of-three logic, are necessary to satisfy the protective requirements because a median signal selector is provided.

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

With the transmitters located inside containment and thus possibly experiencing adverse environmental conditions (due to a feedline break), the Environmental Allowance Modifier (EAM) was devised. The EAM function (Containment Pressure (EAM) with a setpoint of  $< 0.5$  psig) senses the presence of adverse containment conditions (elevated pressure) and enables the Steam Generator Water Level - Low-Low setpoint (Adverse) which reflects the increased transmitter uncertainties due to this environment. The EAM allows the use of a lower Steam Generator Water Level - Low-Low (EAM) setpoint when these conditions are not present, thus allowing more margin for normal operating conditions. Additionally, the NTSP reflects the inclusion of both steady state and adverse environmental instrument uncertainties.

The Trip Time Delay (TTD) creates additional operational margin when the plant needs it most, during early escalation to power, by allowing the operator time to recover level when the primary side load is sufficiently small to allow such action. The TTD is based on continuous monitoring of primary side power through the use of RCS loop  $\Delta T$ . Two time delays are calculated, based on the number of steam generators indicating less than the Low-Low Level setpoint and the primary side power level. The magnitude of the delays decreases with increasing primary side power level, up to 50% RTP. Above 50% RTP there are no time delays for the Low-Low level trips.

In the event of failure of a Steam Generator Water Level channel, it is placed in the trip condition as input to the Solid State Protection System and does not affect either the EAM or TTD setpoint calculations for the remaining OPERABLE channels. Failure of the Containment Pressure (EAM) channel to a protection set also does not affect the EAM setpoint calculations. This results in the requirement that the operator adjust the affected Steam Generator Water Level - Low-Low (EAM) trip setpoints to the same value as the Steam Generator Water Level - Low-Low (Adverse) trip setpoints or actuate the SG Water Level Low-Low setpoint. Failure of the RCS loop  $\Delta T$  channel input (failure of more than one  $T_H$  resistance temperature detectors (RTD) or failure of a  $T_C$  RTD) does not affect the TTD calculation for a protection set. This results in the requirement that the operator adjust the threshold power level for zero seconds time delay from 50% RTP to 0% RTP, through the man-machine-interface (MMI) test cart.

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### APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

There are three Steam Generator Water Level Low-Low channels per steam generator arranged in a two-out-of-three logic. These channels are arranged in four protection sets with each channel of the Containment Pressure (EAM) and RCS Loop  $\Delta T$  inputting into its associated protection set.

With the transmitters (d/p cells) located inside containment and the accidents the channel provides protection for occurring outside containment, the NTSP reflects only steady state instrument uncertainties. Because the transmitters (d/p cells) are located inside containment, thus possibly experiencing adverse environmental conditions during a feed line break inside containment, the SG Water Level-Low Low Trip Setpoint may not have sufficient margin to account for adverse environmental instrument uncertainties; in this case, AFW pump start will be provided by a Containment Pressure-High SI signal.

c. Auxiliary Feedwater - Safety Injection

A SI signal starts the motor driven and turbine driven AFW pumps. The AFW initiation functions are the same as the requirements for their SI function. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead, Function 1, SI, is referenced for all initiating functions and requirements.

d. Auxiliary Feedwater - Loss of Offsite Power

A loss of offsite power to the 6.9 kV Unit-boards (RCP buses) will be accompanied by a loss of reactor coolant pumping power and the subsequent need for some method of decay heat removal. The AFW loss of offsite power is detected by a voltage drop on each 6.9 kV shutdown board. Loss of power to either 6.9 kV shutdown board will start the turbine driven AFW pump to ensure that at least one SG contains enough water to serve as the heat sink for reactor decay heat and sensible heat removal following the reactor trip.

The loss-of-voltage relaying on the 6.9 kV shutdown board uses three solid-state voltage sensors in a two-out-of-three voltage sensor logic (27T-S1A, S1B, & S1C) for loss-of-power detection. A two-out-of-three logic from the voltage sensor channels energizes two parallel separate timing relays with a one-out-of-two logic scheme (LV1 and LV2). These voltage sensors and timing relays provide emergency diesel generator start, load-shed initiation, and subsequent turbine driven auxiliary feedwater

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

(TDAFW) pump start through separate blackout relays (BOX and BOY).

A footnote has been added to clarify that this requirement only applies to shutdown board instrumentation on the same unit. This clarification removes the potential to declare the AFW loss-of-power start instrumentation inoperable for a given unit when only the opposite unit's instrumentation is inoperable.

The AFW turbine-driven pump is considered OPERABLE when one train of the AFW loss of power start function is declared inoperable, in accordance with technical specifications, because both 6.9 kilovolt shutdown board logic trains supply this function.

Functions 6.a through 6.d must be OPERABLE in MODES 1, 2, and 3 to ensure that the SGs remain the heat sink for the reactor. SG Water Level - Low Low in any operating SG will cause the motor driven AFW pumps to start. SG Water Level - Low Low in any two operating SGs will cause the turbine driven pump to start. These Functions do not have to be OPERABLE in MODES 5 and 6 because there is not enough heat being generated in the reactor to require the SGs as a heat sink. In MODE 4, AFW actuation does not need to be OPERABLE because either AFW or residual heat removal (RHR) will already be in operation to remove decay heat or sufficient time is available to manually place either system in operation.

e. Auxiliary Feedwater - Trip of All Main Feedwater Pumps

A Trip of all MFW pumps is an indication of a loss of MFW and the subsequent need for some method of decay heat and sensible heat removal to bring the reactor back to no load temperature and pressure. A turbine driven MFW pump is equipped with one pressure switch on the control oil line for the speed control system. A low pressure signal from this pressure switch indicates a trip of that pump. A trip of all MFW pumps starts the motor driven and turbine driven AFW pumps to ensure that at least one SG is available with water to act as the heat sink for the reactor.

This Function includes a footnote indicating that MODE 2 applicability is limited to operation when one or more MFW pumps are supplying feedwater to the steam generators (SGs).

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### APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The footnote limits the Applicability to require the auto-start logic to be operable in MODE 2 only when at least one MFW pump is in service supplying feedwater to the SGs. Because plant conditions at the time of entry into Mode 2 do not allow the MFW pumps to operate, without this footnote the channels would need to be tripped resulting in an AFW start signal, starting the turbine-driven pump in addition to the motor-driven AFW pumps, which is an undesirable situation. This resolves a conflict between the MODE applicability and plant design, which does not support MFW pump operation at the time of entry into MODE 2. Also, modifying the requirement for auto-start of the AFW pumps to be only required when the MFW pumps are in service limits the potential for inadvertent AFW actuations during normal plant startups and shutdowns that could lead to reactivity control issues due to over cooling transients.

Function 6.e must be OPERABLE in MODES 1 and 2. This ensures that at least one SG is provided with water to serve as the heat sink to remove reactor decay heat and sensible heat in the event of an accident. In MODES 3, 4, and 5, the MFW pumps may be normally shut down, and thus neither pump trip is indicative of a condition requiring automatic AFW initiation.

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

f. Auxiliary Feedwater - Pump Suction Transfer on Suction Pressure – Low

A low pressure signal in the AFW pump suction line protects the AFW pumps against a loss of the normal supply of water for the pumps, the CST. Three pressure switches are located on the AFW pump suction line from the CST. A low pressure signal sensed by two of three switches will cause the emergency supply of water for the pump to be aligned to the emergency source of water. ERCW (safety grade) is then lined up to supply the AFW pumps to ensure an adequate supply of water for the AFW System to maintain at least one of the SGs as the heat sink for reactor decay heat and sensible heat removal.

Since the detectors are located in an area not affected by HELBs or high radiation, they will not experience any adverse environmental conditions however the NTSP reflects both steady state and adverse environmental instrument uncertainties.

This Function must be OPERABLE in MODES 1, 2, and 3 to ensure a safety grade supply of water for the AFW System to maintain the SGs as the heat sink for the reactor. This Function does not have to be OPERABLE in MODES 5 and 6 because there is not enough heat being generated in the reactor to require the SGs as a heat sink. In MODE 4, AFW automatic suction transfer does not need to be OPERABLE because RHR will already be in operation, or sufficient time is available to place RHR in operation, to remove decay heat.

g. Auxiliary Feedwater Suction Transfer Time Delays

A low pressure signal in the AFW pump suction line protects the AFW pumps against a loss of the normal supply of water for the pumps, the CST. The pressure switch setpoints and the logic time delays for the AFW pump suction switchover were determined to ensure that adequate net positive suction head (NPSH) for the AFW pumps is maintained during the pump suction transfer sequence.

The available NPSH for the pumps is calculated assuming a water level in the supply header that would not be reached until after the time delays are exceeded, even when accounting for the two TDAFW timers in series. The TDAFW pump has two timers because this pump can be switched to either of the two trains in the ECRW system: one timer is for the transfer to one of

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

the two trains. The timers operate in sequence to assure that the TDAFW pump is transferred to one of the ERCW trains.

This Function must be OPERABLE in MODES 1, 2, and 3 to ensure a safety grade supply of water for the AFW System to maintain the SGs as the heat sink for the reactor. This Function does not have to be OPERABLE in MODES 5 and 6 because there is not enough heat being generated in the reactor to require the SGs as a heat sink. In MODE 4, AFW automatic suction transfer does not need to be OPERABLE because RHR will already be in operation, or sufficient time is available to place RHR in operation, to remove decay heat.

7. Automatic Switchover to Containment Sump

At the end of the injection phase of a LOCA, the RWST will be nearly empty. Continued cooling must be provided by the ECCS to remove decay heat. The source of water for the ECCS pumps is automatically switched to the containment recirculation sump. The low head residual heat removal (RHR) pumps and containment spray pumps draw the water from the containment recirculation sump, the RHR pumps pump the water through the RHR heat exchanger, inject the water back into the RCS, and supply the cooled water to the other ECCS pumps. Switchover from the RWST to the containment sump must occur before the RWST empties to prevent damage to the RHR pumps and a loss of core cooling capability. For similar reasons, switchover must not occur before there is sufficient water in the containment sump to support ESF pump suction. Furthermore, early switchover must not occur to ensure that sufficient borated water is injected from the RWST. This ensures the reactor remains shut down in the recirculation mode.

a. Automatic Switchover to Containment Sump - Automatic Actuation Logic and Actuation Relays

Automatic actuation logic and actuation relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b.

b. Automatic Switchover to Containment Sump - Refueling Water Storage Tank (RWST) Level - Low Coincident With Safety Injection and Coincident With Containment Sump Level – High

During the injection phase of a LOCA, the RWST is the source of water for all ECCS pumps. A low level in the RWST coincident with a SI signal provides protection against a loss of water for the

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### APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

ECCS pumps and indicates the end of the injection phase of the LOCA. The RWST is equipped with four level transmitters. These transmitters provide no control functions. Therefore, a two-out-of-four logic is adequate to initiate the protection function actuation. Although only three channels would be sufficient, a fourth channel has been added for increased reliability.

The RWST - Low Allowable Value has both upper and lower limits. The lower limit is selected to ensure switchover occurs before the RWST empties, to prevent ECCS pump damage. The upper limit is selected to ensure containment sump tall strainer submergence.

The RSWT level transmitters are located in an area not affected by HELBs or post accident high radiation. Thus, they will not experience any adverse environmental conditions and the NTSP reflects only steady state instrument uncertainties.

Automatic switchover occurs only if the RWST low level signal is coincident with SI. This prevents accidental switchover during normal operation. Accidental switchover could damage ECCS pumps if they are attempting to take suction from an empty sump. The automatic switchover Function requirements for the SI Functions are the same as the requirements for their SI function. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead, Function 1, SI, is referenced for all initiating Functions and requirements.

Additional protection from spurious switchover is provided by requiring a Containment Sump Level - High signal as well as RWST Level - Low and SI. This ensures sufficient water is available in containment to support the recirculation phase of the accident. A Containment Sump Level - High signal must be present, in addition to the SI signal and the RWST Level - Low signal, to transfer the suctions of the RHR pumps to the containment sump. The containment sump is equipped with four level transmitters. These transmitters provide no control functions. Therefore, a two-out-of-four logic is adequate to initiate the protection function actuation. Although only three channels would be sufficient, a fourth channel has been added for increased reliability. The containment sump level Trip Setpoint/Allowable Value is selected to ensure that automatic switchover is permitted before RWST level decreases below the RWST Level - Low setpoint. This ensures an adequate suction supply to the ECCS pumps by allowing sufficient time for completion of the switchover before vortexing occurs in the

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### APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

RWST. The transmitters are located inside containment and thus possibly experience adverse environmental conditions. Therefore, the NTSP reflects the inclusion of both steady state and environmental instrument uncertainties.

These Functions must be OPERABLE in MODES 1, 2, 3, and 4 when there is a potential for a LOCA to occur, to ensure a continued supply of water for the ECCS pumps. These Functions are not required to be OPERABLE in MODES 5 and 6 because there is adequate time for the operator to evaluate unit conditions and respond by manually starting systems, pumps, and other equipment to mitigate the consequences of an abnormal condition or accident. System pressure and temperature are very low and many ESF components are administratively locked out or otherwise prevented from actuating to prevent inadvertent overpressurization of unit systems.

#### 8. Engineered Safety Feature Actuation System Interlocks

To allow some flexibility in unit operations, several interlocks are included as part of the ESFAS. These interlocks permit the operator to block some signals, automatically enable other signals, prevent some actions from occurring, and cause other actions to occur. The interlock Functions back up manual actions to ensure bypassable functions are in operation under the conditions assumed in the safety analyses.

##### a. Engineered Safety Feature Actuation System Interlocks - Reactor Trip, P-4

The P-4 interlock is enabled when a reactor trip breaker and its associated bypass breaker is open. Once the P-4 interlock is enabled, automatic SI initiation may be blocked after a 60 second time delay. This Function allows operators to take manual control of SI systems after the initial phase of injection is complete. Once SI is blocked, automatic actuation of SI cannot occur until the reactor trip breakers have been manually closed. The functions of the P-4 interlock are:

- Trip the main turbine,
- Isolate MFW with coincident low  $T_{avg}$ ,
- Prevent automatic reactivation of SI after a manual reset of SI, and

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

- Prevent opening of the MFW isolation valves if they were closed on SI or SG Water Level - High High.

Each of the above Functions is interlocked with P-4 to avert or reduce the continued cooldown of the RCS following a reactor trip. An excessive cooldown of the RCS following a reactor trip could cause an insertion of positive reactivity with a subsequent increase in generated power. To avoid such a situation, the noted Functions have been interlocked with P-4 as part of the design of the unit control and protection system. There are two P-4 channels arranged in a one-out-of-one logic per channel.

None of the noted Functions serves a mitigation function in the unit licensing basis safety analyses. Only the turbine trip Function is explicitly assumed since it is an immediate consequence of the reactor trip Function. Neither turbine trip, nor any of the other three Functions associated with the reactor trip signal, is required to show that the unit licensing basis safety analysis acceptance criteria are not exceeded.

The reactor trip breaker position switches provide input to the P-4 interlock to indicate open or close. Therefore, this Function has no adjustable trip setpoint with which to associate a NTSP and Allowable Value.

This Function must be OPERABLE in MODES 1, 2, and 3 when the reactor may be critical or approaching criticality. This Function does not have to be OPERABLE in MODE 4, 5, or 6 because the main turbine, and the MFW System are not in operation.

b. Engineered Safety Feature Actuation System Interlocks - Pressurizer Pressure, P-11

The P-11 interlock permits a normal unit cooldown and depressurization without actuation of SI or main steam line isolation. With two-out-of-three pressurizer pressure channels (discussed previously) less than the P-11 setpoint, the operator can manually block the Pressurizer Pressure - Low and Steam Line Pressure - Low SI signals and the Steam Line Pressure - Low steam line isolation signal (previously discussed). When the Steam Line Pressure - Low steam line isolation signal is manually blocked, a main steam isolation signal on Steam Line Pressure - Negative Rate - High is enabled. This provides protection for a SLB by closure of the MSIVs. With two-out-of-three pressurizer pressure channels above the P-11 setpoint, the

## BASES

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### APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Pressurizer Pressure - Low and Steam Line Pressure - Low SI signals and the Steam Line Pressure - Low steam line isolation signal are automatically enabled, and the main steam isolation on Steam Line Pressure - Negative Rate - High is disabled. The NTSP reflects only steady state instrument uncertainties.

This Function must be OPERABLE in MODES 1, 2, and 3 to allow an orderly cooldown and depressurization of the unit without the actuation of SI or main steam isolation. This Function does not have to be OPERABLE in MODE 4, 5, or 6 because system pressure must already be below the P-11 setpoint for the requirements of the heatup and cooldown curves to be met.

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

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## ACTIONS

A Note has been added in the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed on Table 3.3.2-1.

In the event a channel's NTSP is found nonconservative with respect to the Allowable Value, or the channel is not functioning as required, or the transmitter, instrument Loop, signal processing electronics, setpoint comparator output, contact output, or bistable is found inoperable, then all affected Functions provided by that channel must be declared inoperable and the LCO Condition(s) entered for the protection Function(s) affected. When the Required Channels in Table 3.3.2-1 are specified on a "per" basis (e.g., on a per steam line, per loop, per SG, etc., basis), then the Condition may be entered separately for each steam line, loop, SG, etc., as appropriate.

When the number of inoperable channels in a trip function exceed those specified in one or other related Conditions associated with a trip function, then the unit is outside the safety analysis. Therefore, LCO 3.0.3 should be immediately entered if applicable in the current MODE of operation.

### A.1

Condition A applies to all ESFAS protection functions.

Condition A addresses the situation where one or more channels or trains for one or more Functions are inoperable at the same time. The Required Action is to refer to Table 3.3.2-1 and to take the Required Actions for the protection functions affected. The Completion Times are those from the referenced Conditions and Required Actions.

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### ACTIONS (continued)

#### B.1, B.2.1, and B.2.2

Condition B applies to manual initiation of:

- SI,
- Containment Spray,
- Phase A Isolation, and
- Phase B Isolation.

This action addresses the train orientation of the SSPS for the functions listed above. If a channel or train is inoperable, 48 hours is allowed to return it to an OPERABLE status. Note that for containment spray and Phase B isolation, failure of one or both channels in one train renders the train inoperable. Condition B, therefore, encompasses both situations. The specified Completion Time is reasonable considering that there are two automatic actuation trains and another manual initiation train OPERABLE for each Function, and the low probability of an event occurring during this interval. If the train cannot be restored to OPERABLE status, the unit must be placed in a MODE in which the LCO does not apply. This is done by placing the unit in at least MODE 3 within an additional 6 hours (54 hours total time) and in MODE 5 within an additional 30 hours (84 hours total time). The allowable 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

Condition C applies to the automatic actuation logic and actuation relays for the following functions:

- SI,
- Containment Spray,
- Phase A Isolation, and
- Phase B Isolation.

## BASES

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### ACTIONS (continued)

This action addresses the train orientation of the SSPS and the master and slave relays. If one train is inoperable, 24 hours are allowed to restore the train to OPERABLE status. The 24 hours allowed for restoring the inoperable train to OPERABLE status is justified in Reference 9. The specified Completion Time is reasonable considering that there is another train OPERABLE, and the low probability of an event occurring during this interval. If the train cannot be restored to OPERABLE status, the unit must be placed in a MODE in which the LCO does not apply. This is done by placing the unit in at least MODE 3 within an additional 6 hours (30 hours total time) and in MODE 5 within an additional 30 hours (60 hours total time). The Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

The Required Actions are modified by a Note that allows one train to be bypassed for up to 4 hours for surveillance testing, provided the other train is OPERABLE. This allowance is based on the reliability analysis assumption of WCAP-10271-P-A (Ref. 10) that 4 hours is the average time required to perform train surveillance.

#### D.1, D.2.1, and D.2.2

Condition D applies to:

- Containment Pressure - High,
- Pressurizer Pressure – Low,
- Steam Line Pressure - Low,
- Steam Line Pressure - Negative Rate - High, and
- SG Water level - High High (P-14).

If one channel is inoperable, 72 hours are allowed to restore the channel to OPERABLE status or to place it in the tripped condition. Generally this Condition applies to functions that operate on two-out-of-three logic. Therefore, failure of one channel places the Function in a two-out-of-two configuration. One channel must be tripped to place the Function in a one-out-of-two configuration that satisfies redundancy requirements. The 72 hours allowed to restore the channel to OPERABLE status or to place it in the tripped condition is justified in Reference 9.

## BASES

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### ACTIONS (continued)

Failure to restore the inoperable channel to OPERABLE status or place it in the tripped condition within 72 hours requires the unit be placed in MODE 3 within the following 6 hours and MODE 4 within the next 6 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. In MODE 4, these Functions are no longer required OPERABLE.

The Required Actions are modified by a Note that allows the inoperable channel to be bypassed for up to 12 hours for surveillance testing of other channels. The 12 hours allowed for testing, are justified in Reference 9.

#### E.1, E.2.1, and E.2.2

Condition E applies to:

- Containment Spray Containment Pressure - High - High,
- Containment Phase B Isolation Containment Pressure – High - High, and
- Steam Line Isolation Containment Pressure – High – High.

None of these signals has input to a control function. Thus, two-out-of-three logic is necessary to meet acceptable protective requirements. However, a two-out-of-three design would require tripping a failed channel. This is undesirable because a single failure would then cause spurious containment spray initiation. Spurious spray actuation is undesirable because of the cleanup problems presented. Therefore, these channels are designed with two-out-of-four logic so that a failed channel may be bypassed rather than tripped. Note that one channel may be bypassed and still satisfy the single failure criterion. Furthermore, with one channel bypassed, a single instrumentation channel failure will not spuriously initiate containment spray.

To avoid the inadvertent actuation of containment spray and Phase B containment isolation, the inoperable channel should not be placed in the tripped condition. Instead it is bypassed. Restoring the channel to OPERABLE status, or placing the inoperable channel in the bypass condition within 72 hours, is sufficient to assure that the Function remains OPERABLE and minimizes the time that the Function may be in a partial trip condition (assuming the inoperable channel has failed high). The Completion Time is further justified based on the low probability of an

## BASES

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### ACTIONS (continued)

event occurring during this interval. Failure to restore the inoperable channel to OPERABLE status, or place it in the bypassed condition within 72 hours, requires the unit be placed in MODE 3 within the following 6 hours and MODE 4 within the next 6 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. In MODE 4, these Functions are no longer required OPERABLE.

The Required Actions are modified by a Note that allows one additional channel to be bypassed for up to 12 hours for surveillance testing. Placing a second channel in the bypass condition for up to 12 hours for testing purposes is acceptable based on the results of Reference 9.

#### F.1 and F.2

Condition F applies to the Steam Line Isolation, Manual Initiation ESFAS Function.

If a train or channel is inoperable, 48 hours is allowed to return it to OPERABLE status. The specified Completion Time is reasonable considering the nature of this Function, the available redundancy, and the low probability of an event occurring during this interval. If the Function cannot be returned to OPERABLE status, the associate MSIV is declared inoperable and the associated Required Actions followed for an inoperable MSIV.

#### G.1, G.2.1, and G.2.2

Condition G applies to the P-4 interlock.

For the P-4 Interlock Function, this action addresses the train orientation of the SSPS. If a train or channel is inoperable, 48 hours is allowed to return it to OPERABLE status. The specified Completion Time is reasonable considering the nature of the Function, the available redundancy, and the low probability of an event occurring during this interval. If the Function cannot be returned to OPERABLE status, the unit must be placed in MODE 3 within the next 6 hours and MODE 4 within the following 6 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. In MODE 4, the unit does not have any analyzed transients or conditions that require the explicit use of the protection function noted above.

## BASES

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### ACTIONS (continued)

#### H.1, H.2.1, and H.2.2

Condition H applies to the automatic actuation logic and actuation relays for the Steam Line Isolation Turbine Trip and Feedwater Isolation, and AFW actuation Functions.

The action addresses the train orientation of the SSPS and the master and slave relays for these functions. If one train is inoperable, 24 hours are allowed to restore the train to OPERABLE status. The 24 hours allowed for restoring the inoperable train to OPERABLE status is justified in Reference 9. The Completion Time for restoring a train to OPERABLE status is reasonable considering that there is another train OPERABLE, and the low probability of an event occurring during this interval. If the train cannot be returned to OPERABLE status, the unit must be brought to MODE 3 within the next 6 hours and MODE 4 within the following 6 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. Placing the unit in MODE 4 removes all requirements for OPERABILITY of the protection channels and actuation functions. In this MODE, the unit does not have analyzed transients or conditions that require the explicit use of the protection functions noted above.

The Required Actions are modified by a Note that allows one train to be bypassed for up to 4 hours for surveillance testing provided the other train is OPERABLE. This allowance is based on the reliability analysis (Ref. 10) assumption that 4 hours is the average time required to perform channel surveillance.

#### I.1 and I.2

Condition I applies to the following ESFAS Functions:

- Steam Generator Water Level--Low-Low (Adverse), and
- Steam Generator Water Level--Low-Low (EAM)

A known inoperable channel must be placed in the tripped condition within 6 hours. Placing the channel in the tripped condition results in a partial trip condition requiring only one-out-of-two logic for actuation of the two-out-of-three trips.

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### ACTIONS (continued)

In addition to placing the channel in the tripped condition, it is necessary to force the use of the shorter TTD by adjustment of the single steam generator time delay calculation ( $T_S$ ) to match the multiple steam generator time delay calculation ( $T_M$ ) for the affected protection set within 4 hours.

The Required Actions have been modified by a Note that allows placing the inoperable channel in the bypassed condition for up to 4 hours while performing routine surveillance testing of the other channels.

#### J.1, J.2, J.3.1, and J.3.2

Condition J applies to the Containment Pressure (EAM) coincidence with Steam Generator Water Level – Low-Low (Adverse) ESFAS Function.

Failure of the Containment Pressure (EAM) channel to a protection set does not affect the EAM setpoint calculations. A known inoperable Containment Pressure channel results in the requirement to adjust the affected Steam Generator Water Level – Low-Low (EAM) trip setpoints for the affected protection set to the same value as the Steam Generator Water Level – Low-Low (Adverse) trip setpoint within 6 hours.

An alternative to adjusting the affected Steam Generator Water Level - Low-Low (EAM) trip setpoints to the same value as the Steam Generator Water Level – Low-Low (Adverse) trip setpoints is to place the associated protection set's SG Water Level – Low-Low channels in the tripped condition within 6 hours.

If neither of the above Required Actions are completed within their associated Completion Time, then the unit must be placed in a MODE where these Functions are not required OPERABLE. This requires the unit be placed in MODE 3 within 12 hours and MODE 4 within 18 hours. The allowed Completion Times are reasonable to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. In MODE 4, these Functions are no longer required OPERABLE.

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### ACTIONS (continued)

#### K.1, K.2, K.3.1, and K.3.2

Condition K applies to the RCS Loop  $\Delta T$  coincidence with SG Water Level – Low-Low.

Failure of the RCS loop  $\Delta T$  channel input (failure of more than one  $T_H$  RTD or failure of a  $T_C$  RTD) does not affect the TTD calculation for a protection set. This results in the requirement that the operator adjust the threshold power level for zero seconds time delay from 50% RTP to 0% RTP within 6 hours. With the trip time delay adjusted to zero seconds the additional operational margin that allows the operator time to recover SG level is removed.

An alternative to adjusting the threshold power level for zero seconds time delay is to place the affected protection set's SG Water Level Low-Low level channels in the tripped condition within 6 hours.

If neither of the above Required Actions can be completed within their associated Completion Times then the unit must be placed in a MODE where these Functions are not required OPERABLE. This requires the unit be placed in MODE 3 within 12 hours and MODE 4 within 18 hours. The allowed Completion Times are reasonable to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. In MODE 4, these Functions are no longer required OPERABLE.

#### L.1 and L.2

Condition L applies to the Loss of Voltage sensors associated with the Loss of Power AFW pump start ESFAS Function. These are the same sensors for the DG loss of Voltage start.

This function is provided by voltage sensors for each train arranged in a two-out-of-three logic scheme. If a sensor is inoperable, 6 hours is allowed to return it to OPERABLE status.

If the inoperable sensor cannot be restored to OPERABLE status within the specified Completion Time, the associated AFW pump must be declared inoperable. The TDAFW pump is considered OPERABLE when at least one train of the AFW loss of power start function is OPERABLE because both 6.9 kV shutdown board logic trains supply this function.

## BASES

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### ACTIONS (continued)

#### M.1.1, M.1.2, and M.2

Condition M applies to the Loss of Voltage sensors and load shed timers associated with the Loss of Power AFW pump start ESFAS Function. These are the same sensors and timers for the DG loss of Voltage start.

This function is provided by voltage sensors for each train arranged in a two-out-of-three logic scheme with associated load shed timers arranged in a one-out-of-two logic. If two or more voltage sensors or one required load shed timer are inoperable, 1 hour is allowed to return the inoperable channel(s) to OPERABLE status.

If the inoperable sensors cannot be made OPERABLE such that only one sensor is inoperable or one required load shed timer cannot be made OPERABLE within the specified Completion Time, the associated auxiliary feedwater pump must be declared inoperable. The AFW turbine-driven pump is considered OPERABLE when at least one train of the AFW loss of power start function is OPERABLE because both 6.9 kV shutdown board logic trains supply this function.

#### N.1 and N.2

Condition N applies to the AFW pump start on trip of all MFW pumps.

This action addresses the train orientation of the SSPS for the auto start function of the AFW System on loss of all MFW pumps. The OPERABILITY of the AFW System must be assured by allowing automatic start of the AFW System pumps. If a channel is inoperable, 48 hours are allowed to return it to an OPERABLE status. If the function cannot be returned to an OPERABLE status, 6 hours are allowed to place the unit in MODE 3. The allowed Completion Time is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging unit systems. In MODE 3, the unit does not have any analyzed transients or conditions that require the explicit use of the protection function noted above. The allowance of 48 hours to return the train to an OPERABLE status is justified in Reference 10.

The Required Actions are modified by a note delaying the entry into the Required Action statement when starting or stopping MFW pumps during MODE 1. Starting and stopping MFW pumps during plant startup and shutdown is a normal evolution, which will normally be accomplished within a short time. It was not intended to result in unnecessary entries into the Required Actions, which provides a timeframe to correct unplanned equipment failures.

## BASES

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### ACTIONS (continued)

The 4 hours is consistent with similar allowances in other SQN TSs.

#### O.1

Condition O applies to the following ESFAS Functions:

- Auxiliary Feedwater Pump Suction Transfer on Suction Pressure - Low,
- Auxiliary Feedwater Suction Transfer Time Delays, Motor-Driven Pump, and
- Auxiliary Feedwater Suction Transfer Time Delays, Turbine-Driven Pump.

These functions are provided by three pressure sensors located on the suction of each AFW pump arranged in a two-out-of-three logic scheme. The motor driven AFW pumps have one time delay, while the TDAFW pump has two. The motor driven and the first TDAFW pump time delays prevent spurious transfer. The TDAFW Pump second time delay ensures ERCW Train A valves stroke open sufficiently.

If a pressure sensor channel or a time delay channel is inoperable, the associated AFW pump must be declared inoperable immediately.

#### P.1, P.2.1, and P.2.2

Condition P applies to the RWST Level - Low Coincident with Safety Injection and Coincident with Containment Sump Level - High.

RWST Level - Low Coincident With SI and Coincident With Containment Sump Level - High provides actuation of switchover to the containment sump. Note that this Function requires the comparators to energize to perform their required action. The failure of up to two channels will not prevent the operation of this Function. However, placing a failed channel in the tripped condition could result in a premature switchover to the sump, prior to the injection of the minimum volume from the RWST. Placing the inoperable channel in bypass results in a two-out-of-three logic configuration, which satisfies the requirement to allow another failure

## BASES

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### ACTIONS (continued)

without disabling actuation of the switchover when required. Restoring the channel to OPERABLE status or placing the inoperable channel in the bypass condition within 6 hours is sufficient to ensure that the Function remains OPERABLE, and minimizes the time that the Function may be in a partial trip condition (assuming the inoperable channel has failed high). The 6 hour Completion Time is justified in Reference 11. If the channel cannot be returned to OPERABLE status or placed in the bypass condition within 6 hours, the unit must be brought to MODE 3 within the following 6 hours and MODE 5 within the next 30 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. In MODE 5, the unit does not have any analyzed transients or conditions that require the explicit use of the protection functions noted above.

The Required Actions are modified by a Note that allows placing a second channel in the bypass condition for up to 4 hours for surveillance testing. The total of 12 hours to reach MODE 3 and 4 hours for a second channel to be bypassed is acceptable based on the results of Reference 11.

#### Q.1, Q.2.1, and Q.2.2

Condition Q applies to the P-11 interlock.

With one or more channels inoperable, the operator must verify that the interlock is in the required state for the existing unit condition. This action manually accomplishes the function of the interlock. Determination must be made within 1 hour. The 1 hour Completion Time is equal to the time allowed by LCO 3.0.3 to initiate shutdown actions in the event of a complete loss of ESFAS function. If the interlock is not in the required state (or placed in the required state) for the existing unit condition, the unit must be placed in MODE 3 within the next 6 hours and MODE 4 within the following 6 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. Placing the unit in MODE 4 removes all requirements for OPERABILITY of these interlocks.

BASES

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ACTIONS (continued)

R.1 and R.2

If the inoperable channel cannot be placed in the tripped condition or the TTD of the single steam generator time delay calculation ( $T_S$ ) adjusted to match the multiple steam generator time delay calculation ( $T_M$ ) for the affected protection set within the specified Completion Time, the unit must be placed in a MODE where these Functions are not required OPERABLE. This requires the unit placed in MODE 3 within 6 hours and MODE 4 within 12 hours. The allowed Completion Times are reasonable to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. In MODE 4, these Functions are no longer required OPERABLE.

S.1 and S.2

Condition S applies to the automatic actuation logic and actuation relays for the Automatic Switchover to Containment Sump.

This action addresses the train orientation of the SSPS and the master and slave relays. If one train is inoperable the unit must be placed in a MODE in which the LCO does not apply. This is done by placing the unit in at least MODE 3 within 12 hours and in MODE 5 within an additional 30 hours (42 hours total time). The Completion Times are reasonable to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

The Required Actions are modified by a Note that allows one train to be bypassed for up to 4 hours for surveillance testing, provided the other train is OPERABLE.

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SURVEILLANCE  
REQUIREMENTS

The SRs for each ESFAS Function are identified by the SRs column of Table 3.3.2-1.

A Note has been added to the SR Table to clarify that Table 3.3.2-1 determines which SRs apply to which ESFAS Functions.

Note that each channel of process protection supplies both trains of the ESFAS. When testing channel I, train A and train B must be examined. Similarly, train A and train B must be examined when testing channel II, channel III, and channel IV (if applicable). The CHANNEL CALIBRATION and COTs are performed in a manner that is consistent with the assumptions used in analytically calculating the required channel accuracies.

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### SURVEILLANCE REQUIREMENTS (continued)

#### SR 3.3.2.1

Performance of the CHANNEL CHECK ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the unit staff, based on a combination of the channel instrument uncertainties, including indication and reliability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

#### SR 3.3.2.2

SR 3.3.2.2 is the performance of an ACTUATION LOGIC TEST using the semiautomatic tester. The train being tested is placed in the bypass condition, thus preventing inadvertent actuation. Through the semiautomatic tester, all possible logic combinations, with and without applicable permissives, are tested for each protection function. In addition, the master relay coil is pulse tested for continuity. This verifies that the logic modules are OPERABLE and that there is an intact voltage signal path to the master relay coils

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

## BASES

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### SURVEILLANCE REQUIREMENTS (continued)

#### SR 3.3.2.3

SR 3.3.2.3 is the performance of a MASTER RELAY TEST. The MASTER RELAY TEST is the energizing of the master relay, verifying contact operation and a low voltage continuity check of the slave relay coil. Upon master relay contact operation, a low voltage is injected to the slave relay coil. This voltage is insufficient to pick up the slave relay, but large enough to demonstrate signal path continuity. The time allowed for the testing on a STAGGERED TEST BASIS (4 hours) is justified in Reference 12.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

#### SR 3.3.2.4

SR 3.3.2.4 is the performance of a COT.

A COT is performed on each required channel to ensure the entire channel will perform the intended Function. Setpoints must be found conservative with respect to the Allowable Values specified in Table 3.3.2-1. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable COT of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions.

The difference between the current "as-found" values and the previous test "as-left" values must be consistent with the drift allowance used in the setpoint methodology. The setpoint shall be left set consistent with the assumptions of the current unit specific setpoint methodology.

The "as-found" and "as-left" values must also be recorded and reviewed for consistency with the assumptions of Reference 7.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.3.2.4 is modified by two Notes as identified in Table 3.3.2-1. The first Note requires evaluation of channel performance for the condition where the as-found setting for the channel setpoint is outside its as-found tolerance but conservative with respect to the Allowable Value. Evaluation of channel performance will verify that the channel will

## BASES

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### SURVEILLANCE REQUIREMENTS (continued)

continue to behave in accordance with safety analysis assumptions and the channel performance assumptions in the setpoint methodology. The purpose of the assessment is to ensure confidence in the channel performance prior to returning the channel to service. For channels determined to be OPERABLE but degraded, after returning the channel to service the performance of these channels will be evaluated under the plant Corrective Action Program. Entry into the Corrective Action Program will ensure required review and documentation of the condition. The second Note requires that the as-left setting for the channel be returned to within the as-left tolerance of the NTSP. Where a setpoint more conservative than the NTSP is used in the plant surveillance procedures (field setting), the as-left and as-found tolerances, as applicable, will be applied to the surveillance procedure setpoint. This will ensure that sufficient margin to the Safety Limit and/or Analytical Limit is maintained. If the as-left channel setting cannot be returned to a setting within the as-left tolerance of the NTSP, then the channel shall be declared inoperable.

The second Note also requires that the methodologies for calculating the as-left and the as-found tolerances be in UFSAR, Section 7.1.2.

#### SR 3.3.2.5

SR 3.3.2.5 is the performance of a SLAVE RELAY TEST. The SLAVE RELAY TEST is the energizing of the slave relays. Contact operation is verified in one of two ways. Actuation equipment that may be operated in the design mitigation MODE is either allowed to function, or is placed in a condition where the relay contact operation can be verified without operation of the equipment. Actuation equipment that may not be operated in the design mitigation MODE is prevented from operation by the SLAVE RELAY TEST circuit. For this latter case, contact operation is verified by a continuity check of the circuit containing the slave relay.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.2.6

SR 3.3.2.6 is the performance of a TADOT. This test is a check of the Loss of Offsite Power Function. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable TADOT of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions.

The test also includes trip channels that provide actuation signals directly to the SSPS. The SR is modified by a Note that excludes verification of setpoints for relays. Relay setpoints require elaborate bench calibration and are verified during CHANNEL CALIBRATION.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.3.2.7

SR 3.3.2.7 is the performance of a TADOT. This test is a check of the Manual Actuation Functions and AFW pump start on trip of all MFW pumps. Each Manual Actuation Function is tested up to, and including, the master relay coils. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable TADOT of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. In some instances, the test includes actuation of the end device (i.e., pump starts, valve cycles, etc.).

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

The SR is modified by a Note that excludes verification of setpoints during the TADOT for manual initiation Functions. The manual initiation Functions have no associated setpoints.

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.2.8

SR 3.3.2.8 is the performance of a CHANNEL CALIBRATION.

CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to measured parameter within the necessary range and accuracy.

CHANNEL CALIBRATIONS must be performed consistent with the assumptions of the unit specific setpoint methodology. The difference between the current "as-found" values and the previous test "as-left" values must be consistent with the drift allowance used in the setpoint methodology.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

This SR is modified by a Note stating that this test should include verification that the time constants are adjusted to the prescribed values where applicable.

SR 3.3.2.8 is modified by two Notes as identified in Table 3.3.2-1. The first Note requires evaluation of channel performance for the condition where the as-found setting for the channel setpoint is outside its as-found tolerance but conservative with respect to the Allowable Value. Evaluation of channel performance will verify that the channel will continue to behave in accordance with safety analysis assumptions and the channel performance assumptions in the setpoint methodology. The purpose of the assessment is to ensure confidence in the channel performance prior to returning the channel to service. For channels determined to be OPERABLE but degraded, after returning the channel to service the performance of these channels will be evaluated under the plant Corrective Action Program. Entry into the Corrective Action Program will ensure required review and documentation of the condition. The second Note requires that the as-left setting for the channel be returned to within the as-left tolerance of the NTSP. Where a setpoint more conservative than the NTSP is used in the plant surveillance procedures (field setting), the as-left and as-found tolerances, as applicable, will be applied to the surveillance procedure setpoint. This will ensure that sufficient margin to the Safety Limit and/or Analytical Limit is maintained. If the as-left channel setting cannot be returned to a setting within the as-left tolerance of the NTSP, then the channel shall be declared inoperable.

The second Note also requires that the methodologies for calculating the as-left and the as-found tolerances be in UFSAR Section 7.1.2.

BASES

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.2.9

This SR ensures the individual channel ESF RESPONSE TIMES are less than or equal to the maximum values assumed in the accident analysis. Response Time testing acceptance criteria are included in the Updated Final Safety Analysis Report, Section 7.3 (Ref. 13). Individual component response times are not modeled in the analyses. The analyses model the overall or total elapsed time, from the point at which the parameter exceeds the Trip Setpoint value at the sensor, to the point at which the equipment in both trains reaches the required functional state (e.g., pumps at rated discharge pressure, valves in full open or closed position).

For channels that include dynamic transfer functions (e.g., lag, lead/lag, rate/lag, etc.), the response time test may be performed with the transfer functions set to one with the resulting measured response time compared to the appropriate UFSAR response time. Alternately, the response time test can be performed with the time constants set to their nominal value provided the required response time is analytically calculated assuming the time constants are set at their nominal values. The response time may be measured by a series of overlapping tests such that the entire response time is measured.

Response time may be verified by actual response time tests in any series of sequential, overlapping or total channel measurements, or by the summation of allocated sensor, signal processing and actuation logic response times with actual response time tests on the remainder of the channel. Allocations for sensor response times may be obtained from: (1) historical records based on acceptable response time tests (hydraulic, noise, or power interrupt tests), (2) in place, onsite, or offsite (e.g., vendor) test measurements, or (3) utilizing vendor engineering specifications. WCAP-13632-P-A, Revision 2, "Elimination of Pressure Sensor Response Time Testing Requirements," (Ref. 14) dated January 1996, provides the basis and methodology for using allocated sensor response times in the overall verification of the channel response time for specific sensors identified in the WCAP. Response time verification for other sensor types must be demonstrated by test.

WCAP-14036-P, Revision 1, "Elimination of Periodic Protection Channel Response Time Tests," (Ref. 15) provides the basis and methodology for using allocated signal processing and actuation logic response times in the overall verification of the protection system channel response time. The allocations for sensor, signal conditioning, and actuation logic response times must be verified prior to placing the component in operational service and re-verified following maintenance that may adversely affect response time. In general, electrical repair work does not

BASES

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SURVEILLANCE REQUIREMENTS (continued)

impact response time provided the parts used for repair are of the same type and value. Specific components identified in the WCAP may be replaced without verification testing. One example where response time could be affected is replacing the sensing assembly of a transmitter.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

This SR is modified by a Note that clarifies that the turbine driven AFW pump is tested within 24 hours after reaching 842 psig in the SGs.

SR 3.3.2.10

SR 3.3.2.10 is the performance of a TADOT as described in SR 3.3.2.7, except that it is performed for the P-4 Reactor Trip Interlock, and the Frequency is once per reactor trip breaker cycle. A successful test of the required contact(s) of a channel may be performed by the verification of the change of state of a single contact. This clarifies what is an acceptable TADOT. This Frequency is based on operating experience demonstrating that undetected failure of the P-4 interlock sometimes occurs when the reactor trip breaker is cycled.

The SR is modified by a Note that excludes verification of setpoints during the TADOT. The Function tested has no associated setpoint.

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REFERENCES

1. Regulatory Guide 1.105, "Setpoint for Safety Related Instrumentation," Revision 3.
2. UFSAR, Chapter 6.
3. UFSAR, Chapter 7.
4. UFSAR, Chapter 15.
5. IEEE-279-1971.
6. 10 CFR 50.49.
7. Calculation SQN-EEB-PL&S, Precautions, Limitations, and Setpoints for NSSS.
8. NUREG-1218, April 1988.
9. WCAP-14333-P-A, Rev. 1, October 1998.

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REFERENCES (continued)

10. WCAP-10271-P-A, Supplement 2, Rev. 1, June 1990.
  11. License Amendment dated June 13, 1995, Issuance of Amendments to Technical Specifications – Sequoyah Nuclear Plant, Units 1 and 2 (TAC NOS. M91990 and 91991) (ML013320052).
  12. WCAP-15376, Rev. 0. October 2000.
  13. UFSAR, Section 7.3.
  14. WCAP-13632-P-A, Revision 2, "Elimination of Pressure Sensor Response Time Testing Requirements," January 1996.
  15. WCAP-14036-P, Revision 1, "Elimination of Periodic Protection Channel Response Time Tests," December 1995.
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## B 3.3 INSTRUMENTATION

### B 3.3.3 Post Accident Monitoring (PAM) Instrumentation

#### BASES

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**BACKGROUND** The primary purpose of the PAM instrumentation is to display unit variables that provide information required by the control room operators during accident situations. This information provides the necessary support for the operator to take the manual actions for which no automatic control is provided and that are required for safety systems to accomplish their safety functions for Design Basis Accidents (DBAs).

The OPERABILITY of the accident monitoring instrumentation ensures that there is sufficient information available on selected unit parameters to monitor and to assess unit status and behavior following an accident.

The availability of accident monitoring instrumentation is important so that responses to corrective actions can be observed and the need for, and magnitude of, further actions can be determined. These essential instruments are identified by unit specific documents (Ref. 1) addressing the recommendations of Regulatory Guide 1.97 (Ref. 2) as required by Supplement 1 to NUREG-0737 (Ref. 3).

The instrument channels required to be OPERABLE by this LCO include two classes of parameters identified during unit specific implementation of Regulatory Guide 1.97 as Type A and Category 1 variables.

Type A variables are included in this LCO because they provide the primary information required for the control room operator to take specific manually controlled actions for which no automatic control is provided, and that are required for safety systems to accomplish their safety functions for DBAs.

Category 1 variables are the key variables deemed risk significant because they are needed to:

- Permit the operator to take preplanned manual actions to accomplish safe plant shutdown,
- Determine whether other systems important to safety are performing their intended functions,
- Monitor the process of accomplishing or maintaining critical safety functions, and

## BASES

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### BACKGROUND (continued)

- Provide information to the operators that will enable them to determine the likelihood of a gross breach of the barriers to radioactivity release and to determine if a gross breach of a barrier has occurred.

These key variables are identified by the unit specific Regulatory Guide 1.97 analyses (Ref. 1). These analyses identify the unit specific Type A and Category 1 variables and provide justification for deviating from the NRC proposed list of Category 1 variables.

The specific instrument Functions listed in Table 3.3.3-1 are discussed in the LCO section.

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### APPLICABLE SAFETY ANALYSES

The PAM instrumentation ensures the operability of Regulatory Guide 1.97 Type A and Category 1 variables so that the control room operating staff can:

- Take the specified, pre-planned, manually controlled actions, for which no automatic control is provided, and that are required for safety systems to accomplish their safety function,
- Determine whether systems important to safety are performing their intended functions,
- Monitor the process of accomplishing or maintaining critical safety functions,
- Determine the likelihood of a gross breach of the barriers to radioactivity release, and
- Determine if a gross breach of a barrier has occurred.

PAM instrumentation that meets the definition of Type A in Regulatory Guide 1.97 satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii). Category 1, non-Type A, instrumentation must be retained in TS because it is intended to assist operators in minimizing the consequences of accidents. Therefore, Category 1, non-Type A, variables are important for reducing public risk.

## BASES

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### LCO

The PAM instrumentation LCO provides OPERABILITY requirements for Regulatory Guide 1.97 Type A monitors, which provide information required by the control room operators to perform certain manual actions specified in the unit Emergency Operating Procedures. These manual actions ensure that a system can accomplish its safety function, and are credited in the safety analyses. Additionally, this LCO addresses Regulatory Guide 1.97 instruments that have been designated Category 1, non-Type A.

The OPERABILITY of the PAM instrumentation ensures there is sufficient information available on selected unit parameters to monitor and assess unit status following an accident. This capability is consistent with the recommendations of Reference 2.

LCO 3.3.3 requires two OPERABLE channels for most Functions. Two OPERABLE channels ensure no single failure prevents operators from getting the information necessary for them to determine the safety status of the unit, and to bring the unit to and maintain it in a safe condition following an accident.

Furthermore, OPERABILITY of two channels allows a CHANNEL CHECK during the post accident phase to confirm the validity of displayed information. More than two channels may be required at some units if the unit specific Regulatory Guide 1.97 analyses (Ref. 1) determined that failure of one accident monitoring channel results in information ambiguity (that is, the redundant displays disagree) that could lead operators to defeat or fail to accomplish a required safety function.

The exception to the two channel requirement is Containment Isolation Valve (CIV) Position. In this case, the important information is the status of the containment penetrations. The LCO requires one position indicator for each active CIV. This is sufficient to redundantly verify the isolation status of each isolable penetration either via indicated status of the active valve and prior knowledge of a passive valve, or via system boundary status. If a normally active CIV is known to be closed and deactivated, position indication is not needed to determine status. Therefore, the position indication for valves in this state is not required to be OPERABLE.

Type A and Category 1 variables are required to meet Regulatory Guide 1.97 Category 1 (Ref. 2) design and qualification requirements for seismic and environmental qualification, single failure criterion, utilization of emergency standby power, immediately accessible display, continuous readout, and recording of display.

Listed below are discussions of the specified instrument Functions listed in Table 3.3.3-1.

## BASES

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### LCO (continued)

1, 2. Reactor Coolant T<sub>HOT</sub> and Reactor Coolant T<sub>COLD</sub> (Wide Range)

Reactor Coolant T<sub>HOT</sub> and Reactor Coolant T<sub>COLD</sub> (Wide Range) are Category 1 variables provided for verification of core cooling and long term surveillance.

Reactor Coolant T<sub>HOT</sub> temperatures are used to determine reactor coolant subcooling margin. Reactor coolant subcooling margin will allow termination of safety injection (SI), if still in progress, or reinitiation of SI if it has been stopped. Reactor coolant subcooling margin is also used for unit stabilization and cooldown control.

In addition, RCS cold leg temperature is used in conjunction with RCS hot leg temperature to verify the unit conditions necessary to establish natural circulation in the RCS.

Reactor outlet temperature inputs to the Reactor Protection System are provided by two fast response resistance elements and associated transmitters in each loop. The channels provide indication over a range of 0°F to 700°F.

There are a total of four Reactor Coolant System (RCS) Hot and Cold Leg Temperature channels each. One RCS Hot leg temperature channel per loop and one Cold Leg temperature channel per loop. The instrument loops associated with RCS T<sub>HOT</sub> are 68-001, -024, -043, and -065. The instrument loops associated with RCS T<sub>COLD</sub> are 68-018, 68-041, 68-060, and 68-083.

3. Containment Pressure (Wide Range)

Containment Pressure (Wide Range) is provided for determination of potential for containment breach.

The channels provide indication over a range of -5 to 60 psig. There are a total of two Containment Pressure (Wide Range) channels. The instrument loops associated with Containment Pressure (Wide Range) are 30-310 and 30-311.

## BASES

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### LCO (continued)

#### 4. Containment Pressure (Narrow Range)

Containment Pressure (Narrow Range) is provided for determination of an actual containment breach and if a break is inside or outside containment. Additionally it is provided to monitor containment conditions following a break inside containment and verifying the accident is properly controlled.

The channels provide indication over a range of -1 to 15 psig. There are a total of two Containment Pressure (Narrow Range) channels. The instrument loops associated with Containment Pressure (Narrow Range) 30-044 and 30-045.

#### 5. Refueling Water Storage Tank Level

Refueling Water Storage Tank Level is provided to verify a water source to emergency core cooling systems and containment spray system, determine the time for initiation of cold leg recirculation following a loss of coolant accident, and for event diagnosis.

The channels provide indication over a range of 0% to 100%. There are a total of two Refueling Water Storage Tank Level channels. The instrument loops associated with Refueling Water Storage Tank Level are 63-050 and 63-051.

#### 6. Reactor Coolant Pressure (Wide Range)

Reactor Coolant Pressure (Wide Range) is provided to determine if the plant is in safe shutdown condition. It is also used for maintaining the proper relationship between RCS pressure and temperature, verifying vessel nondestructive testing criteria, maintain primary inventory subcooled (particularly with loss of offsite power), establish correct conditions for residual heat removal operation, determine whether reactor coolant pump operation should be continued, and determine whether high-head SI should be terminated or reinitiated.

The channels provide indication over a range of 0 to 3000 psig. There are a total of three Reactor Coolant Pressure (Wide Range) channels. The instrument loops associated with Reactor Coolant Pressure (Wide Range) are 68-062, 68-066, and 68-069.

## BASES

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### LCO (continued)

#### 7. Pressurizer Level (Wide Range)

Pressurizer Level (Wide Range) is provided to confirm if plant is in a safe shutdown condition. It is also provided to monitor RCS inventory, maintain pressurizer water level, and determine whether SI should be terminated or reinitiated.

The channels provide indication over a range of 0% to 100%. There are a total of three Pressurizer Level (Wide Range) channels. The instrument loops associated with Pressurizer Level (Wide Range) are 68-320, 68-335, and 68-339.

#### 8. Steam Line Pressure

Steam Line Pressure is provided to determine if a high-energy secondary line rupture occurred. It is also provided to maintain an adequate reactor heat sink and verify auxiliary feedwater to steam generator associated with pipe rupture is isolated. It can be used to monitor secondary side pressure to: (1) verify operation of pressure control steam dump system, (2) maintain plant in safe shutdown condition, and (3) monitor RCS cooldown rate. It is diverse to  $T_{\text{COLD}}$  for natural circulation determination. In addition, it can be used for identification of steam generator tube rupture and determination that faulted steam generator is isolated.

The channels provide indication over a range of 0 to 1200 psig. There are a total of eight Steam Line Pressure channels, two per loop. The instrument loops associated with Steam Line Pressure are 1-002A, 1-002B, 1-009A, 1-009B, 1-020A, 1-020B, 1-027A, and 1-027B.

#### 9. Steam Generator Level - (Wide Range)

Steam Generator Level (Wide Range) is provided to determine if heat sink is being maintained and is used for SI termination for secondary break outside containment.

The channels provide indication over a range of 0 to 100 percent. There are a total of four Steam Generator Level - (Wide Range) channels, one per steam generator. The instrument loops associated with Steam Generator Level - (Wide Range) are 3-043, 3-056, 3-098, and 3-111.

## BASES

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### LCO (continued)

#### 10. Steam Generator Level - (Narrow Range)

Steam Generator Level (Narrow Range) is provided to monitor heat sink, maintain steam generator water level, determine whether SI should be terminated, and determine which loop has SG tube rupture.

The channels provide indication over a range of 0 to 100 percent. There are a total of eight Steam Generator Level - (Narrow Range) channels, two per steam generator. The instrument loops associated with Steam Generator Level - (Narrow Range) are 3-039, 3-042, 3-052, 3-055, 3-094, 3-097, 3-107, and 3-110.

#### 11. Auxiliary Feedwater

Auxiliary Feedwater (AFW) flow is provided to determine if sufficient flow exists to maintain heat sink and for SI termination. The channels provide indication over a range of 0 to 440 gpm. The redundant channel capability for AFW flow consists of a single AFW flow channel for each Steam Generator (four total, one per steam generator) with a diverse channel consisting of three AFW valve position indicators (two level control valves for the motor driven AFW flowpath and one level control valve for the turbine driven AFW flowpath) for each steam generator (12 total).

The instrument loops associated with AFW flow are 3-163, 3-155, 3-147, and 3-170. The instrument loops associated with AFW valve position indication are 3-164, 3-164A, 3-174, 3-156, 3-156A, 3-173, 3-148, 3-148A, 3-172, 3-171, 3-171A, and 3-175.

#### 12. Reactor Coolant System Subcooling Margin Monitor

Reactor Coolant System Subcooling is provided for SI termination or reinitiation and maintenance of subcooling during depressurization.

The channels provide indication over a range of 200°F subcooled to 35°F superheat. There are a total of two Reactor Coolant System Subcooling Margin Monitor channels. The instrument loops associated with Reactor Coolant System Subcooling Margin Monitor are 94-101 and 94-102.

## BASES

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### LCO (continued)

#### 13. Containment Water Level (Wide Range)

Containment Water Level (Wide Range) is provided to verify water source for recirculation mode cooling, determine whether high energy line rupture has occurred inside or outside containment, and determine potential for containment breach caused by very high water levels.

The channels provide indication over a range of 0% to 100%. There are a total of two Containment Water Level (Wide Range) channels. The instrument loops associated with Containment Water Level (Wide Range) are 63-178 and 63-179.

#### 14. Incore Thermocouples

Incore thermocouples are provided to verify that the core is being adequately cooled, verify that RCS remains subcooled, and for monitoring the potential for fuel clad breach.

The channels provide indication over a range of 0°F to 2300°F. There are a total of 65 Incore Thermocouples. Each channel consists of one incore thermocouple. The minimum number of channels required is two channels per quadrant, eight per core, one/core quadrant/train. The two required channels in each quadrant shall be in different trains.

#### 15. Reactor Vessel Level Instrumentation

Reactor Vessel Level indication is provided for determination of core cooling. It is considered to be a more direct and less ambiguous indication of core cooling.

The channels provide indication over a range of 0% to 120% (dynamic range), 0% to 70% (lower range), and 64% to 120% (upper range). There are a total of six Reactor Vessel Level Instrument channels. The instrument loops associated with Reactor Vessel Level Dynamic Range are 68-367 and 68-370. The instrument loops associated with Reactor Vessel Level Lower Range are 68-368 and 68-371. The instrument loops associated with Reactor Vessel Level Upper Range are 68-369 and 68-372.

## BASES

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### LCO (continued)

#### 16. Containment Area Radiation Monitors

Containment area radiation monitors are provided for accident diagnosis and SI Termination/Reinitiation.

The channels provide indication over a range of  $10^0$  to  $10^8$  R/hr. There are a total of four Containment Area Radiation Monitors. The instrument loops associated with Containment Area Radiation Monitors Upper Compartment are 90-271 and 90-272. The instrument loops associated with Containment Area Radiation Monitors Lower Compartment are 90-273 and 90-274.

#### 17. Neutron Flux

The Intermediate Range and Source Range Neutron Flux are provided for monitoring reactivity control, determining if plant is subcritical, and to diagnose positive reactivity insertion.

The channels provide indication over a range of 1 to  $10^6$  CPS (Source Range) and  $10^{-8}$  to 200% RTP (Intermediate Range). There are a total of two Source Range channels and two Intermediate Range channels. The instrument loops associated with the Source Range are 92-5001 and 92-5002. The instrument loops associated with the Intermediate Range are 92-5003 and 92-5004.

#### 18. ERCW to AFW Valve Position

The ERCW to AFW valve position is provided for verification of heat sink availability. There is a total of four motor driven pump instrument loops. For the turbine driven AFW pump there is a total of four instrument loops. Each ERCW to AFW suction line contains two in-series isolation valves, each with its own position indication. Thus, position indication on both valves on a suction line is necessary. Each channel consists of the two valve position indications associated with the in-series valves in a single suction line.

The instrument loops associated with ERCW to AFW valve position for the Motor Driven Pumps are 3-116A, 3-116B, 3-126A, and 3-126B. The instrument loops associated with ERCW to AFW valve position for the Turbine Driven Pump are 3-136A, 3-136B, 3-179A, and 3-179B.

BASES

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LCO (continued)

19. Containment Isolation Valve Position

Containment Isolation valve position is provided for verification of containment isolation. There is one position indication instrument per isolation valve. The Containment Isolation valve position indications are located on Panels TR-A XX-55-6K and TR-B XX-55-6L.

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APPLICABILITY

The PAM instrumentation LCO is applicable in MODES 1, 2, and 3. These variables are related to the diagnosis and pre-planned actions required to mitigate DBAs. The applicable DBAs are assumed to occur in MODES 1, 2, and 3. In MODES 4, 5, and 6, unit conditions are such that the likelihood of an event that would require PAM instrumentation is low; therefore, the PAM instrumentation is not required to be OPERABLE in these MODES.

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ACTIONS

A Note has been added in the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed on Table 3.3.3-1. The Completion Time(s) of the inoperable channel(s) of a Function will be tracked separately for each Function starting from the time the Condition was entered for that Function.

A.1

Condition A applies when one or more Functions have one required channel that is inoperable. A Note is added stating that this ACTION is not applicable to Function 16, which has its own ACTION for one channel inoperable. Required Action A.1 requires restoring the inoperable channel to OPERABLE status within 30 days. The 30 day Completion Time is based on operating experience and takes into account the remaining OPERABLE channel (or in the case of a Function that has only one required channel, other non-Regulatory Guide 1.97 instrument channels to monitor the Function), the passive nature of the instrument (no critical automatic action is assumed to occur from these instruments), and the low probability of an event requiring PAM instrumentation during this interval. On a penetration where the position indication is declared inoperable on a valve but on the opposite side of the penetration a required containment isolation valve does not exist (such as with a closed system or a check valve), only Condition A must be entered. However, valves FCV-63-158 & -172 are both inboard penetration valves, but if both valves have inoperable position indication, Condition C must be entered until at least one of the valve's position indication is restored to OPERABLE status. Valves FCV-30-46 & VLV-30-571, FCV-30-47 & VLV-30-572, and FCV-30-48 & VLV-30-573 are all outboard penetration valves, but if both valves have inoperable

## BASES

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### ACTIONS (continued)

position indication, Condition C must be entered until at least one of the valve's position indication is restored to OPERABLE status.

#### B.1

Condition B applies when the Required Action and associated Completion Time for Condition A are not met. This Required Action specifies initiation of actions in Specification 5.6.5, which requires a written report to be submitted to the NRC immediately. This report discusses the results of the root cause evaluation of the inoperability and identifies proposed restorative actions. This action is appropriate in lieu of a shutdown requirement since alternative actions are identified before loss of functional capability, and given the likelihood of unit conditions that would require information provided by this instrumentation.

#### C.1

Condition C applies when one or more Functions have two inoperable required channels (i.e., two channels inoperable in the same Function). Required Action C.1 requires restoring one channel in the Function(s) to OPERABLE status within 7 days. The Completion Time of 7 days is based on the relatively low probability of an event requiring PAM instrument operation and the availability of alternate means to obtain the required information. Continuous operation with two required channels inoperable in a Function is not acceptable because the alternate indications may not fully meet all performance qualification requirements applied to the PAM instrumentation. Therefore, requiring restoration of one inoperable channel of the Function limits the risk that the PAM Function will be in a degraded condition should an accident occur.

#### D.1

Condition D applies when one or more Functions have three required channels that are inoperable. A Note is included that states that this ACTION is only applicable to Functions 6 and 7. Required Action D.1 requires restoring one inoperable channel to OPERABLE status within 48 hours.

## BASES

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### ACTIONS (continued)

#### E.1

Condition E applies when one or more steam generators have one AFW flow rate channel and one AFW valve position channel on the same steam generator inoperable. Required Action E.1 requires restoring one inoperable channel to OPERABLE status within 7 days.

#### F.1 and F.2

Condition F applies when one or more Containment Area Radiation Monitors have one required channel inoperable. Required Action F.1 requires initiating an alternate method of monitoring containment area radiation within 72 hours. Required Action F.2 requires restoring the inoperable channel(s) to OPERABLE status within 30 days.

#### G.1

Condition G applies when the Required Action and associated Completion Time of Conditions C, D, E, or F are not met. Required Action G.1 requires entering the appropriate Condition referenced in Table 3.3.3-1 for the channel immediately. The applicable Condition referenced in the Table is Function dependent. Each time an inoperable channel has not met the Required Action of Condition C, D, E, or F, and the associated Completion Time has expired, Condition G is entered for that channel and provides for transfer to the appropriate subsequent Condition.

#### H.1 and H.2

If the Required Action and associated Completion Time of Condition C, D, E, or F is not met and Table 3.3.3-1 directs entry into Condition H, the unit must be brought to a MODE where the requirements of this LCO do not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and MODE 4 within 12 hours.

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

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ACTIONS (continued)

I.1

Condition F requires initiation of alternate means of monitoring Containment Area Radiation. These alternate means may be temporarily installed if the normal PAM channel cannot be restored to OPERABLE status within the allotted time. If these alternate means are used, the Required Action is not to shut down the unit but rather to follow the directions of Specification 5.6.5, in the Administrative Controls section of the TS. The report provided to the NRC should discuss the alternate means used, describe the degree to which the alternate means are equivalent to the installed PAM channels, justify the areas in which they are not equivalent, and provide a schedule for restoring the normal PAM channels.

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SURVEILLANCE  
REQUIREMENTS

A Note has been added to the SR Table to clarify that SR 3.3.3.1 and SR 3.3.3.2 apply to each PAM instrumentation Function in Table 3.3.3-1.

SR 3.3.3.1

Performance of the CHANNEL CHECK ensures that a gross instrumentation failure has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION. The high radiation instrumentation should be compared to similar unit instruments located throughout the unit.

Agreement criteria are determined by the unit staff, based on a combination of the channel instrument uncertainties, including isolation, indication, and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit. If the channels are within the criteria, it is an indication that the channels are OPERABLE.

As specified in the SR, a CHANNEL CHECK is only required for those channels that are normally energized.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.3.2

CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to measured parameter with the necessary range and accuracy. This SR is modified by two Notes. The first Note excludes neutron detectors. The calibration method for neutron detectors is specified in the Bases of LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation." The second Note excludes the Containment Area Radiation Monitors detectors from a CHANNEL CALIBRATION for decade ranges above 10R/h. A CHANNEL CALIBRATION for the Containment Area Radiation Monitors detectors for decade ranges below 10R/h is performed by a single calibration check with either an installed or portable gamma source. Whenever a sensing element is replaced, the next required CHANNEL CALIBRATION of the Incore thermocouple sensors is accomplished by an inplace cross calibration that compares the other sensing elements with the recently installed sensing element.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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REFERENCES

1. SQN-EEB-PS-PAM-0001, PAM Variable QA Data – Base.
  2. Regulatory Guide 1.97, Revision 2, December 1980.
  3. NUREG-0737, Supplement 1, "TMI Action Items."
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## B 3.3 INSTRUMENTATION

### B 3.3.4 Remote Shutdown Monitoring Instrumentation

#### BASES

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**BACKGROUND** The Remote Shutdown Monitoring Instrumentation provides the control room operator with sufficient instrumentation to support placing and maintaining the unit in a safe shutdown condition from a location other than the control room. This capability is necessary to protect against the possibility that the control room becomes inaccessible. A safe shutdown condition is defined as MODE 3. With the unit in MODE 3, the Auxiliary Feedwater (AFW) System and the Main Steam Safety Valves (MSSV) or the atmospheric relief valves (ARVs) can be used to remove core decay heat and meet all safety requirements. The long term supply of water for the AFW System and the ability to borate the Reactor Coolant System (RCS) from outside the control room allows extended operation in MODE 3.

If the control room becomes inaccessible, the operators can monitor the status of the reactor at the locations shown on Table B 3.3.4-1. The unit automatically reaches MODE 3 following a unit shutdown and can be maintained safely in MODE 3 for an extended period of time.

The OPERABILITY of the remote shutdown monitoring instrumentation functions ensures there is sufficient information available on selected unit parameters to support placing and maintaining the unit in MODE 3 should the control room become inaccessible.

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**APPLICABLE SAFETY ANALYSES** The Remote Shutdown Monitoring Instrumentation is required to provide equipment at appropriate locations outside the control room with a capability to support placing and maintaining the unit in a safe condition in MODE 3.

The criteria governing the design and specific system requirements of the Remote Shutdown Monitoring Instrumentation are located in 10 CFR 50, Appendix A, GDC 19 (Ref. 1).

The Remote Shutdown Monitoring Instrumentation satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

BASES

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LCO

The Remote Shutdown Monitoring Instrumentation LCO provides the OPERABILITY requirements of the instrumentation necessary to support placing and maintaining the unit in MODE 3 from a location other than the control room. The instrumentation required are listed in Table B 3.3.4-1.

The monitoring instrumentation is required for:

- Core reactivity control (initial and long term);
- RCS pressure control;
- Reactor heat removal; and
- RCS makeup.

A Function of remote shutdown monitoring instrumentation is OPERABLE if each instrument needed to support the remote shutdown monitoring instrumentation is OPERABLE. For Table B 3.3.4-1 Function 7, the required information is available from several alternate sources. In this case, the Function is OPERABLE as long as one channel of any of the alternate information sources is OPERABLE.

The remote shutdown monitoring instrumentation covered by this LCO does not need to be energized to be considered OPERABLE. This LCO is intended to ensure the instruments will be OPERABLE if unit conditions require that the Remote Shutdown Monitoring Instrumentation be placed in operation.

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APPLICABILITY

The Remote Shutdown Monitoring Instrumentation LCO is applicable in MODES 1, 2, and 3. This is required so that the unit can be placed and maintained in MODE 3 for an extended period of time from a location other than the control room.

This LCO is not applicable in MODE 4, 5, or 6. In these MODES, the facility is already subcritical and in a condition of reduced RCS energy. Under these conditions, considerable time is available to restore the necessary instrument channels if control room instruments become unavailable.

## BASES

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### ACTIONS

The Remote Shutdown Monitoring Instrumentation LCO is not met when any required channel does not satisfy its OPERABILITY criteria for the channel's Function. These criteria are outlined in the LCO section of the Bases.

A Note has been added to the ACTIONS to clarify the application of Completion Time rules. Separate Condition entry is allowed for each Function. The Completion Time(s) of the inoperable channel(s) of a Function will be tracked separately for each Function starting from the time the Condition was entered for that Function.

#### A.1

Condition A addresses the situation where one or more required Functions of Remote Shutdown Monitoring Instrumentation are inoperable.

The Required Action is to restore the required Function to OPERABLE status within 30 days. The Completion Time is based on operating experience and the low probability of an event that would require evacuation of the control room.

#### B.1 and B.2

If the Required Action and associated Completion Time of Condition A is not met, 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 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

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### SURVEILLANCE REQUIREMENTS

#### SR 3.3.4.1

Performance of the CHANNEL CHECK ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

BASES

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SURVEILLANCE REQUIREMENTS (continued)

Agreement criteria are determined by the unit staff, based on a combination of the channel instrument uncertainties, including indication and readability. If the channels are within the criteria, it is an indication that the channels are OPERABLE. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

As specified in the Surveillance, a CHANNEL CHECK is only required for those channels which are normally energized.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.3.4.2

CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy.

Whenever a temperature sensing element is replaced, the next required CHANNEL CALIBRATION of the resistance temperature detectors (RTD) sensors is accomplished by an in-place cross calibration that compares the other sensing elements with the recently installed sensing element.

This SR is modified by two Notes, Note 1 excludes the neutron detectors and Note 2 excludes the reactor trip breaker indication from the CHANNEL CALIBRATION.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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REFERENCES      1. 10 CFR 50, Appendix A, GDC 19.

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Table B 3.3.4-1 (page 1 of 1)  
Remote Shutdown Monitoring Instrumentation

FUNCTION	READOUT LOCATION	MEASUREMENT RANGE	REQUIRED NUMBER OF CHANNELS
1. Source Range Nuclear Flux	Auxiliary Control Room Panel 1-L-10	1 to $1 \times 10^6$ cps	1
2. Reactor Trip Breaker Indication	at trip switchgear	OPEN-CLOSE	1/trip breaker
3. Reactor Coolant Temperature - Hot Leg	Auxiliary Control Room Panel 1-L-10	0-650°F	1/loop
4. Pressurizer Pressure	Auxiliary Control Room Panel 1-L-10	0-3000 psig	1
5. Pressurizer Level	Auxiliary Control Room Panel 1-L-10	0-100%	1
6. Steam Generator Pressure	Auxiliary Control Room Panel 1-L-10	0-1200 psig	1/steam generator
7. Steam Generator Level	Auxiliary Control Room Panels 1-L-11A and 1-L-11B or near Auxiliary Feedwater Pump	0-100%	1/steam generator
8. RHR Flow Rate	Auxiliary Control Room Panel 1-L-10	0-4500 gpm	1
9. RHR Temperature	Auxiliary Control Room Panel 1-L-10	50-400°F	1
10. Auxiliary Feedwater Flow Rate	Auxiliary Control Room Panel 1-L-10	0-440 gpm	1/steam generator
11. Pressurizer Relief Tank Pressure	Auxiliary Control Room Panel 1-L-10	0-100 psig	1
12. Containment Pressure	Auxiliary Control Room Panel 1-L-10	-1 to +15 psig	1

## B 3.3 INSTRUMENTATION

### B 3.3.5 Loss of Power (LOP) Diesel Generator (DG) Start Instrumentation

#### BASES

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**BACKGROUND** The DGs provide a source of emergency power when offsite power is either unavailable or is insufficiently stable to allow safe unit operation. Undervoltage protection will generate a LOP start if a loss of voltage, unbalanced voltage, or degraded voltage condition occurs in the switchyard. There are three LOP start signals for each 6.9 kV Shutdown Board.

Six undervoltage relays (two per phase) and three unbalanced voltage relays (each three phase) are provided on each 6.9 kV Shutdown Board for detecting a sustained degraded voltage condition, unbalanced voltage condition, or a loss of bus voltage. The relays are combined in different logic configurations. Loss of Voltage Function and Degraded Voltage Functions are both two-out-of-three logic circuits. Unbalanced Voltage Relay is a permissive one-out-of-two logic circuit. The Loss of Voltage Function (Function 1.a) logic generates a LOP signal if the voltage is below a nominal 80% for a short time while the Degraded Voltage Function (Function 2.a) logic generates a LOP signal if the voltage is below a nominal 93.5% for a longer time. The Unbalanced Voltage Function generates a LOP signal if the alarm relay (1.30 V at 2.95 sec) and either the Low (2.96 V at 9.95 sec) or High (18.13 V at 3.95 sec) relay actuate to the determined voltage unbalance settings.

Six timers are provided on each 6.9 kV Shutdown Board, two timers associated with the Loss of Voltage Function logic and four timers associated with the Degraded Voltage Function logic. The two Loss of Voltage timers (Diesel Start and Load Shed Timers, Function 1.b) are arranged in a one-out-of-two logic with each timer set at a nominal 1.25 seconds. The Degraded Voltage timers are arranged in two sets of two; each set in a one-out-of-two logic. One set of Degraded Voltage timers (Diesel Start and Load Shed Timers, Function 2.b) are set at a nominal 300 seconds. The other set of Degraded Voltage timers (SI/Degraded Voltage Logic Enable Timers, Function 2.c) are set at a nominal 9.5 seconds. The timing functions for the Unbalanced Voltage relays are internal to the relays and set accordingly. These timers along with the under voltage relays, ensure adequate voltage is available to the safety related loads and that unintended actuations from degraded voltage or voltage perturbations will not occur.

## BASES

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### BACKGROUND (continued)

The Loss of Voltage Function voltage sensors monitor 6.9kV Shutdown Board voltage and actuate if voltage drops below 5520 volts. If two-out-of-three Loss of Voltage, Voltage Sensors detect less than 5520 volts, a signal is sent to the Diesel Generator Start and Load Shed Timers starting the 1.25 second timer. If Shutdown Board voltage increases to above the Loss of Voltage, Voltage Sensors setpoint before the Diesel Generator Start and Load Shed Timers reach their set time, the circuit returns to normal and the timers reset. If Shutdown Board voltage does not increase above the Loss of Voltage, Voltage Sensor setpoint within 1.25 seconds, a LOP signal is generated that trips the normal and alternate feeder breakers, starts the diesel generator, and trips major 6.9kV and 480V Shutdown Board loads.

The Degraded Voltage Function voltage sensors monitor 6.9kV Shutdown Board voltage and actuate if voltage drops below 6456 volts. If two-out-of-three Degraded Voltage, Voltage Sensors detect less than 6456 volts, a signal is sent to the Diesel Generator Start and Load Shed Timers starting their 300 second timer and to the SI/Degraded Voltage Logic Enable Timers starting their 9.5 second timer. If Shutdown Board voltage increases to above the Degraded Voltage, Voltage Sensors setpoint before the Diesel Generator Start and Load Shed Timers or the SI/Degraded Voltage Logic Enable Timers reach their set time, the circuit returns to normal and the timers reset. If Shutdown Board voltage does not increase above the Degraded Voltage, Voltage Sensor setpoint within 300 seconds a LOP signal is generated that trips the normal and alternate feeder breakers, starts the Diesel Generator, and trips major 6.9kV and 480V Shutdown Board loads. If Shutdown Board voltage does not increase above the Degraded Voltage, Voltage Sensor setpoint within 9.5 seconds and a safety injection signal is present or if a safety injection signal is generated after 9.5 seconds, a signal is generated that trips major 6.9kV and 480V Shutdown Board loads.

The Unbalanced Voltage Relay Function monitors 6.9 kV Shutdown Board voltage and actuates if the alarm relay (1.30 V unbalance) and either the Low (2.96 V unbalance) or High (18.13 V unbalance) Relays have actuated. If the voltage unbalance levels drop below setpoint values before the relays time settings are met, the circuit returns to normal and timers reset. If Shutdown Board voltage unbalance remains following the pre-determined time settings, a LOP signal is generated that is sent to the Degraded Voltage trip circuitry and trips the normal and alternate feeder breakers, starts the diesel generator, and trips major 6.9 kV and 480 V Shutdown Board loads.

The LOP start actuation is described in UFSAR, Section 8.3 (Ref. 1).

## BASES

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### BACKGROUND (continued)

The Allowable Value in conjunction with the trip setpoint and LCO establishes the threshold for Engineered Safety Features Actuation System (ESFAS) action to prevent exceeding acceptable limits such that the consequences of Design Basis Accidents (DBAs) will be acceptable. The Allowable Value is considered a limiting value such that a channel is OPERABLE if the setpoint is found not to exceed the Allowable Value during the CHANNEL CALIBRATION. Note that although a channel is OPERABLE under these circumstances, the setpoint must be left adjusted to within the established calibration tolerance band of the setpoint in accordance with uncertainty assumptions stated in the referenced setpoint methodology, (as-left-criteria) and confirmed to be operating within the statistical allowances of the uncertainty terms assigned.

#### Allowable Values and LOP DG Start Instrumentation Setpoints

The Trip Setpoints used in the relays are based on the analytical limits presented in the associated setpoint scaling document/calculation. The selection of these trip setpoints is such that adequate protection is provided when all sensor and processing time delays are taken into account.

Setpoints adjusted consistent with the requirements of the Allowable Value ensure that the consequences of accidents will be acceptable, providing the unit is operated from within the LCOs at the onset of the accident and that the equipment functions as designed.

Allowable Values and/or Nominal Trip Setpoints are specified for each Function in Table 3.3.5-1. Nominal Trip Setpoints are also specified in the unit specific setpoint calculations. The trip setpoints are selected to ensure that the setpoint measured by the surveillance procedure does not exceed the Allowable Value if the relay is performing as required. If the measured setpoint does not exceed the Allowable Value, the relay is considered OPERABLE. Operation with a trip setpoint less conservative than the nominal Trip Setpoint, but within the Allowable Value, is acceptable provided that operation and testing is consistent with the assumptions of the unit specific setpoint calculation (Refs. 2, 3, 4, 6 and 7).

BASES

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APPLICABLE  
SAFETY  
ANALYSES

The LOP DG start instrumentation is required for the Engineered Safety Features (ESF) Systems to function in any accident with a loss of offsite power. Its design basis is that of the ESF Actuation System (ESFAS).

Accident analyses credit the loading of the DG based on the loss of offsite power during a loss of coolant accident (LOCA). The actual DG start has historically been associated with the ESFAS actuation. The DG loading has been included in the delay time associated with each safety system component requiring DG supplied power following a loss of offsite power. The analyses assume a non-mechanistic DG loading, which does not explicitly account for each individual component of loss of power detection and subsequent actions.

The required channels of LOP DG start instrumentation, in conjunction with the ESF systems powered from the DGs, provide unit protection in the event of any of the analyzed accidents discussed in Reference 5, in which a loss of offsite power is assumed.

The delay times assumed in the safety analysis for the ESF equipment include the 10 second DG start delay, and the appropriate sequencing delay, if applicable. The response times for ESFAS actuated equipment in LCO 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," include the appropriate DG loading and sequencing delay.

The LOP DG start instrumentation channels satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

BASES

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LCO The LCO for LOP DG start instrumentation requires that the loss of voltage, unbalanced voltage relay, and degraded voltage Functions shall be OPERABLE in MODES 1, 2, 3, and 4 when the LOP DG start instrumentation supports safety systems associated with the ESFAS, as required by Table 3.3.5-1. In MODES 5 and 6, the Functions must be OPERABLE, as required by Table 3.3.5-1, whenever the associated DG is required to be OPERABLE to ensure that the automatic start of the DG is available when needed. A channel is OPERABLE with a trip setpoint value outside its calibration tolerance band provided the trip setpoint "as-found" value does not exceed its associated Allowable Value and provided the trip setpoint "as-left" value is adjusted to a value within the "as-left" calibration tolerance band of the Nominal Trip Setpoint. A trip setpoint may be set more conservative than the Nominal Trip Setpoint as necessary in response to plant conditions. Loss of the LOP DG Start Instrumentation Function could result in the delay of safety systems initiation when required. This could lead to unacceptable consequences during accidents. During the loss of offsite power the DG powers the motor driven auxiliary feedwater pumps. Failure of these pumps to start would leave only one turbine driven pump, as well as an increased potential for a loss of decay heat removal through the secondary system.

---

APPLICABILITY The LOP DG Start Instrumentation Functions are required in MODES 1, 2, 3, and 4 because ESF Functions are designed to provide protection in these MODES. Actuation in MODE 5 or 6 is required whenever the required DG must be OPERABLE so that it can perform its function on a LOP or degraded power to the associated 6.9 kV Shutdown Boards.

---

ACTIONS In the event a channel's trip setpoint is found nonconservative with respect to the Allowable Value, or the channel is found inoperable, then the function that channel provides must be declared inoperable and the LCO Condition entered for the particular protection function affected.

Because the required channels are specified on a per shutdown board basis, the Condition may be entered separately for each shutdown board as appropriate.

A Note has been added in the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed in the LCO. The Completion Time(s) of the inoperable channel(s) of a Function will be tracked separately for each Function starting from the time the Condition was entered for that Function.

BASES

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ACTIONS (continued)

A.1

Condition A applies to the LOP DG start Functions with one or more Functions with one voltage sensor channel inoperable.

If one channel of the voltage sensors is inoperable, Required Action A.1 requires that channel to be restored to OPERABLE status within 6 hours.

The specified Completion Time is reasonable considering the Function remains fully OPERABLE and the low probability of an event occurring during these intervals.

B.1

Condition B applies when one or more Functions have two or more voltage sensor channels inoperable or one or more Functions have one required timer inoperable.

Required Action B.1.1 requires restoring all but one voltage sensor channel to OPERABLE status. Required Action B.1.2 requires restoring the required load shed timer to OPERABLE status. The 1 hour Completion Time takes into account the low probability of an event requiring a LOP start occurring during this interval.

C.1

Condition C applies when one or more unbalanced voltage relays are inoperable.

Required Action C.1 requires restoring all unbalanced voltage relays to OPERABLE status. The 1 hour Completion Time takes into account the low probability of an event requiring a LOP start occurring during this interval.

BASES

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ACTIONS (continued)

D.1

Condition D applies to each of the LOP DG start Functions when the Required Action and associated Completion Time for Condition A, B, or C are not met.

In these circumstances the Conditions specified in LCO 3.8.1, "AC Sources - Operating," or LCO 3.8.2, "AC Sources - Shutdown," for the DG made inoperable by failure of the LOP DG start instrumentation are required to be entered immediately. The actions of those LCOs provide for adequate compensatory actions to assure unit safety.

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SURVEILLANCE  
REQUIREMENTS

SR 3.3.5.1

SR 3.3.5.1 is the performance of a TADOT. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable TADOT of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. The test checks trip devices that provide actuation signals directly, bypassing the analog process control equipment. The SR is modified by a Note that excludes verification of setpoints for relays. Relay setpoints require elaborate bench calibration and are verified during CHANNEL CALIBRATION. There is a plant specific program which verifies that the instrument channel functions as required by verifying the as-left and as-found setting are consistent with those established by the setpoint methodology.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.3.5.2

SR 3.3.5.2 is the performance of a CHANNEL CALIBRATION.

The setpoints, as well as the response to a loss of voltage, unbalanced voltage, and a degraded voltage test, shall include a single point verification that the trip occurs within the required time delay, as shown in Reference 1.

BASES

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SURVEILLANCE REQUIREMENTS (continued)

CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy. There is a plant specific program which verifies that the instrument channel functions as required by verifying the as-left and as-found setting are consistent with those established by the setpoint methodology.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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REFERENCES

1. UFSAR, Section 8.3.
  2. TVA Calculation 27 DAT, "Demonstrated Accuracy Calculation 27 DAT."
  3. TVA Calculation DS1-2, "Demonstrated Accuracy Calculation DS1-2."
  4. TVA Calculation SQN-EEB-MS-TI06-0008, "Degraded Voltage Analysis."
  5. UFSAR, Chapter 15.
  6. TVA Calculation EDQ0002022016000331, "Determination of Unbalance Voltage Relay Analytical Limits."
  7. TVA Calculation EDQ0002022016000329, "Demonstrated Accuracy Calculation for Voltage Unbalance Relays."
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### B 3.3 INSTRUMENTATION

#### B 3.3.6 Containment Ventilation Isolation Instrumentation

##### BASES

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**BACKGROUND** Containment Ventilation isolation instrumentation closes the containment isolation valves in the Containment Purge System. This action isolates the containment atmosphere from the environment to minimize releases of radioactivity in the event of an accident. The Containment Purge System may be in use during reactor operation and with the reactor shutdown.

Containment Ventilation isolation initiates on a automatic safety injection (SI) signal or by manual actuation. The Bases for LCO 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," discuss initiation of SI signals.

The containment purge system has inner and outer containment isolation valves in its supply and exhaust ducts. A high radiation signal initiates containment ventilation isolation, which closes both inner and outer containment isolation valves in the Containment Purge System. This system is described in the Bases for LCO 3.6.3, "Containment Isolation Valves."

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**APPLICABLE SAFETY ANALYSES** The safety analyses assume that the containment remains intact with containment purge isolated early in the event, within approximately 300 seconds. The containment ventilation isolation radiation monitors, in addition to the SI signal, ensure closing of the containment purge supply and exhaust valves. They are also the primary means for automatically isolating containment in the event of a fuel handling accident during shutdown. Containment isolation in turn ensures meeting the containment leakage rate assumptions of the safety analyses, and ensures that the calculated accidental offsite radiological doses are below 10 CFR 100 (Ref. 1) limits (10 CFR 50.67 limits for a fuel handling accident).

The containment ventilation isolation instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

## BASES

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### LCO

The LCO requirements ensure that the instrumentation necessary to initiate Containment Ventilation Isolation, listed in Table 3.3.6-1, is OPERABLE.

#### 1. Manual Initiation

The LCO requires two channels OPERABLE. The operator can initiate Containment Ventilation Isolation at any time by using one of three sets of manual initiation switches in the control room. Either of the two Phase A and Containment Ventilation Isolation switches (HS-30-63A and HS-30-63B) or, both Phase B and Containment Ventilation Isolation switches (HS-30-64A and HS-30-64B), or both Phase B Containment Isolation switches (HS-30-68A and HS-30-68B), will actuate both trains of CVI. This action will cause actuation of all components in the same manner as any of the automatic actuation signals.

The LCO for Manual Initiation ensures the proper amount of redundancy is maintained in the manual actuation circuitry to ensure the operator has manual initiation capability.

Each channel consists of one selector switch and the interconnecting wiring to the actuation logic cabinet.

#### 2. Automatic Actuation Logic and Actuation Relays

The LCO requires two trains of Automatic Actuation Logic and Actuation Relays OPERABLE to ensure that no single random failure can prevent automatic actuation.

Automatic Actuation Logic and Actuation Relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b, SI. The applicable MODES and specified conditions for the containment ventilation isolation portion of the SI Function is different and less restrictive than those for the SI role. If one or more of the SI Functions becomes inoperable in such a manner that only the Containment Ventilation Isolation Function is affected, the Conditions applicable to the SI Functions need not be entered. The less restrictive Actions specified for inoperability of the Containment Ventilation Isolation Functions specify sufficient compensatory measures for this case.

BASES

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LCO (continued)

3. Containment Radiation

Table 3.3.6-1 specifies the number of required channels of radiation monitors to ensure that the radiation monitoring instrumentation necessary to initiate Containment Ventilation Isolation remains OPERABLE.

For sampling systems, channel OPERABILITY involves more than OPERABILITY of the channel electronics. OPERABILITY also requires correct valve lineup and sample pump operation, as well as detector OPERABILITY, for trip to occur under the conditions assumed by the safety analyses.

4. Safety Injection (SI)

Refer to LCO 3.3.2, Function 1, for all initiating Functions and requirements.

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APPLICABILITY

The Manual Initiation, Automatic Actuation Logic and Actuation Relays, Safety Injection, and Containment Radiation Functions are required OPERABLE as annotated on Table 3.3.6-1. Under these conditions, the potential exists for an accident that could release significant fission product radioactivity into containment. Therefore, the containment ventilation isolation instrumentation must be OPERABLE in these MODES.

While in MODES 5 and 6 without fuel handling in progress, the containment ventilation isolation instrumentation need not be OPERABLE since the potential for radioactive releases is minimized and operator action is sufficient to ensure post accident offsite doses are maintained within the limits of Reference 1.

The Applicability for the containment ventilation isolation on the ESFAS Safety Injection Functions are specified in LCO 3.3.2. Refer to the Bases for LCO 3.3.2 for discussion of the Safety Injection Function Applicability.

## BASES

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### ACTIONS

The most common cause of channel inoperability is outright failure or drift sufficient to exceed the tolerance allowed by unit specific calibration procedures. Typically, the drift is found to be small and results in a delay of actuation rather than a total loss of function. This determination is generally made during the performance of a COT, when the process instrumentation is set up for adjustment to bring it within specification. If the Trip Setpoint is less conservative than the tolerance specified by the calibration procedure, the channel must be declared inoperable immediately and the appropriate Condition entered.

A Note has been added to the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed in Table 3.3.6-1. The Completion Time(s) of the inoperable channel(s)/train(s) of a Function will be tracked separately for each Function starting from the time the Condition was entered for that Function.

#### A.1

Condition A applies to all Containment Ventilation Isolation Functions and addresses the train orientation of the Solid State Protection System (SSPS) and the master and slave relays for these Functions. It also addresses the failure of required radiation monitoring channel.

If a train is inoperable or the required channel is inoperable, operation may continue as long as the Required Action for the applicable Conditions of LCO 3.6.3 is met for each valve made inoperable by failure of isolation instrumentation.

A Note is added stating that Condition A is only applicable in MODE 1, 2, 3, or 4.

#### B.1 and B.2

Condition B applies to all Containment Ventilation Isolation Functions and addresses the train orientation of the SSPS and the master and slave relays for these Functions. It also addresses the failure of the required radiation monitoring channel. If a train or the required radiation monitoring channel is inoperable, operation may continue as long as the Required Action to place and maintain containment ventilation isolation valves in their closed position is met or the applicable Conditions of LCO 3.9.4, "Containment Penetrations," are met for each valve made inoperable by failure of isolation instrumentation. The Completion Time for these Required Actions is Immediately.

BASES

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ACTIONS (continued)

A Note states that Condition B is applicable during movement of irradiated fuel assemblies within containment.

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SURVEILLANCE  
REQUIREMENTS

A Note has been added to the SR Table to clarify that Table 3.3.6-1 determines which SRs apply to which Containment Ventilation Isolation Functions.

SR 3.3.6.1

Performance of the CHANNEL CHECK ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the unit staff, based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.3.6.2

SR 3.3.6.2 is the performance of an ACTUATION LOGIC TEST. The train being tested is placed in the bypass condition, thus preventing inadvertent actuation. Through the semiautomatic tester, all possible logic combinations, with and without applicable permissives, are tested for each protection function. In addition, the master relay coil is pulse tested for continuity. This verifies that the logic modules are OPERABLE and there is an intact voltage signal path to the master relay coils.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

## BASES

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### SURVEILLANCE REQUIREMENTS (continued)

The SR is modified by a Note stating that the Surveillance is only applicable to the actuation logic of the ESFAS Instrumentation.

#### SR 3.3.6.3

SR 3.3.6.3 is the performance of a MASTER RELAY TEST. The MASTER RELAY TEST is the energizing of the master relay, verifying contact operation and a low voltage continuity check of the slave relay coil. Upon master relay contact operation, a low voltage is injected to the slave relay coil. This voltage is insufficient to pick up the slave relay, but large enough to demonstrate signal path continuity.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

The SR is modified by a Note stating that the Surveillance is only applicable to the master relays of the ESFAS Instrumentation.

#### SR 3.3.6.4

A COT is performed on each required channel to ensure the entire channel will perform the intended Function. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable COT of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

This test verifies the capability of the instrumentation to provide the containment ventilation system isolation. There is a plant specific program which verifies that the instrument channel functions as required by verifying the as-left and as-found setting are consistent with those established by the setpoint methodology.

BASES

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.6.5

SR 3.3.6.5 is the performance of a SLAVE RELAY TEST. The SLAVE RELAY TEST is the energizing of the slave relays. Contact operation is verified in one of two ways. Actuation equipment that may be operated in the design mitigation mode is either allowed to function or is placed in a condition where the relay contact operation can be verified without operation of the equipment. Actuation equipment that may not be operated in the design mitigation mode is prevented from operation by the SLAVE RELAY TEST circuit. For this latter case, contact operation is verified by a continuity check of the circuit containing the slave relay.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.3.6.6

SR 3.3.6.6 is the performance of a TADOT. This test is a check of the Manual Actuation Functions. Each Manual Actuation Function is tested up to, and including, the master relay coils. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable TADOT of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. In some instances, the test includes actuation of the end device (i.e., pump starts, valve cycles, etc.).

The SR is modified by a Note that excludes verification of setpoints during the TADOT. The Functions tested have no setpoints associated with them.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.3.6.7

CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy. There is a plant specific program which verifies that the instrument channel functions as required by verifying the as-left and as-found setting are consistent with those established by the setpoint methodology.

BASES

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SURVEILLANCE REQUIREMENTS (continued)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.3.6.8

This SR ensures the individual channel response times are less than or equal to the maximum values assumed in the accident analysis. Response Time testing acceptance criteria are included in the Updated Final Safety Analysis Report, Table 7.3.1-4 (Ref. 2). Individual component response times are not modeled in the analyses. The analyses model the overall or total elapsed time, from the point at which the parameter exceeds the Trip Setpoint value to the point at which the equipment in both trains reaches the required functional state (e.g., valves in full open or closed position).

Response time may be verified by actual response time tests in any series of sequential, overlapping or total channel measurements, or by the summation of allocated signal processing and actuation logic response times with actual response time tests on the remainder of the channel.

WCAP-14036-P, Revision 1, "Elimination of Periodic Protection Channel Response Time Tests," (Ref. 3) provides the basis and methodology for using allocated signal processing and actuation logic response times in the overall verification of the protection system channel response time. The allocations for signal conditioning and actuation logic response times must be verified prior to placing the component in operational service and re-verified following maintenance that may adversely affect response time. In general, electrical repair work does not impact response time provided the parts used for repair are of the same type and value. Specific components identified in the WCAP may be replaced without verification testing.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.3.6.8 is modified by a Note stating that radiation detectors are excluded from response time testing.

BASES

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- REFERENCES
1. 10 CFR 100.11.
  2. UFSAR Table 7.3.1-4.
  3. WCAP-14036-P-A, Revision 1, "Elimination of Periodic Protection Channel Response Time Tests," December 1995.
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## B 3.3 INSTRUMENTATION

### B 3.3.7 Control Room Emergency Ventilation System (CREVS) Actuation Instrumentation

#### BASES

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**BACKGROUND** The CREVS provides an enclosed control room environment from which the unit can be operated following an uncontrolled release of radioactivity. During normal operation, the Control Building Ventilation System provides control room ventilation. Upon receipt of an actuation signal, the CREVS initiates filtered ventilation and pressurization of the control room. This system is described in the Bases for LCO 3.7.10, "Control Room Emergency Ventilation System (CREVS)."

The actuation instrumentation consists of redundant radiation monitors in the air intake. A high radiation signal from any detector will initiate its associated train of the CREVS. The control room operator can also initiate CREVS trains by manual switches in the control room. The CREVS is also actuated by a safety injection (SI) signal. The SI Function is discussed in LCO 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation."

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**APPLICABLE SAFETY ANALYSES** The control room must be kept habitable for the operators stationed there during accident recovery and post accident operations.

The CREVS acts to terminate the supply of unfiltered outside air to the control room, initiate filtration, and pressurize the control room. These actions are necessary to ensure the control room is kept habitable for the operators stationed there during accident recovery and post accident operations by minimizing the radiation exposure of control room personnel.

In MODES 1, 2, 3, and 4, the radiation monitor actuation of the CREVS is a backup for the SI signal actuation. This ensures initiation of the CREVS during a loss of coolant accident or main steam line break.

The radiation monitor actuation of the CREVS in MODES 5 and 6, during movement of irradiated fuel assemblies, and during CORE ALTERATIONS are the primary means to ensure control room habitability in the event of a fuel handling accident.

The CREVS actuation instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

BASES

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LCO

The LCO requirements ensure that instrumentation necessary to initiate the CREVS is OPERABLE.

1. Manual Initiation

The LCO requires two channels OPERABLE. The operator can initiate the CREVS at any time by using either of two switches in the control room. This action will cause actuation of all components in the same manner as any of the automatic actuation signals.

The LCO for Manual Initiation ensures the proper amount of redundancy is maintained in the manual actuation circuitry to ensure the operator has manual initiation capability.

Each channel consists of one hand switch and the interconnecting wiring to the actuation logic cabinet.

2. Control Room Radiation

The LCO specifies two required Control Room Air Intake Radiation Monitors to ensure that the radiation monitoring instrumentation necessary to initiate the CREVS remains OPERABLE.

For sampling systems, channel OPERABILITY involves more than OPERABILITY of channel electronics. OPERABILITY also requires correct valve lineups, and sample pump operation, as well as detector OPERABILITY.

3. Safety Injection

Refer to LCO 3.3.2, Function 1, for all initiating Functions and requirements.

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APPLICABILITY

The CREVS Functions must be OPERABLE in MODES 1, 2, 3, 4, 5, and 6, during movement of irradiated fuel assemblies, and during CORE ALTERATIONS to ensure a habitable environment for the control room operators.

The Applicability for the CREVS actuation on the ESFAS Safety Injection Functions are specified in LCO 3.3.2. Refer to the Bases for LCO 3.3.2 for discussion of the Safety Injection Function Applicability.

## BASES

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### ACTIONS

The most common cause of channel inoperability is outright failure or drift sufficient to exceed the tolerance allowed by the unit specific calibration procedures. Typically, the drift is found to be small and results in a delay of actuation rather than a total loss of function. This determination is generally made during the performance of a COT, when the process instrumentation is set up for adjustment to bring it within specification. If the Trip Setpoint is less conservative than the tolerance specified by the calibration procedure, the channel must be declared inoperable immediately and the appropriate Condition entered.

A Note has been added to the ACTIONS indicating that separate Condition entry is allowed for each Function. The Conditions of this Specification may be entered independently for each Function listed in Table 3.3.7-1 in the accompanying LCO. The Completion Time(s) of the inoperable channel(s)/train(s) of a Function will be tracked separately for each Function starting from the time the Condition was entered for that Function.

#### A.1

Condition A applies to the actuation logic train Function of the CREVS, the radiation monitor channel Functions, and the manual channel Functions.

If one train is inoperable, or one radiation monitor channel is inoperable in one or more Functions, 7 days are permitted to restore it to OPERABLE status. The 7 day Completion Time is the same as is allowed if one train of the mechanical portion of the system is inoperable. The basis for this Completion Time is the same as provided in LCO 3.7.10. If the channel/train cannot be restored to OPERABLE status, one CREVS train must be placed in the recirculation mode of operation. This accomplishes the actuation instrumentation Function and places the unit in a conservative mode of operation.

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

Condition B applies to the failure of two CREVS actuation trains, two radiation monitor channels, or two manual channels. The first Required Action is to place one CREVS train in the recirculation mode of operation immediately. This accomplishes the actuation instrumentation Function that may have been lost and places the unit in a conservative mode of operation. The applicable Conditions and Required Actions of LCO 3.7.10 must also be entered for the CREVS train made inoperable by the inoperable actuation instrumentation. This ensures appropriate limits are placed upon train inoperability as discussed in the Bases for LCO 3.7.10.

## BASES

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### ACTIONS (continued)

Alternatively, both trains may be placed in the recirculation mode. This ensures the CREVS function is performed even in the presence of a single failure.

#### C.1 and C.2

Condition C applies when the Required Action and associated Completion Time for Condition A or B have not been met and the unit is in MODE 1, 2, 3, or 4. The unit must be brought to a MODE in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

#### D.1 and D.2

Condition D applies when the Required Action and associated Completion Time for Condition A or B have not been met when irradiated fuel assemblies are being moved or when CORE ALTERATIONS are being performed. Movement of irradiated fuel assemblies and CORE ALTERATIONS must be suspended immediately to reduce the risk of accidents that would require CREVS actuation.

#### E.1

Condition E applies when the Required Action and associated Completion Time for Condition A or B have not been met in MODE 5 or 6. Actions must be initiated to restore the inoperable train(s) to OPERABLE status immediately to ensure adequate isolation capability in the event of a fuel handling accident.

BASES

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SURVEILLANCE  
REQUIREMENTS

A Note has been added to the SR Table to clarify that Table 3.3.7-1 determines which SRs apply to which CREVS Actuation Functions.

SR 3.3.7.1

Performance of the CHANNEL CHECK ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the unit staff, based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.3.7.2

A COT is performed on each required channel to ensure the entire channel will perform the intended function. This test verifies the capability of the instrumentation to provide the CREVS actuation. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable COT of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. There is a plant specific program which verifies that the instrument channel functions as required by verifying the as-left and as-found setting are consistent with those established by the setpoint methodology.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.7.3

SR 3.3.7.3 is the performance of a TADOT. This test is a check of the Manual Actuation Functions. Each Manual Actuation Function is tested up to, and including, the master relay coils. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable TADOT of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. In some instances, the test includes actuation of the end device (i.e., pump starts, valve cycles, etc.).

The test also includes trip devices that provide actuation signals directly to the Solid State Protection System, bypassing the analog process control equipment.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

The SR is modified by a Note that excludes verification of setpoints during the TADOT. The Functions tested have no setpoints associated with them.

SR 3.3.7.4

CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy.

There is a plant specific program which verifies that the instrument channel functions as required by verifying the as-left and as-found setting are consistent with those established by the setpoint methodology.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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REFERENCES

1. WCAP-15376, Rev. 0, October 2000.
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### B 3.3 INSTRUMENTATION

#### B 3.3.8 Auxiliary Building Gas Treatment System (ABGTS) Actuation Instrumentation

##### BASES

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**BACKGROUND** The ABGTS ensures that radioactive materials in the auxiliary building atmosphere following a fuel handling accident involving handling irradiated fuel or a loss of coolant accident (LOCA) are filtered and adsorbed prior to exhausting to the environment. The system is described in the Bases for LCO 3.7.12, "Auxiliary Building Gas Treatment System (ABGTS)." The system initiates filtered ventilation of exhaust air from the fuel handling area, ECCS pump rooms, and waste packaging area automatically following receipt of a spent fuel pool area high radiation signal or a Containment Phase A Isolation signal. Initiation may also be performed manually as needed from the main control room.

High area radiation, monitored by either of two monitors, provides ABGTS initiation. Each ABGTS train is initiated by high radiation detected by a channel dedicated to that train. There are a total of two channels, one for each train. High radiation detected by the required monitor or Containment Phase A Isolation signal from the Engineered Safety Features Actuation System (ESFAS) initiates auxiliary building isolation and starts the ABGTS. These actions function to prevent exfiltration of contaminated air by initiating filtered ventilation, which imposes a negative pressure on the Auxiliary Building Secondary Containment Enclosure (ABSCE). During plant operations with the containment open to the auxiliary building, the ABSCE boundary is extended to include the area inside the containment building and the shield building.

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**APPLICABLE SAFETY ANALYSES** The ABGTS ensures that radioactive materials in the ABSCE atmosphere following a LOCA are filtered and adsorbed prior to being exhausted to the environment. This action reduces the radioactive content in the auxiliary building exhaust following a LOCA so that offsite doses remain within the limits specified in 10 CFR 100 (Ref. 1).

High radiation initiation of ABGTS for a fuel handling accident will ensure the auxiliary building secondary containment enclosure (ABSCE) boundary is established such that the release point for the fission products will correspond to the release point assumed in the fuel handling accident analysis so that control room doses remain within the limits specified in 10 CFR 50.67. (Ref. 2)

The ABGTS actuation instrumentation for LOCA satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

BASES

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LCO

The LCO requirements ensure that instrumentation necessary to initiate the ABGTS is OPERABLE.

1. Spent Fuel Pool Area Radiation

The LCO specifies one Spent Fuel Pool Area Radiation Monitor channel to ensure that the radiation monitoring instrumentation necessary to initiate the ABGTS remains OPERABLE. One radiation monitor is dedicated to each train of ABGTS.

For sampling systems, channel OPERABILITY involves more than OPERABILITY of channel electronics. OPERABILITY may also require correct valve lineups, sample pump operation, filter motor operation, detector OPERABILITY, if these supporting features are necessary for actuation to occur under the conditions assumed by the safety analyses. The measurement range for the Spent Fuel Pool Area Monitors is  $10^{-1}$  to  $10^4$  mR/hr.

The Required Channels value is modified by a footnote stating that the Required Channel shall be associated with the ABGTS train required OPERABLE per LCO 3.7.12. This ensures a valid actuation signal will start a train of ABGTS.

2. Containment Isolation - Phase A

Refer to LCO 3.3.2, Function 3.a., for all initiating Functions and requirements.

Only the Trip Setpoint is specified for each ABGTS Function in the LCO. The Trip Setpoint limits account for instrument uncertainties, which are defined in TI-18, Radiation Monitoring Procedure (Ref. 3).

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APPLICABILITY

The automatic ABGTS actuation instrumentation is required to be OPERABLE in MODES 1, 2, 3, and 4 to remove fission products caused by post LOCA Emergency Core Cooling Systems leakage and is addressed in LCO 3.3.2.

High radiation initiation of the ABGTS must be OPERABLE in any MODE during movement of irradiated fuel assemblies or storage of fuel assemblies in the spent fuel pool to ensure automatic initiation of the ABGTS when the potential for a fuel handling accident exists. High radiation initiation of ABGTS for a fuel handling accident will ensure the auxiliary building secondary containment enclosure (ABSCE) boundary is established such that the release point for the fission products will correspond to the release point assumed in the dose consequences analysis for the fuel handling accident.

## BASES

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### ACTIONS

The most common cause of channel inoperability is outright failure or drift sufficient to exceed the tolerance allowed by unit specific calibration procedures. Typically, the drift is found to be small and results in a delay of actuation rather than a total loss of function. This determination is generally made during the performance of a COT, when the process instrumentation is set up for adjustment to bring it within specification.

If the Trip Setpoint is less conservative than the tolerance specified by the calibration procedure, the channel must be declared inoperable immediately and the appropriate Condition entered.

LCO 3.0.3 is not applicable while in MODE 5 or 6. However, since irradiated fuel assembly movement and fuel assembly storage in the spent fuel pool can occur in MODE 1, 2, 3, or 4, the ACTIONS have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving or storing irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving or storing irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement or fuel storage is independent of reactor operations. Entering LCO 3.0.3, while in MODE 1, 2, 3, or 4 would require the unit to be shutdown unnecessarily.

#### A.1

Condition A applies to the area radiation monitor being inoperable solely due to the trip setpoint not within limits. Condition A applies to the failure of the required radiation monitor channel. If the required channel is inoperable, a period of 4 hours is allowed to adjust the setpoint and restore it to OPERABLE status. If the channel cannot be restored to OPERABLE status, the Required Action and associated Completion Time of Condition B would apply.

#### B.1

Condition B applies when the Required Action and associated Completion Time for Condition A has not been met or the required channel is inoperable for reasons other than the trip setpoint not within limit. Entry into the applicable Conditions and Required Actions of LCO 3.7.12 will require that movement of irradiated fuel assemblies or movement of loads over the spent fuel pool be suspended immediately to eliminate the potential for events that could require ABGTS actuation.

BASES

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SURVEILLANCE  
REQUIREMENTS

A Note has been added to the SR Table to clarify that Table 3.3.8-1 determines which SRs apply to which ABGTS Actuation Functions.

SR 3.3.8.1

Performance of the CHANNEL CHECK ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the unit staff, based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.3.8.2

A COT is performed on each required channel to ensure the entire channel will perform the intended function. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable COT of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. This test verifies the capability of the instrumentation to provide the ABGTS actuation. There is a plant specific program which verifies that the instrument channel functions as required by verifying the as-left and as-found setting are consistent with those established by the setpoint methodology.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.8.3

CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy. There is a plant specific program which verifies that the instrument channel functions as required by verifying the as-left and as-found setting are consistent with those established by the setpoint methodology.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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- REFERENCES
1. 10 CFR 100.11.
  2. 10 CFR 50.67.
  3. TI-18, Radiation Monitoring Procedure.
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## B 3.3 INSTRUMENTATION

### B 3.3.9 Boron Dilution Monitoring Instrumentation (BDMI)

#### BASES

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**BACKGROUND** The primary purpose of the BDMI is to provide an indication of inadvertent positive reactivity changes when the reactor is in a shutdown condition (i.e., MODES 3, 4, and 5). Based on this indication, operator action can be taken to mitigate the consequences of the inadvertent addition of unborated primary grade water into the Reactor Coolant System (RCS).

The required BDMI consists of one OPERABLE channel of the two channels of source range instrumentation. The requirement for an OPERABLE source range channel ensures the capability to monitor core reactivity and detect a boron dilution event. In order to promptly detect the event in MODES 3, 4, and 5 the required source range instrumentation must provide visual, audible (count rate indication), and alarm in the control room.

The source range neutron flux monitors are used to monitor the core reactivity condition. The installed source range neutron flux monitors are part of the Nuclear Instrumentation System (NIS). These detectors are located external to the reactor vessel and detect neutrons leaking from the core.

The installed source range neutron flux monitors are dual chamber unguarded fission chamber detectors. The detectors monitor the neutron flux in counts per second. The instrument range covers six decades of neutron flux (1E+6 cps). The detectors also provide continuous visual indication in the control room, an audible count rate (selectable between the two source range neutron flux channels), and a high flux at shutdown alarm to alert operators to a possible dilution accident.

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**APPLICABLE SAFETY ANALYSES** The BDMI senses abnormal increases in source range counts per minute (flux rate). One OPERABLE source range neutron flux channel is required to provide a signal to alert the operator to unexpected changes in core reactivity. Following reactor shutdown the high flux at shutdown alarm setting will be automatically adjusted downward to a nominal value of 3 times the background count rate as the background count rate reduces. The operator does not depend entirely on this alarm setpoint but has audible indication of increasing neutron flux from the audible count rate drawer and visual indication from counts per second meters for each channel on the main control board and source range drawer. The audible count rate from the source range neutron flux monitors provides prompt and definite indication of any boron dilution. The count rate increase is proportional to the subcritical multiplication factor and allows

BASES

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APPLICABLE SAFETY ANALYSES (continued)

operators to promptly recognize the initiation of a boron dilution event. Two cases are analyzed for a boron dilution accident following reactor shutdown, beginning of life (BOL) with equilibrium xenon and BOL with a clean core. The analysis shows that for both cases > 15 minutes following the high flux at shutdown alarm for operator action time is available before reaching a  $k_{\text{eff}}$  of 1.0. Therefore, the acceptance criterion for this event is met. The source range neutron flux monitoring channel is credited in the boron dilution accident in UFSAR, Section 15.2.4 (Ref. 1).

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

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LCO

LCO 3.3.9 provides the requirements for OPERABILITY for instrumentation necessary to detect a boron dilution event and monitor core reactivity. In the applicable plant condition, the specified instrumentation is required to provide a core reactivity monitoring function and is not required to provide a trip function. Therefore, in MODES 3, 4, and 5 a single OPERABLE source range channel with visual, audible (count rate indication), and alarm in the control room is required to provide prompt indication of an inadvertent boron dilution.

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APPLICABILITY

The BDMI must be OPERABLE in MODES 3, 4, and 5 because the safety analysis identifies this system as the primary means to indicate the need for operator action to mitigate an inadvertent boron dilution of the RCS.

The BDMI OPERABILITY requirements are not applicable in MODES 1 and 2 because an inadvertent boron dilution would be terminated by a source range trip, a trip on the Power Range Neutron Flux - High (low setpoint nominally 25% RTP), or Overtemperature  $\Delta T$ . These RTS Functions are discussed in LCO 3.3.1, "RTS Instrumentation."

In MODE 6, a dilution event is precluded by locked valves that isolate the RCS from the potential source of unborated water (according to LCO 3.9.2, "Unborated Water Source Isolation Valves").

The Applicability is modified by a Note that allows the high flux at shutdown alarm to be blocked during reactor startup in MODE 3. Blocking the high flux at shutdown alarm is acceptable during startup while in MODE 3, provided the reactor trip breakers are closed with the intent to withdraw rods for startup.

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ACTIONS

The most common cause of channel inoperability is outright failure or drift of the bistable or process module sufficient to exceed the tolerance allowed by the unit specific calibration procedure. Typically, the drift is found to be small and results in a delay of actuation rather than a total loss of function. This determination of drift is generally made during the

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BASES

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## ACTIONS (continued)

performance of a COT when the process instrumentation is set up for adjustment to bring it to within specification. If the channel is outside the tolerance specified by the calibration procedure, the channel must be evaluated and the appropriate Condition entered.

A.1

Required Action A.1 requires the verification of SDM according to SR 3.1.1.1 within 1 hour and once per 12 hours thereafter. This action is intended to confirm that no unintended boron dilution has occurred while the BDMI was inoperable, and that the required SDM has been maintained. The specified Completion Time takes into consideration sufficient time for the initial determination of SDM and other information available in the control room related to SDM.

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SURVEILLANCE  
REQUIREMENTSSR 3.3.9.1

Performance of the CHANNEL CHECK ensures that gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the unit staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.9.2

SR 3.3.9.2 requires the performance of a COT to ensure that the required channel of the BDMI and associated alarm setpoint are fully operational. This test shall include verification that the High Flux at Shutdown alarm setpoint is three times the background count rate. There is a plant specific program which verifies that the instrument channel functions as required by verifying the as-left and as-found settings are consistent with those established by the setpoint methodology.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. (Ref. 2)

SR 3.3.9.3

SR 3.3.9.3 is the performance of a CHANNEL CALIBRATION. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. There is a plant specific program which verifies that the instrument channel functions as required by verifying the as-left and as-found setting are consistent with those established by the setpoint methodology. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy. The CHANNEL CALIBRATION shall include checking the discriminator voltage and adjusting if necessary.

This SR is modified by a Note that states that neutron detectors are excluded from the CHANNEL CALIBRATION.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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REFERENCES

1. UFSAR, Section 15.2.4.
  2. WCAP-15376, Revision 0, October 2000.
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## B 3.4 REACTOR COOLANT SYSTEM (RCS)

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

#### BASES

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

The RCS pressure limit is consistent with operation within the nominal operational envelope. Each pressurizer pressure indication is compared to the limit. A lower pressure will cause the reactor core to approach DNB limits.

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

The RCS flow rate normally remains constant during an operational fuel cycle with all pumps running. The minimum RCS flow limit corresponds to that assumed for DNB analyses. Each OPERABLE flow rate indication is compared to the limit. If one or more flow rate indications are unavailable, the remaining flow rate indications are averaged to come up with a value for comparison to the limit. A lower RCS flow will cause the core to approach DNB limits.

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

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**APPLICABLE SAFETY ANALYSES** The requirements of this LCO represent the initial conditions for DNB limited transients analyzed in the plant safety analyses (Ref. 1). The safety analyses have shown that transients initiated from the limits of this LCO will result in meeting the DNBR criterion. This is the acceptance limit for the RCS DNB parameters. Changes to the unit that could impact these parameters must be assessed for their impact on the DNBR criteria. The transients analyzed for include loss of coolant flow events and dropped or misaligned rod events. A key assumption for the analysis of these events is that the core power distribution is within the limits of LCO 3.1.6, "Control Bank Insertion Limits," LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)."

BASES

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APPLICABLE SAFETY ANALYSES (continued)

The pressurizer pressure limit and RCS average temperature limit correspond to the analytical limits used in the safety analyses, with allowance for measurement uncertainty.

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

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LCO

This LCO specifies limits on the monitored process variables - pressurizer pressure, RCS average temperature, and RCS total flow rate - to ensure the core operates within the limits assumed in the safety analyses. The minimum RCS flow is based on the maximum analyzed steam generator tube plugging. Operating within these limits will result in meeting the DNBR criterion in the event of a DNB limited transient.

RCS flow indication calibration must include appropriate considerations for the accuracy of feedwater flow measurement. Sequoyah Nuclear Plant (SQN) can employ either of two methods to measure feedwater flow; an installed Leading Edge Flow Meter (LEFM), or in-line feedwater flow venturis. Unlike the feedwater venturis, the LEFM is not susceptible to fouling during use and possesses a higher accuracy. These attributes make the LEFM the preferred method of measuring feedwater flow as an input to the determination of RCS flow.

In the event the LEFM is unavailable, the feedwater venturis are used to calibrate the RCS flow indicators. However, the calibration assumptions for flow measurement uncertainties is not applicable to the case where the power calorimetric is based on the venturi feedwater flow indication, even if the LEFM is used to correct the venturi feedwater flow indications for the effects of fouling. For those instances where the LEFM is unavailable, SQN Technical Requirements Manual (TRM) specifies the appropriate actions to be taken.

The numerical values for pressure, temperature, and flow rate specified in the LCO are given for the measurement location and have been adjusted for instrument error.

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APPLICABILITY

In MODE 1, the limits on pressurizer pressure, RCS coolant average temperature, and RCS flow rate must be maintained during steady state operation in order to ensure 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 that DNB is not a concern.

BASES

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APPLICABILITY (continued)

A Note has been added to indicate the limit on pressurizer pressure is not applicable during short term operational transients such as a THERMAL POWER ramp increase > 5% RTP per minute or a THERMAL POWER step increase > 10% RTP. Additionally, the limit on pressurizer pressure is not applicable during PHYSICS TESTS and during the performance of SR 3.1.3.2, moderator temperature coefficient (MTC) determination. Measurement of MTC has a high probability of causing a drop in pressure below the specified value, because the reactor coolant system temperature must be dropped several degrees below  $T_{avg}$  for an accurate MTC measurement. This results in an associated drop in pressurizer level and in a downswing of pressurizer pressure, making it difficult to maintain pressurizer pressure above the limit. These conditions represent short term perturbations where actions to control pressure variations might be counterproductive. Also, since they represent transients initiated from power levels < 100% RTP, an increased DNBR margin exists to offset the temporary pressure variations.

The DNBR limit is provided in SL 2.1.1, "Reactor Core SLs." The conditions which define the DNBR limit are less restrictive than the limits of this LCO, but violation of a Safety Limit (SL) merits a stricter, more severe Required Action. Should a violation of this LCO occur, the operator must check whether or not an SL may have been exceeded.

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ACTIONS

A.1

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

RCS total flow rate is not a controllable parameter and is not expected to vary during steady state operation. If the indicated RCS total flow rate is below the LCO limit, power must be reduced, 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 for the off normal condition, and to restore the readings within limits, and 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

BASES

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ACTIONS (continued)

potential for violation of the accident analysis bounds. The Completion Time of 6 hours is reasonable to reach the required plant conditions in an orderly manner.

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SURVEILLANCE  
REQUIREMENTS

SR 3.4.1.1

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.4.1.2

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.4.1.3

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.4.1.4

Measurement of RCS total flow rate by performance of an elbow tap differential flow method (Ref. 2) allows the installed RCS flow instrumentation to be calibrated and verifies the actual RCS flow rate is greater than or equal to the minimum required RCS flow rate.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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REFERENCES

1. UFSAR, Chapter 15.
  2. UFSAR, Section 7.2.2.2.2.
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## B 3.4 REACTOR COOLANT SYSTEM (RCS)

### B 3.4.2 RCS Minimum Temperature for Criticality

#### BASES

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BACKGROUND	<p>This LCO is based upon meeting several major considerations before the reactor can be made critical and while the reactor is critical.</p> <p>The first consideration is moderator temperature coefficient (MTC), LCO 3.1.3, "Moderator Temperature Coefficient (MTC)." In the transient and accident analyses, the MTC is assumed to be in a range from slightly positive to negative and the operating temperature is assumed to be within the nominal operating envelope while the reactor is critical. The LCO on minimum temperature for criticality helps ensure the plant is operated consistent with these assumptions.</p> <p>The second consideration is the protective instrumentation. Because certain protective instrumentation (e.g., excore neutron detectors) can be affected by moderator temperature, a temperature value within the nominal operating envelope is chosen to ensure proper indication and response while the reactor is critical.</p> <p>The third consideration is the pressurizer operating characteristics. The transient and accident analyses assume that the pressurizer is within its normal startup and operating range (i.e., saturated conditions and steam bubble present). It is also assumed that the RCS temperature is within its normal expected range for startup and power operation. Since the density of the water, and hence the response of the pressurizer to transients, depends upon the initial temperature of the moderator, a minimum value for moderator temperature within the nominal operating envelope is chosen.</p> <p>The fourth consideration is that the reactor vessel is above its minimum nil ductility reference temperature when the reactor is critical.</p>
APPLICABLE SAFETY ANALYSES	<p>Although the RCS minimum temperature for criticality is not itself an initial condition assumed in Design Basis Accidents (DBAs), the closely aligned temperature for hot zero power (HZP) is a process variable that is an initial condition of DBAs, such as the rod cluster control assembly (RCCA) withdrawal, RCCA ejection, and main steam line break accidents performed at zero power that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier.</p>

BASES

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APPLICABLE SAFETY ANALYSES (continued)

All low power safety analyses assume initial RCS loop temperatures  $\geq$  the HZP temperature of 547°F (Ref. 1). The minimum temperature for criticality limitation provides a small band, 6°F, for critical operation below HZP. This band allows critical operation below HZP during plant startup and does not adversely affect any safety analyses since the MTC is not significantly affected by the small temperature difference between HZP and the minimum temperature for criticality.

The RCS minimum temperature for criticality satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

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

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APPLICABILITY In MODE 1 and MODE 2 with  $k_{\text{eff}} \geq 1.0$ , LCO 3.4.2 is applicable since the reactor can only be critical ( $k_{\text{eff}} \geq 1.0$ ) in these MODES.

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ACTIONS A.1

If the parameters that are outside the limit cannot be restored, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 2 with  $k_{\text{eff}} < 1.0$  within 30 minutes. Rapid reactor shutdown can be readily and practically achieved within a 30 minute period. The allowed time is reasonable, based on operating experience, to reach MODE 2 with  $k_{\text{eff}} < 1.0$  in an orderly manner and without challenging plant systems.

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SURVEILLANCE REQUIREMENTS SR 3.4.2.1

RCS loop average temperature is required to be verified at or above 541°F.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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REFERENCES 1. UFSAR, Section 15.1.

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## B 3.4 REACTOR COOLANT SYSTEM (RCS)

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

#### BASES

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

The PTLR contains P/T limit curves for heatup, cooldown, 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 specific material fracture toughness requirements of the RCPB materials. Reference 2 requires an adequate margin to brittle failure during normal operation, anticipated operational occurrences, and system hydrostatic tests. It mandates the use of the American Society of Mechanical Engineers (ASME) Code, Section III, Appendix G (Ref. 3).

The neutron embrittlement effect on the material toughness is reflected by increasing the nil ductility reference temperature ( $RT_{NDT}$ ) as exposure to neutron fluence increases.

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, as necessary, based on the evaluation findings and the recommendations of Regulatory Guide 1.99 (Ref. 6).

## BASES

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### BACKGROUND (continued)

The P/T limit curves are composite curves established by superimposing limits derived from stress analyses of those portions of the reactor vessel and head that are the most restrictive. At any specific pressure, temperature, and temperature rate of change, one location within the reactor vessel will dictate the most restrictive limit. Across the 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 curve includes the Reference 2 requirement that it be  $\geq 40^{\circ}\text{F}$  above the heatup curve or the cooldown curve, and not less than the minimum permissible temperature for ISLH testing. However, the criticality curve is not operationally limiting; a more restrictive limit exists in LCO 3.4.2, "RCS Minimum Temperature for Criticality."

The 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. 7), provides a recommended methodology for evaluating an operating event that causes an excursion outside the limits.

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### APPLICABLE SAFETY ANALYSES

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

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

## BASES

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### 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 follow:

- a. The severity of the departure from the allowable operating P/T regime or the severity of the rate of change of temperature,
- b. The length of time the limits were violated (longer violations allow the temperature gradient in the thick vessel walls to become more pronounced), and
- c. The existences, sizes, and orientations of flaws in the vessel material.

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### APPLICABILITY

The RCS P/T limits LCO 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

## BASES

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### APPLICABILITY (continued)

maximum pressure. Furthermore, MODES 1 and 2 are above the temperature range of concern for nonductile failure, and stress analyses have been performed for normal maneuvering profiles, such as power ascension or descent.

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### ACTIONS

#### A.1 and A.2

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

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

Besides restoring operation 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 within 72 hours. 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. 7), 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.

## BASES

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### ACTIONS (continued)

#### B.1 and B.2

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

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

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

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

#### C.1 and C.2

Actions must be initiated immediately to correct operation outside of the P/T limits at times other than when in MODE 1, 2, 3, or 4, so that the RCPB is returned to a condition that has been verified by stress analysis.

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

## BASES

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### ACTIONS (continued)

Besides restoring operation within limits, an evaluation is required to determine if RCS operation can continue. The evaluation must verify that the RCPB integrity remains acceptable and must be completed prior to entry into MODE 4. Several methods may be used, including comparison with pre-analyzed transients in the stress analyses, or inspection of the components.

ASME Code, Section XI, Appendix E (Ref. 7), may be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline.

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

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### SURVEILLANCE REQUIREMENTS

#### SR 3.4.3.1

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

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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

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### REFERENCES

1. WCAP-7924-A, April 1975.
  2. 10 CFR 50, Appendix G.
  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. Regulatory Guide 1.99, Revision 2, May 1988.
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BASES

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REFERENCES (continued)

7. ASME, Boiler and Pressure Vessel Code, Section XI, Appendix E.
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## B 3.4 REACTOR COOLANT SYSTEM (RCS)

### B 3.4.4 RCS Loops - MODES 1 and 2

#### BASES

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

The secondary functions of the RCS include:

- Moderating the neutron energy level to the thermal state, to increase the probability of fission,
- Improving the neutron economy by acting as a reflector,
- Carrying the soluble neutron poison, boric acid,
- Providing a second barrier against fission product release to the environment, and
- Removing the heat generated in the fuel due to fission product decay following a unit shutdown.

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

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

Both transient and steady state analyses have been performed to establish the effect of flow on the departure from nucleate boiling (DNB). The transient and accident analyses for the plant have been performed assuming four RCS loops are in operation. The majority of the plant safety analyses are based on initial conditions at high core power or zero power. The accident analyses that are most important to RCP operation are the partial and complete loss of reactor coolant flow, rod withdrawal

BASES

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APPLICABLE SAFETY ANALYSES (continued)

events, startup of an inactive reactor coolant loop, and single RCP locked rotor (Ref. 1).

Steady state DNB analysis has been performed for the four RCS loop operation. These analyses establish typical allowable RCS loop average temperature and  $\Delta T$  for the design power distribution and flow as a function of RCS pressure. These analyses also establish a locus of power, pressure, and temperature conditions for which the departure from nucleate boiling ratio (DNBR) is equal to its Safety Limit value. The area of permissible operation is bounded by the combination of assumed reactor trips for high neutron flux (fixed setpoint), high pressure (fixed setpoint), low pressure (fixed setpoint), overtemperature  $\Delta T$  (variable setpoint), and overpower  $\Delta T$  (variable setpoint). The difference between the reactor trip values assumed in the safety analyses and the nominal reactor trip setpoints provides an allowance for instrumentation channel error and setpoint error.

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

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

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LCO

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

An OPERABLE RCS loop consists of an OPERABLE RCP in operation providing forced flow for heat transport and an associated OPERABLE SG.

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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, and 5.

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BASES

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APPLICABILITY (continued)

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 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level", and
  - LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level".
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ACTIONS

A.1

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

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

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SURVEILLANCE  
REQUIREMENTS

SR 3.4.4.1

This SR requires verification that each RCS loop is in operation. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal while maintaining the margin to DNB.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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REFERENCES

1. UFSAR, Chapter 15.
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## B 3.4 REACTOR COOLANT SYSTEM (RCS)

### B 3.4.5 RCS Loops - MODE 3

#### BASES

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BACKGROUND	<p>In MODE 3, the primary function of the reactor coolant is removal of decay heat and transfer of this heat, via the steam generator (SG), to the secondary plant fluid. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid.</p> <p>The reactor coolant is circulated through four RCS loops, connected in parallel to the reactor vessel, each containing an SG, a reactor coolant pump (RCP), and appropriate flow, pressure, level, and temperature instrumentation for control, protection, and indication. The reactor vessel contains the clad fuel. The SGs provide the heat sink. The RCPs circulate the water through the reactor vessel and SGs at a sufficient rate to ensure proper heat transfer and prevent fuel damage.</p> <p>In MODE 3, RCPs are used to provide forced circulation for heat removal during heatup and cooldown. The MODE 3 decay heat removal requirements are low enough that a single RCS loop with one RCP running is sufficient to remove core decay heat. However, two RCS loops are required to be OPERABLE to ensure redundant capability for decay heat removal.</p>
APPLICABLE SAFETY ANALYSES	<p>Whenever the reactor trip breakers (RTBs) are in the closed position and the control rod drive mechanisms (CRDMs) are energized, an inadvertent rod withdrawal from subcritical, resulting in a power excursion, is possible. Such a transient could be caused by a malfunction of the rod control system. In addition, the possibility of a power excursion due to the ejection of an inserted control rod is possible with the breakers closed or open. Such a transient could be caused by the mechanical failure of a CRDM.</p> <p>Therefore, in MODE 3 with the Rod Control System capable of rod withdrawal, accidental control rod withdrawal from subcritical is postulated and requires at least two RCS loops to be OPERABLE and in operation to ensure that the accident analyses limits are met. For those conditions when the Rod Control System is not capable of rod withdrawal, two RCS loops are required to be OPERABLE, but only one RCS loop is required to be in operation to ensure removal of decay heat from the core and a homogeneous boron concentration throughout the RCS.</p>

BASES

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APPLICABLE SAFETY ANALYSES (continued)

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

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

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LCO

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

When the Rod Control System is not capable of rod withdrawal, only one RCS loop in operation is necessary to ensure removal of decay heat from the core and homogenous boron concentration throughout the RCS. An additional RCS loop is required to be OPERABLE to ensure that safety analyses limits are met.

The Note permits all RCPs to be removed from operation for  $\leq 1$  hour per 8 hour period. The purpose of the Note is to perform tests that are designed to validate various accident analyses values. One of these tests is validation of the pump coastdown curve used as input to a number of accident analyses including a loss of flow accident. This test is generally performed in MODE 3 during the initial startup testing program, and as such should only be performed once. If, however, changes are made to the RCS that would cause a change to the flow characteristics of the RCS, the input values of the coastdown curve must be revalidated by conducting the test again. Another test performed during the startup testing program is the validation of rod drop times during cold conditions, both with and without flow.

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

## BASES

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### LCO (continued)

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

- a. No operations are permitted that would dilute the RCS boron concentration with coolant at boron concentrations less than required to assure the SDM of LCO 3.1.1, thereby maintaining the margin to criticality. Boron reduction with coolant at boron concentrations less than required to assure SDM is maintained is prohibited because a uniform concentration distribution throughout the RCS cannot be ensured when in natural circulation and
- b. Core outlet temperature is maintained at least 10°F below saturation temperature, so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

An OPERABLE RCS loop consists of one OPERABLE RCP and one associated OPERABLE SG, which has the minimum water level specified in SR 3.4.5.2. An RCP is OPERABLE if it is capable of being powered and is able to provide forced flow if required.

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### APPLICABILITY

In MODE 3, this LCO ensures forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. The most stringent condition of the LCO, that is, two RCS loops OPERABLE and two RCS loops in operation, applies to MODE 3 with the Rod Control System capable of rod withdrawal. The least stringent condition, that is, two RCS loops OPERABLE and one RCS loop in operation, applies to MODE 3 with the Rod Control System not capable of rod withdrawal.

Operation in other MODES is covered by:

- LCO 3.4.4, "RCS Loops - MODES 1 and 2;"
- LCO 3.4.6, "RCS Loops - MODE 4;"
- LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled;"
- LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled;"
- LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level"; and
- LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level".

BASES

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ACTIONS

A.1

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

B.1

If restoration for Required Action A.1 is not possible within 72 hours, the unit must be brought to MODE 4. In MODE 4, the unit may be placed on the Residual Heat Removal System. The additional Completion Time of 12 hours is compatible with required operations to achieve cooldown and depressurization from the existing plant conditions in an orderly manner and without challenging plant systems.

C.1

If one required RCS loop is not in operation, and the Rod Control System is capable of rod withdrawal, the Required Action is to place the Rod Control System in a condition incapable of rod withdrawal (e.g., de-energize all CRDMs by opening the RTBs or de-energizing the motor generator (MG) sets). When the Rod Control System is capable of rod withdrawal, it is postulated that a power excursion could occur in the event of an inadvertent control rod withdrawal. This mandates having the heat transfer capacity of two RCS loops in operation. If only one loop is in operation, the Rod Control System must be rendered incapable of rod withdrawal. The Completion Time of 1 hour to defeat the Rod Control System is adequate to perform this operation in an orderly manner without exposing the unit to risk for an undue time period.

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

If two required RCS loops are inoperable or a required RCS loop is not in operation, except as during conditions permitted by the Note in the LCO section, the Rod Control System must be placed in a condition incapable of rod withdrawal (e.g., all CRDMs must be de-energized by opening the RTBs or de-energizing the MG sets). All operations involving

BASES

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ACTIONS (continued)

introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 must be suspended, and action to restore one of the RCS loops to OPERABLE status and operation must be initiated. Boron dilution requires forced circulation for proper mixing, and opening the RTBs or de-energizing the MG sets removes the possibility of an inadvertent rod withdrawal. Suspending the introduction of coolant into the RCS of coolant with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 is required to assure continued safe operation. With coolant added without forced circulation, unmixed coolant could be introduced to the core, however coolant added with boron concentration meeting the minimum SDM maintains acceptable margin to subcritical operations. The immediate Completion Time reflects the importance of maintaining operation for heat removal. The action to restore must be continued until one loop is restored to OPERABLE status and operation.

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SURVEILLANCE  
REQUIREMENTS

SR 3.4.5.1

This SR requires verification that the required loops are in operation. Verification includes flow rate, temperature, and pump status monitoring, which help ensure that forced flow is providing heat removal.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.4.5.2

SR 3.4.5.2 requires verification of SG OPERABILITY. SG OPERABILITY is verified by ensuring that the secondary side narrow range water level is  $\geq 21\%$  for required RCS loops. If the SG secondary side narrow range water level is  $< 21\%$  the tubes may become uncovered and the associated loop may not be capable of providing the heat sink for removal of the decay heat.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.5.3

Verification that each required RCP is OPERABLE ensures that safety analyses limits are met. The requirement also ensures that an additional RCP can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power availability to each required RCP. Alternatively, verification that a pump is in operation also verifies proper breaker alignment and power availability.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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

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REFERENCES      None.

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## B 3.4 REACTOR COOLANT SYSTEM (RCS)

### B 3.4.6 RCS Loops - MODE 4

#### BASES

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

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

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

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

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

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**LCO** The purpose of this LCO is to require that at least two loops be OPERABLE in MODE 4 and that one of these loops be in operation. The LCO allows the two loops that are required to be OPERABLE to consist of any combination of RCS loops and RHR loops. Any one loop in operation provides enough flow to remove the decay heat from the core with forced circulation. An additional loop is required to be OPERABLE to provide redundancy for heat removal.

Note 1 permits all RCPs or RHR pumps to be removed from operation for  $\leq 1$  hour per 8 hour period. The purpose of the Note is to permit tests that are designed to validate various accident analyses values. One of the tests performed during the startup testing program is the validation of rod drop times during cold conditions, both with and without flow. The no flow test may be performed in MODE 3, 4, or 5 and requires that the pumps be

## BASES

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### LCO (continued)

stopped for a short period of time. The Note permits the stopping of the pumps in order to perform this test and validate the assumed analysis values. If changes are made to the RCS that would cause a change to the flow characteristics of the RCS, the input values must be revalidated by conducting the test again. The 1 hour time period is adequate to perform the test, and operating experience has shown that boron stratification is not a problem during this short period with no forced flow.

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

- a. No operations are permitted that would dilute the RCS boron concentration with coolant with boron concentrations less than required to meet SDM of LCO 3.1.1, "SHUTDOWN MARGIN", thereby maintaining the margin to criticality. Boron reduction with coolant at boron concentrations less than required to assure SDM is maintained is prohibited because a uniform concentration distribution throughout the RCS cannot be ensured when in natural circulation; and
- b. Core outlet temperature is maintained at least 10°F below saturation temperature, so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

Note 2 requires that a steam bubble exists in the pressurizer prior to starting any RCP.

An OPERABLE RCS loop comprises an OPERABLE RCP and an associated OPERABLE SG, which has the minimum water level specified in SR 3.4.6.2.

Similarly for the RHR System, an OPERABLE RHR loop (either A or B) comprises an OPERABLE RHR pump capable of providing forced flow to an OPERABLE RHR heat exchanger. RCPs and RHR pumps are OPERABLE if they are capable of being powered and are able to provide forced flow if required.

## BASES

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

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 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level"; and

LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level".

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## ACTIONS

### A.1

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

### A.2

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

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

BASES

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ACTIONS (continued)

B.1 and B.2

If two required loops are inoperable or a required loop is not in operation, except during conditions permitted by Note 1 in the LCO section, all operations involving introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 must be suspended and action to restore one RCS or RHR loop to OPERABLE status and operation must be initiated. The required margin to criticality must not be reduced in this type of operation. Suspending operations that would cause the introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 is required to assure continued safe operation. With coolant added without forced circulation, unmixed coolant could be introduced to the core, however coolant added with boron concentration meeting the minimum SDM maintains acceptable margin to subcritical operations. The immediate Completion Times reflect the importance of maintaining operation for decay heat removal. The action to restore must be continued until one loop is restored to OPERABLE status and operation.

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SURVEILLANCE  
REQUIREMENTS

SR 3.4.6.1

This SR requires verification that the required RCS or RHR loop is in operation and circulating reactor coolant. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.4.6.2

SR 3.4.6.2 requires verification of SG OPERABILITY. SG OPERABILITY is verified by ensuring that the secondary side water level is  $\geq 21\%$  (narrow range indication). If the SG secondary side water level is  $< 21\%$  (narrow range indication), the tubes may become uncovered and the associated loop may not be capable of providing the heat sink necessary for removal of decay heat.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.6.3

Verification that each required pump is OPERABLE ensures that an additional RCS or RHR 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 each required pump. Alternatively, verification that a pump is in operation also verifies proper breaker alignment and power availability.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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

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REFERENCES      None.

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## B 3.4 REACTOR COOLANT SYSTEM (RCS)

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

#### BASES

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**BACKGROUND** In MODE 5 with the RCS loops filled, the primary function of the reactor coolant is the removal of decay heat and transfer of this heat either to the steam generator (SG) secondary side coolant via natural circulation (Ref. 1) or the component cooling water via the residual heat removal (RHR) heat exchangers. While the principal means for decay heat removal is via the RHR System, the SGs via natural circulation (Ref. 1) are specified as a backup means for redundancy. Even though the SGs cannot produce steam in this MODE, they are capable of being a heat sink due to their large contained volume of secondary water. As long as the SG 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 reactor coolant is circulated by means of two RHR loops connected to the RCS, each loop containing an RHR heat exchanger, an RHR pump, and appropriate flow and temperature instrumentation for control, protection, and indication. One RHR pump circulates the water through the RCS at a sufficient rate to prevent boric acid stratification.

The number of loops in operation can vary to suit the operational needs. The intent of this LCO is to provide forced flow from at least one RHR loop for decay heat removal and transport. The flow provided by one RHR loop is adequate for decay heat removal. The other intent of this LCO is to require that a second path be OPERABLE to provide redundancy for heat removal.

The LCO provides for redundant paths of decay heat removal capability. The first path is an RHR loop that must be OPERABLE and in operation. The second path can be another OPERABLE RHR loop or maintaining two SGs with secondary side water levels  $\geq 21\%$  (narrow range indication) to provide an alternate method for decay heat removal via natural circulation (Ref. 1).

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

RCS Loops - MODE 5 (Loops Filled) satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

BASES

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LCO

The purpose of this LCO is to require that at least one of the RHR loops be OPERABLE and in operation with an additional RHR loop OPERABLE or two SGs with secondary side water level  $\geq 21\%$  (narrow range indication). One RHR loop provides sufficient forced circulation to perform the safety functions of the reactor coolant under these conditions. An additional RHR loop is required to be OPERABLE to meet single failure considerations. However, if the standby RHR loop is not OPERABLE, an acceptable alternate method is two SGs with their secondary side water levels  $\geq 21\%$  (narrow range indication). Should the operating RHR loop fail, the SGs could be used to remove the decay heat via natural circulation.

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

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

- a. No operations are permitted that would dilute the RCS boron concentration with coolant with boron concentrations less than required to meet SDM of LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," thereby maintaining the margin to criticality. Boron reduction with coolant at boron concentrations less than required to assure SDM is maintained is prohibited because a uniform concentration distribution throughout the RCS cannot be ensured when in natural circulation; and
- b. Core outlet temperature is maintained at least  $10^{\circ}\text{F}$  below saturation temperature, so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

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

BASES

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LCO (continued)

Note 3 requires that a steam bubble exists in the pressurizer or that the secondary side water temperature of each SG be  $\leq 25^{\circ}\text{F}$  above each of the RCS cold leg temperatures before the start of a reactor coolant pump (RCP). This restriction is to prevent a low temperature overpressure event due to a thermal transient when an RCP is started.

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

RHR pumps are OPERABLE if they are capable of being powered and are able to provide flow if required. A SG can perform as a heat sink via natural circulation when it has an adequate water level and is OPERABLE.

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APPLICABILITY

In MODE 5 with RCS loops filled, this LCO requires forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. One loop of RHR provides sufficient circulation for these purposes. However, one additional RHR loop is required to be OPERABLE, or the secondary side water level of at least two SGs is required to be  $\geq 21\%$  (narrow range indication).

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 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level"; and
  - LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level".
- 

ACTIONS

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

If one RHR loop is OPERABLE and either the required SGs have secondary side water levels  $< 21\%$  (narrow range indication), or one required RHR loop is inoperable, redundancy for heat removal is lost. Action must be initiated immediately to restore a second RHR loop to OPERABLE status or to restore the required SG secondary side water levels. Either Required Action will restore redundant heat removal paths. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal.

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BASES

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ACTIONS

C.1 and C.2

If a required RHR loop is not in operation, except during conditions permitted by Note 1, or if no required loop is OPERABLE, all operations involving introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 must be suspended and action to restore one RHR loop to OPERABLE status and operation must be initiated. Suspending the introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 is required to assure continued safe operation. With coolant added without forced circulation, unmixed coolant could be introduced to the core, however coolant added with boron concentration meeting the minimum SDM maintains acceptable margin to subcritical operations. The immediate Completion Times reflect the importance of maintaining operation for heat removal.

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SURVEILLANCE  
REQUIREMENTS

SR 3.4.7.1

This SR requires verification that the required loop is in operation and circulating reactor coolant. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.4.7.2

Verifying that at least two SGs are OPERABLE by ensuring their secondary side water levels are  $\geq 21\%$  (narrow range indication) ensures an alternate decay heat removal method via natural circulation in the event that the second RHR loop is not OPERABLE. If both RHR loops are OPERABLE, this Surveillance is not needed.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.7.3

Verification that each required RHR pump is OPERABLE ensures that an additional 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 each required RHR pump. Alternatively, verification that a pump is in operation also verifies proper breaker alignment and power availability. If secondary side water level is  $\geq 21\%$  (narrow range indication) in at least two SGs, this Surveillance is not needed.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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

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REFERENCES

1. NRC Information Notice 95-35, "Degraded Ability of Steam Generators to Remove Decay Heat by Natural Circulation."
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## B 3.4 REACTOR COOLANT SYSTEM (RCS)

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

#### BASES

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**BACKGROUND** In MODE 5 with the RCS loops not filled, the primary function of the reactor coolant is the removal of decay heat generated in the fuel, and the transfer of this heat to the component cooling water via the residual heat removal (RHR) heat exchangers. The steam generators (SGs) are not available as a heat sink when the loops are not filled. The secondary function of the reactor coolant is to act as a carrier for the soluble neutron poison, boric acid.

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

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

RCS loops in MODE 5 (loops not filled) satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

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

Note 1 permits all RHR pumps to be removed from operation for  $\leq 15$  minutes when switching from one loop to another. The circumstances for stopping both RHR pumps are to be limited to situations when the outage time is short and core outlet temperature is maintained  $\geq 10^\circ\text{F}$  below saturation temperature. The Note prohibits boron dilution with coolant at boron concentrations less than required to assure SDM of LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," is maintained or draining operations when RHR forced flow is stopped.

BASES

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LCO (continued)

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

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

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APPLICABILITY

In MODE 5 with loops not filled, this LCO requires core heat removal and coolant circulation by the RHR 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 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level"; and

LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level".

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ACTIONS

A.1

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

B.1 and B.2

If no required loop is OPERABLE or the required loop is not in operation, except during conditions permitted by Note 1, all operations involving introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 must be suspended and action must be initiated immediately to restore an RHR loop to OPERABLE status and operation. The required margin to criticality must not be reduced in this type of operation. Suspending the introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 is required to assure continued safe operation. With coolant added without forced circulation, unmixed coolant could be introduced to the core, however coolant added with boron concentration meeting the minimum SDM maintains acceptable margin to

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BASES

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ACTIONS (continued)

subcritical operations. The immediate Completion Time reflects the importance of maintaining operation for heat removal. The action to restore must continue until one loop is restored to OPERABLE status and operation.

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SURVEILLANCE  
REQUIREMENTS

SR 3.4.8.1

This SR requires verification that the required loop is in operation and circulating reactor coolant. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.4.8.2

Verification that each required pump is OPERABLE ensures that an additional 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 each required pump. Alternatively, verification that a pump is in operation also verifies proper breaker alignment and power availability.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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

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REFERENCES

None.

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## B 3.4 REACTOR COOLANT SYSTEM (RCS)

### B 3.4.9 Pressurizer

#### BASES

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

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

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

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

## BASES

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### APPLICABLE SAFETY ANALYSES

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

Safety analyses presented in the UFSAR (Ref. 1) do not take credit for pressurizer heater operation; however, an implicit initial condition assumption of the safety analyses is that the RCS is operating at normal pressure.

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

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### LCO

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

The LCO requires two groups of OPERABLE pressurizer heaters, each with a capacity  $\geq 150$  kW. The minimum heater capacity required is sufficient to provide assurance that the heaters can be energized during a loss of offsite power condition to provide adequate subcooling margin in the RCS to maintain natural circulation conditions in MODE 3.

## BASES

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**APPLICABILITY** The need for pressure control is most pertinent when core heat can cause the greatest effect on RCS temperature, resulting in the greatest effect on pressurizer level and RCS pressure control. Thus, applicability has been designated for MODES 1 and 2. The applicability is also provided for MODE 3. 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.

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

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**ACTIONS** A.1, A.2, A.3, and A.4

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

If the pressurizer water level is not within the limit, action must be taken to bring the plant to a MODE in which the LCO does not apply. To achieve this status, within 6 hours the unit must be brought to MODE 3 with all rods fully inserted and incapable of withdrawal. Additionally, the unit must be brought to MODE 4 within 12 hours. This takes the unit out of the applicable MODES.

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.

### 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 the anticipation that a demand caused by loss of offsite power would be unlikely in this period. Pressure control may be maintained during this time using pressurizer control heaters.

## BASES

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### ACTIONS (continued)

#### C.1 and C.2

If one group of pressurizer heaters are inoperable and cannot be restored in 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 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

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### SURVEILLANCE REQUIREMENTS

#### SR 3.4.9.1

This SR requires that during steady state operation, pressurizer level is maintained below the nominal upper limit to provide a minimum space for a steam bubble. The Surveillance is performed by observing the indicated level on the narrow range instrumentation.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

#### SR 3.4.9.2

The SR 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 specified capacity. This may be done by measuring circuit current and voltage to calculate kW capacity.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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### REFERENCES

1. UFSAR, Section 15.1.
  2. NUREG-0737, November 1980.
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## B 3.4 REACTOR COOLANT SYSTEM (RCS)

### B 3.4.10 Pressurizer Safety Valves

#### BASES

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

Because the safety valves are totally enclosed and self actuating, they are considered independent components. The relief capacity for each valve, 420,000 lb/hr, is based on postulated overpressure transient conditions resulting from a complete loss of steam flow to the turbine. This event results in the maximum surge rate into the pressurizer, which specifies the minimum relief capacity for the safety valves. The discharge flow from the pressurizer safety valves is directed to the pressurizer relief tank. This discharge flow is indicated by an increase in temperature downstream of the pressurizer safety valves or increase in the pressurizer relief tank temperature or level.

Overpressure protection is required in MODES 1, 2, 3, 4, and 5; however, in MODE 4, MODE 5, and MODE 6 with the reactor vessel head on, overpressure protection is provided by operating procedures and by meeting the requirements of LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."

The upper and lower pressure limits are based on the  $\pm 3\%$  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 (nominal operating temperature and pressure). 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 American Society of Mechanical Engineers (ASME) pressure limit (Ref. 1) could include damage to RCS components, increased leakage, or a requirement to perform additional stress analyses prior to resumption of reactor operation.

BASES

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APPLICABLE  
SAFETY  
ANALYSES

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

- a. Uncontrolled rod withdrawal from full power,
- b. Loss of reactor coolant flow,
- c. Loss of external electrical load,
- d. Loss of normal feedwater,
- e. Loss of all AC power to station auxiliaries, and
- f. Locked rotor.

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

Pressurizer safety valves satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

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LCO

The three pressurizer safety valves are set to open at the RCS design pressure (2485 psig), and within the ASME specified tolerance, to avoid exceeding the maximum design pressure SL, to maintain accident analyses assumptions, and to comply with ASME requirements. The upper and lower pressure tolerance limits are based on the  $\pm 3\%$  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 more valves could result in exceeding the SL if a transient were to occur. The consequences of exceeding the ASME pressure limit could include damage to one or more RCS components, increased leakage, or additional stress analysis being required prior to resumption of reactor operation.

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APPLICABILITY

In MODES 1, 2, and 3 OPERABILITY of three 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 is conservatively included, although the listed accidents may not require the safety valves for protection.

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## BASES

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### APPLICABILITY (continued)

The LCO is not applicable in MODE 4 or in MODE 5 because LTOP is provided. Overpressure protection is not required in MODE 6 with reactor vessel head detensioned.

The Note allows entry into MODE 3 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 54 hour exception is based on 18 hour outage time for each of the three valves. The 18 hour period is derived from operating experience that hot testing can be performed in this timeframe.

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### ACTIONS

#### A.1

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

#### B.1 and B.2

If the Required Action of A.1 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 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. In MODE 4, overpressure protection is provided by the LTOP System. The change from MODE 1, 2, or 3 to MODE 4 reduces the RCS energy (core power and pressure), lowers the potential for large pressurizer insurges, and thereby removes the need for overpressure protection by three pressurizer safety valves.

BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.4.10.1

SRs are specified in the Inservice Testing Program. Pressurizer safety valves are to be tested in accordance with the requirements of the ASME Code (Ref. 4), which provides the activities and Frequencies necessary to satisfy the SRs. No additional requirements are specified.

The pressurizer safety valve setpoint is  $\pm 3\%$  of 2485 psig for OPERABILITY; however, the valves are reset to  $\pm 1\%$  during the Surveillance to allow for drift.

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REFERENCES

1. ASME, Boiler and Pressure Vessel Code, Section III.
  2. UFSAR, Chapter 15.
  3. WCAP-7769, Rev. 1, June 1972.
  4. ASME Code for Operation and Maintenance of Nuclear Power Plants.
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## B 3.4 REACTOR COOLANT SYSTEM (RCS)

### B 3.4.11 Pressurizer Power Operated Relief Valves (PORVs)

#### BASES

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

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

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

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

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

The plant has two PORVs, each having a relief capacity of 210,000 lb/hr at 2335 psig. The functional design of the PORVs is based on maintaining pressure below the Pressurizer Pressure - High reactor trip setpoint following a step reduction of 50% of full load with steam dump. In addition, the PORVs minimize challenges to the pressurizer safety valves and also may be used for low temperature overpressure protection (LTOP). See LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."

BASES

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APPLICABLE  
SAFETY  
ANALYSES

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

The PORVs are also modeled in safety analyses for events that result in increasing RCS pressure for which departure from nucleate boiling ratio (DNBR) criteria are critical (Ref. 2). By assuming PORV actuation, the primary pressure remains below the high pressurizer pressure trip setpoint; thus, the DNBR calculation is more conservative. As such, this actuation is not required to mitigate these events, and PORV automatic operation is, therefore, not an assumed safety function.

Pressurizer PORVs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

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LCO

The LCO requires the PORVs and their associated block valves to be OPERABLE for manual operation to mitigate the effects associated with an SGTR.

By maintaining two PORVs and their associated block valves OPERABLE, the single failure criterion is satisfied. An OPERABLE block valve may be either open and energized with the capability to be closed, or closed and energized with the capability to be opened, since the required safety function is accomplished by manual operation. Although typically open to allow PORV operation, the block valves may be OPERABLE when closed to isolate the flow path of an inoperable PORV that is capable of being manually cycled (e.g., as in the case of excessive PORV leakage). Similarly, isolation of an OPERABLE PORV does not render that PORV or block valve inoperable provided the relief function remains available with manual action.

An OPERABLE PORV is required to be capable of manually opening and closing, and not experiencing excessive seat leakage. Excessive seat leakage, although not associated with a specific acceptance criteria, exists when conditions dictate closure of the block valve to limit leakage.

Satisfying the LCO helps minimize challenges to fission product barriers.

## BASES

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**APPLICABILITY** In MODES 1, 2, and 3, the PORV and its block valve are required to be OPERABLE to limit the potential for a small break LOCA through the flow path. The most likely cause for a PORV small break LOCA is a result of a pressure increase transient that causes the PORV to open. Imbalances in the energy output of the core and heat removal by the secondary system can cause the RCS pressure to increase to the PORV opening setpoint. The most rapid increases will occur at the higher operating power and pressure conditions of MODES 1 and 2. The PORVs are also required to be OPERABLE in MODES 1, 2, and 3 for manual actuation to mitigate a steam generator tube rupture event.

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

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**ACTIONS** A Note has been added to clarify that all pressurizer PORVs and block valves are treated as separate entities, each with separate Completion Times (i.e., the Completion Time is on a component basis).

### A.1

PORVs may be inoperable and capable of being manually cycled (e.g., excessive seat leakage). In this condition, the associated block valve is required to be closed within 1 hour, but power must be maintained to the associated block valve, since removal of power would render the block valve inoperable.

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 has shown that minor problems can be corrected or closure accomplished in this time period.

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

If one PORV is inoperable and not capable of being manually cycled, it must be either restored, or isolated by closing the associated block valve and removing the power to the associated block valve. The Completion Times of 1 hour are reasonable, based on challenges to the PORVs during this time period, and provide 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

## BASES

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### ACTIONS (continued)

least one PORV that remains OPERABLE, an additional 72 hours is provided to restore the inoperable PORV to OPERABLE status. If the PORV cannot be restored within this additional time, the plant must be brought to a MODE in which the LCO does not apply, as required by Condition D.

#### C.1 and C.2

If one block valve is inoperable, then it is necessary to either restore the block valve(s) to OPERABLE status within the Completion Time of 1 hour or place the associated PORV in manual control. The prime importance for the capability to close the block valve is to isolate a stuck open PORV. Therefore, if the block valve(s) cannot be restored to OPERABLE status within 1 hour, the Required Action is to place the PORV in manual control to preclude its automatic opening for an overpressure event and to avoid the potential for a stuck open PORV at a time that the block valve is inoperable. The Completion Time of 1 hour is reasonable, based on the small potential for challenges to the system during this time period, and provides the operator time to correct the situation. Because at least one PORV remains OPERABLE, the operator is permitted 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 may not be capable of mitigating an event if the inoperable block valve is not full open. If the block valve is restored within the Completion Time of 72 hours, the PORV may be restored to automatic operation. If it cannot be restored within this additional time, the plant must be brought to a MODE in which the LCO does not apply, as required by Condition D.

The Required Actions C.1 and C.2 are modified by a Note stating that the Required Actions do not apply if the sole reason for the block valve being declared inoperable is as a result of power being removed to comply with other Required Actions. In this event, the Required Actions for inoperable PORV(s) (which require the block valve power to be removed once it is closed) are adequate to address the condition. While it may be desirable to also place the PORV(s) in manual control, this may not be possible for all causes of Condition B or E entry with PORV(s) inoperable and not capable of being manually cycled (e.g., as a result of failed control power fuse(s) or control switch malfunctions(s)).

## BASES

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### ACTIONS (continued)

#### D.1 and D.2

If the Required Action of Condition A, B, or C is not met, 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 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. In MODES 4 and 5, automatic PORV OPERABILITY may be required. See LCO 3.4.12.

#### E.1, E.2, E.3, and E.4

If two PORVs are 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 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 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. In MODES 4 and 5, automatic PORV OPERABILITY may be required. See LCO 3.4.12.

#### F.1

If two block valve(s) are inoperable, it is necessary to restore at least one block valve within 1 hour. The Completion Time is reasonable, based on the small potential for challenges to the system during this time and provides the operator time to correct the situation.

Required Action F.1 is modified by a Note stating that the Required Action does not apply if the sole reason for the block valve being declared inoperable is a result of power being removed to comply with other Required Actions. In this event, the Required Actions for inoperable PORV(s) (which require the block valve power to be removed once it is closed) are adequate to address the condition. While it may be desirable to also place the PORV(s) in manual control, this may not be possible for all causes of Condition B or E entry with PORV(s) inoperable and not

BASES

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ACTIONS (continued)

capable of being manually cycled (e.g., as a result of failed control power fuse(s) or control switch malfunctions(s)).

G.1 and G.2

If the Required Action of Condition F is not met, 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 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. In MODES 4 and 5, automatic PORV OPERABILITY may be required. See LCO 3.4.12.

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SURVEILLANCE  
REQUIREMENTS

SR 3.4.11.1

Block valve cycling verifies that the valve(s) can be opened and closed if needed.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

This SR is modified by two Notes. Note 1 modifies this SR by stating that it is not required to be performed with the block valve closed in accordance with the Required Actions of this LCO. Opening the block valve in this condition increases the risk of an unisolable leak from the RCS since the PORV is already inoperable. Note 2 modifies this SR to allow entry into and operation in MODE 3 prior to performing the SR. This allows the test to be performed in MODE 3 under operating temperature and pressure conditions, prior to entering MODE 1 or 2. In accordance with Reference 4, administrative controls require this test be performed in MODE 3 or 4 to adequately simulate operating temperature and pressure effects on PORV operation.

SR 3.4.11.2

SR 3.4.11.2 requires a complete cycle of each PORV. Operating a PORV through one complete cycle ensures that the PORV can be manually actuated for mitigation of an SGTR.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

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SURVEILLANCE REQUIREMENTS (continued)

The Note modifies this SR to allow entry into and operation in MODE 3 prior to performing the SR. This allows the test to be performed in MODE 3 under operating temperature and pressure conditions, prior to entering MODE 1 or 2. In accordance with Reference 4, administrative controls require this test be performed in MODE 3, 4, or 5 with a steam bubble in the pressurizer to adequately simulate operating temperature and pressure effects on PORV operation.

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REFERENCES

1. Regulatory Guide 1.32, February 1977.
  2. UFSAR, Section 15.2.
  3. ASME Code for Operation and Maintenance of Nuclear Power Plants.
  4. Letter from R.W. Hernan (NRC Staff) to J.A. Scalice (TVA), "Issuance of Technical Specification Amendments for Sequoyah Nuclear Plant, Units 1 and 2 (TAC Nos. MA2164 and MA2169) (TS 98-01)," dated November 19, 1998 (ADAMS Accession No. ML013320468).
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## B 3.4 REACTOR COOLANT SYSTEM (RCS)

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

#### BASES

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**BACKGROUND** 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. The PTLR provides the maximum allowable actuation logic setpoints for the power operated relief valves (PORVs) and the maximum RCS pressure for the existing RCS cold leg temperature during cooldown, shutdown, and heatup to meet the Reference 1 requirements during the LTOP MODES.

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

The potential for vessel overpressurization is most acute when the RCS is water solid, occurring only 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, "RCS Pressure and Temperature (P/T) Limits," requires administrative control of RCS pressure and temperature during heatup and cooldown to prevent exceeding the PTLR limits.

This LCO provides RCS overpressure protection by having a minimum coolant input capability and having adequate pressure relief capacity. Limiting coolant input capability requires all safety injection pumps and all but one centrifugal charging pump incapable of injection into the RCS and isolating the accumulators. The pressure relief capacity requires either two redundant PORVs or a depressurized RCS and an RCS vent of sufficient size. One PORV or the open RCS vent is the overpressure protection device that acts to terminate an increasing pressure event.

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

## BASES

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### BACKGROUND (continued)

decay heat levels, the makeup system can provide adequate flow via the makeup control valve. If conditions require the use of more than one 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 a depressurized RCS and an RCS vent of sufficient size. Two PORVs are required for redundancy. One PORV has adequate relieving capability to keep from overpressurization for the required coolant input capability.

#### PORV Requirements

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

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

The PTLR 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 in the PTLR ensures that the Reference 1 limits will not be exceeded in any analyzed event.

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

BASES

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BACKGROUND (continued)

RCS Vent Requirements

Once the RCS is depressurized, a vent exposed to the containment atmosphere will maintain the RCS at 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 flow capacity requirement, it requires an RCS vent opening of a least three square inches. This may be accomplished by 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.

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APPLICABLE  
SAFETY  
ANALYSES

Safety analyses (Ref. 3) demonstrate that the reactor vessel is adequately protected against exceeding the Reference 1 P/T limits. In MODES 1, 2, and 3 the pressurizer safety valves will prevent RCS pressure from exceeding the Reference 1 limits. At about the LTOP arming temperature specified in the PTLR and below, overpressure prevention falls to two 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 PTLR curves are revised, the LTOP System must be re-evaluated to ensure its functional requirements can still be met using the PORV method or the depressurized and vented RCS condition.

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

BASES

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

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

- a. Rendering all safety injection pumps and all but one charging pump incapable of injection,
- b. Deactivating the accumulator discharge isolation valves in their closed positions, and
- c. Disallowing start of an RCP unless a steam bubble exists in the pressurizer or the secondary side water temperature of each SG is  $\leq 25^{\circ}\text{F}$  above each of the RCS cold leg temperatures. LCO 3.4.6, "RCS Loops – MODE 4," and LCO 3.4.7, "RCS Loops – MODE 5, Loops Filled," provides this protection.

## BASES

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### APPLICABLE SAFETY ANALYSES (continued)

The Reference 3 analyses demonstrate that either one PORV or the depressurized RCS and RCS vent can maintain RCS pressure below limits when only one charging pump is actuated. Thus, the LCO allows only one charging pump OPERABLE during the LTOP MODES. Since neither one PORV nor the RCS vent can handle the pressure transient need from accumulator injection, when RCS temperature is low, the LCO also requires the accumulator's 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 accumulators must have their discharge valves closed and the valve power supply breakers fixed in their open positions.

Fracture mechanics analyses established the temperature of LTOP Applicability at the LTOP arming temperature specified in the PTLR.

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 charging pump OPERABLE and SI actuation available.

#### PORV Performance

The fracture mechanics analyses show that the vessel is protected when the PORVs are set to open at or below the limit shown in the PTLR. The setpoints are derived by analyses that model the performance of the LTOP System, assuming the limiting LTOP transient of one charging pump injecting into the RCS. These analyses consider pressure overshoot and undershoot beyond the PORV opening and closing, resulting from signal processing and valve stroke times. The PORV setpoints at or below the derived limit ensures the Reference 1 P/T limits will be met.

The PORV setpoints in the PTLR will be updated when the revised P/T limits conflict with the LTOP analysis limits. The P/T limits are periodically modified as the reactor vessel material toughness decreases due to neutron embrittlement caused by neutron irradiation. Revised 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 is assumed to represent the worst case, single active failure.

BASES

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APPLICABLE SAFETY ANALYSES (continued)

RCS Vent Performance

With the RCS depressurized, analyses show a vent size of 3.0 square inches is capable of mitigating the allowed LTOP overpressure transient. The capacity of a vent this size is greater than the flow of the limiting transient for the LTOP configuration, one charging pump OPERABLE, maintaining RCS pressure less than the maximum pressure on the P/T limit curve.

The RCS vent size will 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.

The LTOP System satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

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LCO

This LCO requires 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 that no safety injection pumps and a maximum of one charging pump be capable of injecting into the RCS, and all accumulator discharge isolation valves be closed and immobilized (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 LCO is modified by three Notes. Note 1 allows two charging pumps to be made capable of injecting for  $\leq 1$  hour during pump swap operations. One hour provides sufficient time to safely complete the actual transfer and to complete the administrative controls and Surveillance Requirements associated with the swap. The intent is to minimize the actual time that more than one charging pump is physically capable of injection. Note 2 states that accumulator isolation is only required when the accumulator pressure is more than or at the maximum RCS pressure for the existing temperature, as allowed by the P/T limit curves. This Note permits the accumulator discharge isolation valve Surveillance to be performed only under these pressure and temperature conditions. Note 3 allows a 4 hour maximum time period for rendering both safety injection and one centrifugal charging pump inoperable after entry in MODE 4 from MODE 3. RCS temperature must remain above 325°F until the pumps are rendered incapable of inadvertent injection.

## BASES

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### LCO (continued)

The 4 hour time period is sufficient for completing this activity and is based on low probability for inadvertent pump start.

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

- a. Two OPERABLE PORVs,

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

- b. A depressurized RCS and an RCS vent.

An RCS vent is OPERABLE when open with an area of  $\geq 3.0$  square inches.

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

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### APPLICABILITY

This LCO is applicable in MODE 4 when any RCS cold leg temperature is  $\leq$  the LTOP arming temperature specified in the PTLR, 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 the LTOP arming temperature specified in the PTLR. 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.

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

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### ACTIONS

A Note prohibits the application of LCO 3.0.4.b to an inoperable LTOP System. There is an increased risk associated with entering MODE 4 from MODE 5 with LTOP inoperable and the provisions of LCO 3.0.4.b, which allow entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, should not be applied in this circumstance.

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## BASES

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### ACTIONS (continued)

#### A.1 and B.1

With any safety injection pump or more than one charging pump capable of injecting into the RCS, RCS overpressurization is possible.

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

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

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

If isolation is needed and cannot be accomplished in 1 hour, Required Action D.1 and Required Action D.2 provide two options, either of which must be performed in the next 12 hours. By increasing the RCS temperature to > the LTOP arming temperature specified in the PTLR, an accumulator pressure of 600 psig cannot exceed the LTOP limits if the accumulators are fully injected. Depressurizing the accumulators below the LTOP limit from the PTLR also gives this protection.

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

#### E.1

In MODE 4 when any RCS cold leg temperature is  $\leq$  the LTOP arming temperature specified in the PTLR, with one required PORV inoperable, the PORV must be restored to OPERABLE status within a Completion Time of 7 days. Two PORVs are required to provide low temperature overpressure mitigation while withstanding a single failure of an active component.

The Completion Time considers the facts that only one of the PORVs 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.

## BASES

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### ACTIONS (continued)

#### F.1

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

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

#### G.1

The RCS must be depressurized and a vent must be established within 12 hours when:

- a. Both required PORVs are inoperable,
- b. A Required Action and associated Completion Time of Condition A, B, D, E, or F is not met, or
- c. The LTOP System is inoperable for any reason other than Condition A, B, C, D, E, or F.

The vent must be sized  $\geq 3.0$  square inches to ensure that the flow capacity is greater than that required for the worst case mass input transient reasonable during the applicable MODES. This action is needed to protect the RCPB from a low temperature overpressure event and a possible brittle failure of the reactor vessel.

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

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### 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, no safety injection pumps and a maximum of one charging pump are verified incapable of injecting into the RCS and the accumulator discharge isolation valves are verified closed and locked out.

## BASES

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### SURVEILLANCE REQUIREMENTS (continued)

The SI pumps and charging pump are rendered incapable 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 Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

The additional frequency for SR 3.4.12.1 and SR 3.4.12.2 is necessary to allow time during the transition from MODE 3 to MODE 4 to make the pumps inoperable.

#### SR 3.4.12.4

The RCS vent of  $\geq 3.0$  square inches is proven OPERABLE by verifying its open condition.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

The passive vent path arrangement must only be open to be OPERABLE. This Surveillance is required to be met if the vent is being used to satisfy the pressure relief requirements of the LCO 3.4.12b.

#### SR 3.4.12.5

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

The block valve is a remotely controlled, motor operated valve. The power to the valve operator is not required removed, and the manual operator is not required locked in the inactive position. Thus, the block valve can be closed in the event the PORV develops excessive leakage or does not close (sticks open) after relieving an overpressure situation.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.12.6

Performance of a COT is required within 12 hours after decreasing RCS temperature to  $\leq$  the LTOP arming temperature specified in the PTLR on each required PORV to verify and, as necessary, adjust its lift setpoint. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable COT of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. The COT will verify the setpoint is within the PTLR allowed maximum limits in the PTLR. PORV actuation could depressurize the RCS and is not required.

The 12 hour Frequency considers the unlikelihood of a low temperature overpressure event during this time.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

A Note has been added indicating that this SR is required to be performed 12 hours after decreasing RCS cold leg temperature to  $\leq$  the LTOP arming temperature specified in the PTLR. The COT cannot be performed until in the LTOP MODES when the PORV lift setpoint can be reduced to the LTOP setting. The test must be performed within 12 hours after entering the LTOP MODES.

SR 3.4.12.7

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

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

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- REFERENCES
1. 10 CFR 50, Appendix G.
  2. Generic Letter 88-11.
  3. UFSAR, Chapter 15.
  4. 10 CFR 50, Section 50.46.
  5. 10 CFR 50, Appendix K.
  6. Generic Letter 90-06.
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## B 3.4 REACTOR COOLANT SYSTEM (RCS)

### B 3.4.13 RCS Operational LEAKAGE

#### BASES

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

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

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

The safety significance of RCS LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring reactor coolant LEAKAGE into the containment area 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 that is 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 analyses radiation release assumptions from being exceeded. The consequences of violating this LCO include the possibility of a loss of coolant accident (LOCA).

BASES

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APPLICABLE  
SAFETY  
ANALYSES

Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event. The safety analysis for an event resulting in steam discharge to the atmosphere assumes 150 gallons per day (gpd) per steam generator (i.e., a total of 0.4 gpm).

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 UFSAR (Ref. 3) analysis for SGTR assumes the contaminated secondary fluid is only briefly released via the atmospheric relief valve for the affected steam generator. The 0.4 gpm operational primary to secondary leakage from all four steam generators is relatively inconsequential.

The safety analysis for the SLB accident assumes the 150 gpd primary to secondary LEAKAGE is through the affected generator as an initial condition. The dose consequences resulting from the SLB accident are well within the limits defined in 10 CFR 100 or the staff approved licensing basis (i.e., a small fraction of these limits).

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

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LCO

RCS operational LEAKAGE shall be limited to:

a. Pressure Boundary LEAKAGE

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

b. Unidentified LEAKAGE

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

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BASES

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LCO (continued)

c. Identified LEAKAGE

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

d. Primary to Secondary LEAKAGE Through Any One SG

The limit of 150 gallons per day per SG is based on the operational LEAKAGE performance criterion in NEI 97-06, Steam Generator Program Guidelines (Ref. 4). The Steam Generator Program operational LEAKAGE performance criterion in NEI 97-06 states, "The RCS operational primary to secondary leakage through any one SG shall be limited to 150 gallons per day." The limit is based on operating experience with SG tube degradation mechanisms that result in tube leakage. The operational leakage rate criterion in conjunction with the implementation of the Steam Generator Program is an effective measure for minimizing the frequency of steam generator tube ruptures.

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APPLICABILITY

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

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

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

BASES

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ACTIONS

A.1

Unidentified LEAKAGE or identified 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 primary to secondary LEAKAGE is not within limit, or if unidentified or identified 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. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. The reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. This action reduces the LEAKAGE and also reduces the factors that tend to degrade the pressure boundary.

The 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. In MODE 5, the pressure stresses acting on the RCPB are much lower, and further deterioration is much less likely.

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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. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. Unidentified LEAKAGE and identified LEAKAGE are determined by performance of an RCS water inventory balance.

The RCS water inventory balance must be met with the reactor at steady state operating conditions (stable temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows). The Surveillance is modified by two Notes. Note 1 states that this SR is not required to be performed until 12 hours after establishing steady state operation. The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established.

## BASES

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### SURVEILLANCE REQUIREMENTS (continued)

Steady state operation is required to perform a proper inventory balance since calculations during maneuvering are not useful. For RCS operational LEAKAGE determination by water inventory balance, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP 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. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. These leakage detection systems are specified in LCO 3.4.15, "RCS Leakage Detection Instrumentation."

Note 2 states that this SR is not applicable to primary to secondary LEAKAGE because LEAKAGE of 150 gallons per day cannot be measured accurately by an RCS water inventory balance.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

#### SR 3.4.13.2

This SR verifies that primary to secondary LEAKAGE is less or equal to 150 gallons per day through any one SG. Satisfying the primary to secondary LEAKAGE limit ensures that the operational LEAKAGE performance criterion in the Steam Generator Program is met. If this SR is not met, compliance with LCO 3.4.17, "Steam Generator Tube Integrity," should be evaluated. The 150 gallons per day limit is measured at ambient temperature as described in Reference 5. The operational LEAKAGE rate limit applies to LEAKAGE through any one SG. If it is not practical to assign the LEAKAGE to an individual SG, all the primary to secondary LEAKAGE should be conservatively assumed to be from one SG.

The Surveillance is modified by a Note which states that the Surveillance is not required to be performed until 12 hours after establishment of steady state operation. For RCS primary to secondary LEAKAGE determination, 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.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

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SURVEILLANCE REQUIREMENTS (continued)

The primary to secondary LEAKAGE is determined using continuous process radiation monitors or radiochemical grab sampling in accordance with the EPRI guidelines (Ref. 5).

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REFERENCES

1. 10 CFR 50, Appendix A, GDC 30.
  2. Regulatory Guide 1.45, May 1973.
  3. UFSAR, Section 15.4.3.
  4. NEI 97-06, "Steam Generator Program Guidelines."
  5. EPRI, "Pressurized Water Reactor Primary-to-Secondary Leak Guidelines."
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## B 3.4 REACTOR COOLANT SYSTEM (RCS)

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

#### BASES

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

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

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

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

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

- a. Residual Heat Removal (RHR) System,

BASES

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BACKGROUND (continued)

- b. Safety Injection System, and
- c. Chemical and Volume Control System.

The PIVs are listed in Table B 3.4.14-1

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

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APPLICABLE  
SAFETY  
ANALYSES

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

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

RCS PIV leakage satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

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LCO

RCS PIV leakage is identified LEAKAGE into closed systems connected to the RCS. Isolation valve leakage is usually on the order of drops per minute. Leakage that increases significantly suggests that something is operationally wrong and corrective action must be taken.

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

## BASES

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### LCO (continued)

Reference 6 permits leakage testing at a lower pressure differential than between the specified maximum RCS pressure and the normal pressure of the connected system during RCS operation (the maximum pressure differential) in those types of valves in which the higher service pressure will tend to diminish the overall leakage channel opening. In such cases, the observed rate may be adjusted to the maximum pressure differential by assuming leakage is directly proportional to the pressure differential to the one half power.

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### APPLICABILITY

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

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

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### ACTIONS

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

#### A.1 and A.2

The flow path must be isolated by two valves. Required Actions A.1 and A.2 are modified by a Note that the valves used for isolation must meet the same leakage requirements as the PIVs and must be within the RCPB.

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

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

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## BASES

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### ACTIONS (continued)

#### B.1 and B.2

If Required Actions and associated Completion Times of Condition A are not met, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. This Action may reduce the leakage and also reduces the potential for a LOCA outside the containment. The allowed Completion Times are reasonable based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

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### SURVEILLANCE REQUIREMENTS

#### SR 3.4.14.1

Performance of leakage testing on each RCS PIV or isolation valve used to satisfy Required Action A.1 and Required Action 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, if the plant does not go into MODE 5 for at least 7 days. The Frequency is consistent with 10 CFR 50.55.a(g) (Ref. 7) as contained in the Inservice Testing Program, is within frequency allowed by the American Society of Mechanical Engineers (ASME) Code (Ref.6), and is based on the need to perform such surveillances under the conditions that apply during an outage and the potential for an unplanned transient if the surveillance were performed with the reactor at power.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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

## BASES

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### SURVEILLANCE REQUIREMENTS (continued)

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 complementary 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 RHR System when the RHR System is aligned to the RCS in the shutdown cooling mode of operation. PIVs contained in the RHR shutdown cooling flow path must be leakage rate tested after RHR is secured and stable unit conditions and the necessary differential pressures are established.

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### 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. ASME Code for Operation and Maintenance of Nuclear Power Plants.
  7. 10 CFR 50.55a(g).
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Table B 3.4.14-1 (page 1 of 1)  
Reactor Coolant System Pressure Isolation Valves

<u>VALVE NUMBER</u>	<u>FUNCTION</u>
63-560	Accumulator Discharge
63-561	Accumulator Discharge
63-562	Accumulator Discharge
63-563	Accumulator Discharge
63-622	Accumulator Discharge
63-623	Accumulator Discharge
63-624	Accumulator Discharge
63-625	Accumulator Discharge
63-551	Safety Injection (Cold Leg)
63-553	Safety Injection (Cold Leg)
63-557	Safety Injection (Cold Leg)
63-555	Safety Injection (Cold Leg)
63-632	Residual Heat Removal (Cold Leg)
63-633	Residual Heat Removal (Cold Leg)
63-634	Residual Heat Removal (Cold Leg)
63-635	Residual Heat Removal (Cold Leg)
63-641	Residual Heat Removal/Safety Injection (Hot Leg)
63-644	Residual Heat Removal/Safety Injection (Hot Leg)
63-558	Safety Injection (Hot Leg)
63-559	Safety Injection (Hot Leg)
63-543	Safety Injection (Hot Leg)
63-545	Safety Injection (Hot Leg)
63-547	Safety Injection (Hot Leg)
63-549	Safety Injection (Hot Leg)
63-640	Residual Heat Removal (Hot Leg)
63-643	Residual Heat Removal (Hot Leg)
FCV-74-1	Residual Heat Removal
FCV-74-2	Residual Heat Removal

## B 3.4 REACTOR COOLANT SYSTEM (RCS)

### B 3.4.15 RCS Leakage Detection Instrumentation

#### BASES

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**BACKGROUND** GDC 30 of Appendix A to 10 CFR 50 (Ref. 1) requires means for detecting and, to the extent practical, identifying the location of the source of RCS LEAKAGE. Regulatory Guide 1.45, Revision 0, (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.

The containment pocket sump used to collect unidentified LEAKAGE is instrumented to alarm for increases above the normal flow rates.

The reactor coolant contains radioactivity that, when released to the containment, may be detected by radiation monitoring instrumentation. A radioactivity detection system is included for monitoring particulate activity because of its sensitivity and rapid response to RCS LEAKAGE.

Other indications may be used to detect an increase in unidentified LEAKAGE; however, they are not required to be OPERABLE by this LCO. An increase in humidity of the containment atmosphere would indicate release of water vapor to the containment. Dew point temperature measurements can thus be used to monitor humidity levels of the containment atmosphere as an indicator of potential RCS LEAKAGE.

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 pocket sump. 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.

BASES

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BACKGROUND (continued)

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 is 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.

The above-mentioned LEAKAGE detection methods or systems differ in sensitivity and response time. Some of these systems could serve as early alarm systems signaling the operators that closer examination of other detection systems is necessary to determine the extent of any corrective action that may be required.

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APPLICABLE  
SAFETY  
ANALYSES

The need to evaluate the severity of an alarm or an indication is important to the operators, and the ability to compare and verify with indications from other systems is necessary.

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

RCS leakage detection instrumentation satisfies Criterion 1 of 10 CFR 50.36(c)(2)(ii).

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LCO

This LCO requires instruments of diverse monitoring principles to be OPERABLE to provide confidence that small amounts of unidentified LEAKAGE are detected in time to allow actions to place the plant in a safe condition, when RCS LEAKAGE indicates possible RCPB degradation.

The LCO requires two instruments to be OPERABLE.

The containment pocket sump is used to collect unidentified LEAKAGE. The monitor on the containment pocket sump detects level and is instrumented to detect when there is an increase above the normal value by 1 gpm. The identification of an increase in unidentified LEAKAGE will be delayed by the time required for the unidentified LEAKAGE to travel to the containment pocket sump and it may take longer than one hour to detect a 1 gpm increase in unidentified LEAKAGE, depending on the

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BASES

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LCO (continued)

origin and magnitude of the LEAKAGE. This sensitivity is acceptable for containment pocket sump level monitor OPERABILITY.

The reactor coolant contains radioactivity that, when released to the containment, can be detected by the lower containment atmosphere particulate radioactivity monitor. A radioactivity detection system is included for monitoring particulate activity because of the sensitivity and rapid response to RCS LEAKAGE, but has recognized limitations. 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. If there are few fuel element cladding defects and low levels of activation products, it may not be possible for the lower containment atmosphere particulate radioactivity monitor to detect a 1 gpm increase within 1 hour during normal operation. However, the lower containment atmosphere particulate radioactivity monitor is OPERABLE when it is capable of detecting a 1 gpm increase in unidentified LEAKAGE within 1 hour given an RCS activity equivalent to that assumed in the design calculations for the monitor (Reference 3).

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

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APPLICABILITY

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

In MODE 5 or 6, the temperature is to be  $\leq 200^{\circ}\text{F}$  and pressure is maintained low or at atmospheric pressure. Since the temperatures and pressures are far lower than those for MODES 1, 2, 3, and 4, the likelihood of leakage and crack propagation are much smaller. Therefore, the requirements of this LCO are not applicable in MODES 5 and 6.

## BASES

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### ACTIONS

#### A.1 and A.2

With the required containment pocket sump level monitor inoperable, no other form of sampling can provide the equivalent information; however, the lower containment atmosphere particulate radioactivity monitor will provide indications of changes in leakage. Together with the lower containment atmosphere particulate radioactivity monitor, the periodic surveillance for RCS water inventory balance, SR 3.4.13.1, must be performed at an increased frequency of 24 hours to provide information that is adequate to detect leakage. A Note is added allowing that SR 3.4.13.1 is not required to be performed until 12 hours after establishing steady state operation (stable temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows). The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established.

Restoration of the required pocket sump level monitor to OPERABLE status within a Completion Time of 30 days is required to regain the function 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.

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

With the lower containment atmosphere particulate radioactivity monitoring instrumentation channel 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 water inventory balance performed every 24 hours, the reactor may be operated for up to 30 days to allow restoration of the required lower containment atmosphere particulate radioactivity monitors.

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

BASES

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ACTIONS (continued)

C.1 and C.2

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

D.1

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

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SURVEILLANCE  
REQUIREMENTS

SR 3.4.15.1

SR 3.4.15.1 requires the performance of a CHANNEL CHECK of the required lower containment atmosphere particulate radioactivity monitor. The check gives reasonable confidence that the channel is operating properly.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.4.15.2

SR 3.4.15.2 requires the performance of a COT on the required lower containment atmosphere particulate radioactivity monitor. The test ensures that the monitor can perform its function in the desired manner. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable COT of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. The test verifies the alarm setpoint and relative accuracy of the instrument string.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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BASES

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.15.3 and SR 3.4.15.4

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 Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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REFERENCES

1. 10 CFR 50, Appendix A, Section IV, GDC 30.
  2. Regulatory Guide 1.45, Revision 0, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.
  3. UFSAR, Section 5.2.7.
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## B 3.4 REACTOR COOLANT SYSTEM (RCS)

### B 3.4.16 RCS Specific Activity

#### BASES

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**BACKGROUND** The maximum dose that an individual at the exclusion area boundary can receive for 2 hours following an accident, or at the low population zone outer boundary for the radiological release duration, is specified in 10 CFR 100.11 (Ref. 1). Doses to control room operators must be limited per GDC 19. The limits on specific activity ensure that the offsite and control room doses are appropriately limited during analyzed transients and accidents.

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

The LCO contains specific activity limits for both DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133. The allowable levels are intended to ensure that offsite and control room doses meet the appropriate acceptance criteria in the Standard Review Plan (SRP) (Ref. 2).

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**APPLICABLE SAFETY ANALYSES** The LCO limits on the specific activity of the reactor coolant ensures that the resulting offsite and control room doses meet the appropriate SRP acceptance criteria following a SLB or SGTR accident. The safety analyses (Refs. 3 and 4) assume the specific activity of the reactor coolant is at the LCO limits, and an existing reactor coolant steam generator (SG) tube leakage rate of 150 gpd exists. The safety analyses assume the specific activity of the secondary coolant is at its limit of 0.1  $\mu\text{Ci/gm}$  DOSE EQUIVALENT I-131 from LCO 3.7.16, "Secondary Specific Activity."

The analyses for the SLB and SGTR accidents establish the acceptance limits for RCS specific activity. Reference to these analyses is used to assess changes to the unit that could affect RCS specific activity, as they relate to the acceptance limits.

The safety analyses consider two cases of reactor coolant iodine specific activity. One case assumes specific activity at 0.35  $\mu\text{Ci/gm}$  DOSE EQUIVALENT I-131 with a concurrent large iodine spike that increases the rate of release of iodine from the fuel rods containing cladding defects to the primary coolant immediately after the SLB (by a factor of 500), or SGTR (by a factor of 500), respectively. The second case assumes the

BASES

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APPLICABLE SAFETY ANALYSES (continued)

initial reactor coolant iodine activity at 21.0  $\mu\text{Ci/gm}$  DOSE EQUIVALENT I-131 due to an iodine spike caused by a reactor or an RCS transient prior to the accident. In both cases, the noble gas activity is assumed to be 1612.6  $\mu\text{Ci/gm}$  DOSE EQUIVALENT XE-133.

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

The loss of offsite power causes the steam dump valves to close to protect the condenser. The rise in pressure in the ruptured SG discharges radioactively contaminated steam to the atmosphere through the SG power operated relief valves and the main steam safety valves. The unaffected SGs remove core decay heat by venting steam to the atmosphere until the cooldown ends and the Residual Heat Removal (RHR) system is placed in service.

The SLB radiological analysis assumes that offsite power is lost at the same time as the pipe break occurs outside containment. Reactor trip occurs after the generation of an SI signal on low steam line pressure. The affected SG blows down completely and steam is vented directly to the atmosphere. The unaffected SGs remove core decay heat by venting steam to the atmosphere until the cooldown ends and the RHR system is placed in service.

Operation with iodine specific activity levels greater than the LCO limit is permissible, if the activity levels do not exceed 21.0  $\mu\text{Ci/gm}$  for more than 48 hours.

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

RCS specific activity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

BASES

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LCO The iodine specific activity in the reactor coolant is limited to 0.35  $\mu\text{Ci/gm}$  DOSE EQUIVALENT I-131, and the noble gas specific activity in the reactor coolant is limited to 1612.6  $\mu\text{Ci/gm}$  DOSE EQUIVALENT XE-133. The limits on specific activity ensure that offsite and control room doses will meet the appropriate SRP acceptance criteria (Ref. 2).

The SLB and SGTR accident analyses (Refs. 3 and 4) show that the calculated doses are within acceptable limits. Violation of the LCO may result in reactor coolant radioactivity levels that could, in the event of a SLB or SGTR, lead to doses that exceed the SRP acceptance criteria (Ref. 2).

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APPLICABILITY In MODES 1, 2, 3, and 4, operation within the LCO limits for DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133 is necessary to limit the potential consequences of a SLB or SGTR to within the SRP acceptance criteria (Ref. 2).

In MODES 5 and 6, the steam generators are not being used for decay heat removal, the RCS and steam generators are depressurized, and primary to secondary leakage is minimal. Therefore, the monitoring of RCS specific activity is not required.

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ACTIONS A.1 and A.2

With the DOSE EQUIVALENT I-131 greater than the LCO limit, samples at intervals of 4 hours must be taken to demonstrate that the specific activity is  $\leq 21.0 \mu\text{Ci/gm}$ . An isotopic analysis of a reactor coolant sample must be performed for at least I-131, I-133, and I-135. The Completion Time of 4 hours is required to obtain and analyze a sample. Sampling is continued every 4 hours to provide a trend.

The DOSE EQUIVALENT I-131 must be restored to within limits within 48 hours. The Completion Time of 48 hours is acceptable since it is expected that, if there were an iodine spike, the normal coolant iodine concentration would be restored within this time period. Also, there is a low probability of a SLB or SGTR occurring during this time period.

A Note permits the use of the provisions of LCO 3.0.4.c. This allowance permits entry into the applicable MODE(S), relying on Required Actions A.1 and A.2 while the DOSE EQUIVALENT I-131 LCO limit is not met. This allowance 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.

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## BASES

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### ACTIONS (continued)

#### B.1 and B.2

If a Required Action and the associated Completion Time of Condition A is not met, or if the DOSE EQUIVALENT I-131 is  $> 21.0 \mu\text{Ci/gm}$ , or if the DOSE EQUIVALENT XE-133 is  $> 1612.6 \mu\text{Ci/gm}$ , the reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

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### SURVEILLANCE REQUIREMENTS

#### SR 3.4.16.1

SR 3.4.16.1 requires performing a gamma isotopic analysis and calculating the DOSE EQUIVALENT XE-133 using the dose conversion factors in the DOSE EQUIVALENT XE-133 definition. 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 the noble gas specific activity.

Trending the results of this Surveillance allows proper remedial action to be taken before reaching the LCO limit under normal operating conditions.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

Due to the inherent difficulty in detecting Kr-85 in a reactor coolant sample due to masking from radioisotopes with similar decay energies, such as F-18 and I-134, it is acceptable to include the minimum detectable activity for Kr-85 in the SR 3.4.16.1 calculation. If a specific noble gas nuclide listed in the definition of DOSE EQUIVALENT XE-133 is not detected, it should be assumed to be present at the minimum detectable activity.

A Note modifies the SR to only require the surveillance to be performed in MODES 1, 2, and 3 with  $T_{\text{avg}} \geq 500^\circ\text{F}$ .

BASES

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.16.2

This Surveillance is performed to ensure iodine specific activity remains within the LCO limit during normal operation and following fast power changes when iodine spiking is more apt to occur.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

The Frequency, between 2 and 6 hours after a power change  $\geq 15\%$  RTP within a 1 hour period, is established because the iodine levels peak during this time following iodine spiking initiation; samples at other times would provide inaccurate results.

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REFERENCES

1. 10 CFR 100.11.
  2. Standard Review Plan (SRP) Section 15.1.5 Appendix A (SLB) and Section 15.6.3 (SGTR).
  3. UFSAR, Section 15.5.4.
  4. UFSAR, Section 15.5.5.
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## B 3.4 REACTOR COOLANT SYSTEM (RCS)

### B 3.4.17 Steam Generator (SG) Tube Integrity

#### BASES

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**BACKGROUND** Steam generator (SG) tubes are small diameter, thin walled tubes that carry primary coolant through the primary to secondary heat exchangers. The SG tubes have a number of important safety functions. Steam generator tubes are an integral part of the reactor coolant pressure boundary (RCPB) and, as such, are relied on to maintain the primary system's pressure and inventory. The SG tubes isolate the radioactive fission products in the primary coolant from the secondary system. In addition, as part of the RCPB, the SG tubes are unique in that they act as the heat transfer surface between the primary and secondary systems to remove heat from the primary system. This Specification addresses only the RCPB integrity function of the SG. The SG heat removal function is addressed 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," and LCO 3.4.7, "RCS Loops – MODE 5, Loops Filled."

SG tube integrity means that the tubes are capable of performing their intended RCPB safety function consistent with the licensing basis, including applicable regulatory requirements.

Steam generator tubing is subject to a variety of degradation mechanisms. Steam generator tubes may experience tube degradation related to corrosion phenomena, such as wastage, pitting, intergranular attack, and stress corrosion cracking, along with other mechanically induced phenomena such as denting and wear. These degradation mechanisms can impair tube integrity if they are not managed effectively. The SG performance criteria are used to manage SG tube degradation.

Specification 5.5.7, "Steam Generator (SG) Program," requires that a program be established and implemented to ensure that SG tube integrity is maintained. Pursuant to Specification 5.5.7, tube integrity is maintained when the SG performance criteria are met. There are three SG performance criteria: structural integrity, accident induced leakage, and operational LEAKAGE. The SG performance criteria are described in Specification 5.5.7. Meeting the SG performance criteria provides reasonable assurance of maintaining tube integrity at normal and accident conditions.

The processes used to meet the SG performance criteria are defined by the Steam Generator Program Guidelines (Ref. 1).

BASES

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APPLICABLE  
SAFETY  
ANALYSES

The steam generator tube rupture (SGTR) accident is the limiting design basis event for SG tubes and avoiding an SGTR is the basis for this Specification. The analysis of a SGTR event assumes a bounding primary to secondary LEAKAGE rate equal to the operational LEAKAGE rate limits in LCO 3.4.13, "RCS Operational LEAKAGE," plus the leakage rate associated with a double-ended rupture of a single tube. The accident analysis for a SGTR assumes the contaminated secondary fluid is only briefly released to the atmosphere via SG atmospheric relief valves and safety valves.

The analysis for design basis accidents and transients other than a SGTR assume the SG tubes retain their structural integrity (i.e., they are assumed not to rupture.) In these analyses, the steam discharge to the atmosphere is based on the total primary to secondary LEAKAGE from all SGs of 0.4 gallons per minute or is assumed to increase to 1 gallon per minute as a result of accident induced conditions. For accidents that do not involve fuel damage, the primary coolant activity level of DOSE EQUIVALENT I-131 is assumed to be equal to the LCO 3.4.16, "RCS Specific Activity," limits. For accidents that assume fuel damage, the primary coolant activity is a function of the amount of activity released from the damaged fuel. The dose consequences of these events are within the limits of GDC 19 (Ref. 2) and 10 CFR 100 (Ref. 3).

Steam generator tube integrity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

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LCO

The LCO requires that SG tube integrity be maintained. The LCO also requires that all SG tubes that satisfy the plugging criteria be plugged in accordance with the Steam Generator Program.

During an SG inspection, any inspected tube that satisfies the Steam Generator Program plugging criteria is removed from service by plugging. If a tube was determined to satisfy the plugging criteria but was not plugged, the tube may still have tube integrity.

In the context of this Specification, a SG tube is defined as the entire length of the tube, including the tube wall, between the tube-to-tubesheet weld at the tube inlet and the tube-to-tubesheet weld at the tube outlet. The tube-to-tubesheet weld is not considered part of the tube.

A SG tube has tube integrity when it satisfies the SG performance criteria. The SG performance criteria are defined in Specification 5.5.7, "Steam Generator Program," and describe acceptable SG tube performance. The Steam Generator Program also provides the evaluation process for determining conformance with the SG performance criteria.

## BASES

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### LCO (continued)

There are three SG performance criteria: structural integrity, accident induced leakage, and operational LEAKAGE. Failure to meet any one of these criteria is considered failure to meet the LCO.

The structural integrity performance criterion provides a margin of safety against tube burst or collapse under normal and accident conditions, and ensures structural integrity of the SG tubes under all anticipated transients included in the design specification. Tube burst is defined as, "The gross structural failure of the tube wall. The condition typically corresponds to an unstable opening displacement (e.g., opening area increased in response to constant pressure) accompanied by ductile (plastic) tearing of the tube material at the ends of the degradation." Tube collapse is defined as, "For the load displacement curve for a given structure, collapse occurs at the top of the load versus displacement curve where the slope of the curve becomes zero." The structural integrity performance criterion provides guidance on assessing loads that have a significant effect on burst or collapse. In that context, the term "significant" is defined as "An accident loading condition other than differential pressure is considered significant when the addition of such loads in the assessment of the structural integrity performance criterion could cause a lower structural limit or limiting burst/collapse condition to be established." For tube integrity evaluations, except for circumferential degradation, axial thermal loads are classified as secondary loads. For circumferential degradation, the classification of axial thermal loads as primary or secondary loads will be evaluated on a case-by-case basis. The division between primary and secondary classifications will be based on detailed analysis and/or testing.

Structural integrity requires that the primary membrane stress intensity in a tube not exceed the yield strength for all ASME Code, Section III, Service Level A (normal operating conditions) and Service Level B (upset or abnormal conditions) transients included in the design specification. This includes safety factors and applicable design basis loads based on ASME Code, Section III, Subsection NB (Ref. 4) and Draft Regulatory Guide 1.121 (Ref. 5).

The accident induced leakage performance criterion ensures that the primary to secondary LEAKAGE caused by a design basis accident, other than a SGTR, is within the accident analysis assumptions. The accident analyses assumptions are discussed in the Applicable Safety Analyses section. The accident induced leakage rate includes any primary to secondary LEAKAGE existing prior to the accident in addition to primary to secondary LEAKAGE induced during the accident.

BASES

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LCO (continued)

The operational LEAKAGE performance criterion provides an observable indication of SG tube conditions during plant operation. The limit on operational LEAKAGE is contained in LCO 3.4.13, "RCS Operational LEAKAGE," and limits primary to secondary LEAKAGE through any one SG to 150 gallons per day. This limit is based on the assumption that a single crack leaking this amount would not propagate to a SGTR under the stress conditions of a LOCA or a steam line break. If this amount of LEAKAGE is due to more than one crack, the cracks are very small, and the above assumption is conservative.

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APPLICABILITY

Steam generator tube integrity is challenged when the pressure differential across the tubes is large. Large differential pressures across SG tubes can only be experienced in MODE 1, 2, 3, or 4.

RCS conditions are far less challenging in MODES 5 and 6 than during MODES 1, 2, 3, and 4. In MODES 5 and 6, primary to secondary differential pressure is low, resulting in lower stresses and reduced potential for LEAKAGE.

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ACTIONS

The ACTIONS are modified by a Note clarifying that the Conditions may be entered independently for each SG tube. This is acceptable because the Required Actions provide appropriate compensatory actions for each affected SG tube. Complying with the Required Actions may allow for continued operation, and subsequent affected SG tubes are governed by subsequent Condition entry and application of associated Required Actions.

A.1 and A.2

Condition A applies if it is discovered that one or more SG tubes examined in an inservice inspection satisfy the tube plugging criteria but were not plugged in accordance with the Steam Generator Program as required by SR 3.4.17.2. An evaluation of SG tube integrity of the affected tube(s) must be made. Steam generator tube integrity is based on meeting the SG performance criteria described in the Steam Generator Program. The SG plugging criteria define limits on SG tube degradation that allow for flaw growth between inspections while still providing assurance that the SG performance criteria will continue to be met. In order to determine if a SG tube that should have been plugged has tube integrity, an evaluation must be completed that demonstrates that the SG performance criteria will continue to be met until the next refueling outage or SG tube inspection. The tube integrity determination is based on the estimated condition of the tube at the time the situation is discovered and the estimated growth of the degradation prior to the next

## BASES

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### ACTIONS (continued)

SG tube inspection. If it is determined that tube integrity is not being maintained, Condition B applies.

A Completion Time of 7 days is sufficient to complete the evaluation while minimizing the risk of plant operation with a SG tube that may not have tube integrity.

If the evaluation determines that the affected tube(s) have tube integrity, Required Action A.2 allows plant operation to continue until the next refueling outage or SG inspection provided the inspection interval continues to be supported by an operational assessment that reflects the affected tubes. However, the affected tube(s) must be plugged prior to entering MODE 4 following the next refueling outage or SG inspection. This Completion Time is acceptable since operation until the next inspection is supported by the operational assessment.

#### B.1 and B.2

If the Required Actions and associated Completion Times of Condition A are not met or if SG tube integrity is not being maintained, the reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours.

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

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### SURVEILLANCE REQUIREMENTS

#### SR 3.4.17.1

During shutdown periods the SGs are inspected as required by this SR and the Steam Generator Program. NEI 97-06, Steam Generator Program Guidelines (Ref. 1), and its referenced EPRI Guidelines, establish the content of the Steam Generator Program. Use of the Steam Generator Program ensures that the inspection is appropriate and consistent with accepted industry practices.

During SG inspections a condition monitoring assessment of the SG tubes is performed. The condition monitoring assessment determines the "as found" condition of the SG tubes. The purpose of the condition monitoring assessment is to ensure that the SG performance criteria have been met for the previous operating period.

## BASES

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### SURVEILLANCE REQUIREMENTS (continued)

The Steam Generator Program determines the scope of the inspection and the methods used to determine whether the tubes contain flaws satisfying the tube plugging criteria. Inspection scope (i.e., which tubes or areas of tubing within the SG are to be inspected) is a function of existing and potential degradation locations. The Steam Generator Program also specifies the inspection methods to be used to find potential degradation. Inspection methods are a function of degradation morphology, non-destructive examination (NDE) technique capabilities, and inspection locations.

The Steam Generator Program defines the Frequency of SR 3.4.17.1. The Frequency is determined by the operational assessment and other limits in the SG examination guidelines (Ref. 6). The Steam Generator Program uses information on existing degradations and growth rates to determine an inspection Frequency that provides reasonable assurance that the tubing will meet the SG performance criteria at the next scheduled inspection. In addition, Specification 5.5.7 contains prescriptive requirements concerning inspection intervals to provide added assurance that the SG performance criteria will be met between scheduled inspections. If crack indications are found in any SG tube, the maximum inspection interval for all affected and potentially affected SGs is restricted by Specification 5.5.7 until subsequent inspections support extending the inspection interval.

#### SR 3.4.17.2

During an SG inspection, any inspected tube that satisfies the Steam Generator Program plugging criteria is removed from service by plugging. The tube plugging criteria delineated in Specification 5.5.7 are intended to ensure that tubes accepted for continued service satisfy the SG performance criteria with allowance for error in the flaw size measurement and for future flaw growth. In addition, the tube plugging criteria, in conjunction with other elements of the Steam Generator Program, ensure that the SG performance criteria will continue to be met until the next inspection of the subject tube(s). Reference 1 provides guidance for performing operational assessments to verify that the tubes remaining in service will continue to meet the SG performance criteria.

The Frequency of prior to entering MODE 4 following a SG inspection ensures that the Surveillance has been completed and all tubes meeting the plugging criteria are plugged prior to subjecting the SG tubes to significant primary to secondary pressure differential.

BASES

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- REFERENCES
1. NEI 97-06, "Steam Generator Program Guidelines."
  2. 10 CFR 50 Appendix A, GDC 19.
  3. 10 CFR 100.
  4. ASME Boiler and Pressure Vessel Code, Section III, Subsection NB.
  5. Draft Regulatory Guide 1.121, "Basis for Plugging Degraded Steam Generator Tubes," August 1976.
  6. EPRI, "Pressurized Water Reactor Steam Generator Examination Guidelines."
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## B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

### B 3.5.1 Accumulators

#### BASES

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

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

In the refill phase of a large break LOCA, which immediately follows the blowdown phase, 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 accumulator inventory is then available to help fill voids in the lower plenum and reactor vessel downcomer so as to establish a recovery level at the bottom of the core and ongoing reflood of the core with the addition of safety injection (SI) water.

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

Each accumulator is piped into an RCS cold leg via an accumulator line and is isolated from the RCS by a motor operated isolation valve and two check valves in series.

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

## BASES

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### APPLICABLE SAFETY ANALYSES

The accumulators are assumed OPERABLE in both the large and small break LOCA analyses at full power (Ref. 1). These are the Design Basis Accidents (DBAs) that establish the acceptance limits for the accumulators. Reference to the analyses for these DBAs is used to assess changes in the accumulators as they relate to the acceptance limits.

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

The limiting large break LOCA is a double ended guillotine break. Based on deterministic studies, the worst break location is in the cold leg piping between the reactor coolant pump and the reactor vessel for the RCS loop containing the pressurizer. During this event, the accumulators discharge to the RCS as soon as RCS pressure decreases to below accumulator pressure.

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

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

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

## BASES

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### APPLICABLE SAFETY ANALYSES (continued)

small break LOCA and there is a high probability that the criteria are met following a large break LOCA:

- a. Maximum fuel element cladding temperature is  $\leq 2200^{\circ}\text{F}$ ,
- b. Maximum cladding oxidation is  $\leq 0.17$  times the total cladding thickness before oxidation,
- c. Maximum hydrogen generation from a zirconium water reaction is  $\leq 0.01$  times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react, and
- d. Core is maintained in a coolable geometry.

Since the accumulators discharge during the blowdown phase and the first few seconds of the refill phase of a large break LOCA, they do not contribute to the long term cooling requirements of 10 CFR 50.46.

For both the large and small break LOCA analyses, a nominal contained accumulator water volume is used. The contained water volume is the same as the deliverable volume for the accumulators, since the accumulators are emptied, once discharged. The large and small break LOCA safety analyses are performed with accumulator volumes that are consistent with the LOCA evaluation models. The realistic large break LOCA safety analysis takes values between 7515 gallons and 8194 gallons. To allow for instrument inaccuracy, values of 7615 gallons and 7960 gallons are specified. The small break LOCA safety analysis assumes a value from within the range of values used for the large break safety analysis.

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

BASES

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APPLICABLE SAFETY ANALYSES (continued)

The large and small break LOCA analyses are performed with accumulator pressures that are consistent with the LOCA evaluation models. The realistic large break LOCA safety analysis takes values between 600 psig and 683 psig. To allow for instrument inaccuracy, values of 624 psig and 668 psig are specified. The small break LOCA safety analysis assumes a value from the low end of the range of values taken for the large break safety analysis. The maximum nitrogen cover pressure limit prevents accumulator relief valve actuation, and ultimately preserves accumulator integrity.

The effects on containment mass and energy releases from the accumulators are accounted for in the appropriate analyses (Refs. 1 and 3).

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

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LCO

The LCO establishes the minimum conditions required to ensure that the accumulators are available to accomplish their core cooling safety function following a LOCA. Four accumulators are required to ensure that 100% of the contents of three of the accumulators will reach the core during a LOCA. This is consistent with the assumption that the contents of one accumulator spill through the break. If less than three accumulators are injected during the blowdown phase of a LOCA, the ECCS acceptance criteria of 10 CFR 50.46 (Ref. 2) could be violated.

For an accumulator to be considered OPERABLE, the isolation valve must be fully open, power removed above 2000 psig, and the limits established in the SRs for contained volume, boron concentration, and nitrogen cover pressure must be met.

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APPLICABILITY

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

This LCO is only applicable at pressures > 1000 psig. At pressures ≤ 1000 psig, the rate of RCS blowdown is such that the ECCS pumps can provide adequate injection to ensure that peak clad temperature remains below the 10 CFR 50.46 (Ref. 2) limit of 2200°F.

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

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BASES

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## ACTIONS

A.1

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

B.1

If one accumulator is inoperable for a reason other than boron concentration, the accumulator must be returned to OPERABLE status within 24 hours. In this Condition, the required contents of three accumulators cannot be assumed to reach the core during a LOCA. Due to the severity of the consequences should a LOCA occur in these conditions, the 24 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 potential for exposure of the plant to a LOCA under these conditions. The 24 hours allowed to restore an inoperable accumulator to OPERABLE status is justified in WCAP-15049-A, Rev. 1 (Ref. 4).

C.1 and C.2

If the accumulator cannot be returned to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and RCS pressure reduced to  $\leq 1000$  psig within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

BASES

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ACTIONS (continued)

D.1

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

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SURVEILLANCE  
REQUIREMENTS

SR 3.5.1.1

Each accumulator isolation valve should be verified to be fully open. This verification ensures that the accumulators are available for injection and ensures timely discovery if a valve should be less than fully open. If an isolation valve is not fully open, the rate of injection to the RCS would be reduced. Although a motor operated valve position should not change with power removed, a closed valve could result in not meeting accident analyses assumptions.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.5.1.2 and SR 3.5.1.3

Borated water volume and nitrogen cover pressure are verified for each accumulator.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.5.1.4

The boron concentration should be verified to be within required limits for each accumulator since the static design of the accumulators limits the ways in which the concentration can be changed.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

Sampling the affected accumulator within 6 hours after a 1% volume increase will identify whether inleakage has caused a reduction in boron concentration to below the required limit. It is not necessary to verify boron concentration if the added water inventory is from the refueling water storage tank (RWST), because the water contained in the RWST is within the accumulator boron concentration requirements. This is consistent with the recommendation of NUREG-1366 (Ref. 5).

BASES

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.1.5

Verification that power is removed from each accumulator isolation valve operator when the RCS pressure is  $\geq 2000$  psig ensures that an active failure could not result in the undetected closure of an accumulator motor operated isolation valve. If this were to occur, only two accumulators would be available for injection given a single failure coincident with a LOCA.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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

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REFERENCES

1. UFSAR, Chapter 6.
  2. 10 CFR 50.46.
  3. UFSAR, Chapter 15.
  4. WCAP-15049-A, Rev. 1, April 1999.
  5. NUREG-1366, February 1990.
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## B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

### B 3.5.2 ECCS - Operating

#### BASES

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##### BACKGROUND

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

- a. Loss of coolant accident (LOCA), coolant leakage greater than the capability of the normal charging system,
- b. Rod ejection accident,
- c. Loss of secondary coolant accident, including uncontrolled steam release or loss of feedwater, and
- d. Steam generator tube rupture (SGTR).

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

There are two modes of ECCS operation, injection and recirculation. The injection mode consists of the injection phase and the recirculation mode consists of the cold leg recirculation phase and hot leg recirculation phase. In the injection phase, water is taken from the refueling water storage tank (RWST) and injected into the Reactor Coolant System (RCS) through the cold legs. When sufficient water is removed from the RWST to ensure that enough boron has been added to maintain the reactor subcritical and the containment sumps have enough water to supply the required net positive suction head to the ECCS pumps, suction is switched to the containment sump for cold leg recirculation. After approximately 5.5 hours, the ECCS flow is shifted to the hot leg recirculation phase to provide a backflush, which would reduce the boiling in the top of the core and any resulting boron precipitation.

The ECCS consists of three separate subsystems: centrifugal charging (high head), safety injection (SI) (intermediate head), and residual heat removal (RHR) (low head). Each subsystem consists of two redundant, 100% capacity trains. The ECCS accumulators and the RWST are also part of the ECCS, but are not considered part of an ECCS flow path as described by this LCO.

BASES

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## BACKGROUND (continued)

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

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

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

During the recirculation phase of LOCA recovery, RHR pump suction is transferred to the containment sump. The RHR pumps then supply the other ECCS pumps. Initially, recirculation is through the same paths as the injection phase. Subsequently, recirculation alternates injection between the hot and cold legs.

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

## BASES

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### BACKGROUND (continued)

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

The ECCS subsystems are actuated upon receipt of an SI signal. If offsite power is available, the safeguard loads start immediately in the programmed sequence. If offsite power is not available, the Engineered Safety Feature (ESF) buses shed normal operating loads and are connected to the emergency diesel generators (EDGs). Safeguard loads are then actuated in the programmed time sequence. The time delay associated with diesel starting, sequenced loading, and pump starting determines the time required before pumped flow is available to the core following a LOCA.

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

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### APPLICABLE SAFETY ANALYSES

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

- a. Maximum fuel element cladding temperature is  $\leq 2200^{\circ}\text{F}$ ,
- b. Maximum cladding oxidation is  $\leq 0.17$  times the total cladding thickness before oxidation,
- c. Maximum hydrogen generation from a zirconium water reaction is  $\leq 0.01$  times the hypothetical amount generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react,
- d. Core is maintained in a coolable geometry, and
- e. Adequate long term core cooling capability is maintained.

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

## BASES

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### APPLICABLE SAFETY ANALYSES (continued)

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

- a. A large break LOCA event, with loss of offsite power and a single failure disabling one ECCS train (both containment spray trains are assumed to operate conservatively reducing containment pressure and increasing break flow) and
- b. A small break LOCA event, with a loss of offsite power and a single failure disabling one ECCS train.

During the blowdown stage of a LOCA, the RCS depressurizes as primary coolant is ejected through the break into the containment. The nuclear reaction is terminated either by moderator voiding during large breaks or control rod insertion for small breaks. Following depressurization, emergency cooling water is injected into the cold legs, flows into the downcomer, fills the lower plenum, and refloods the core.

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

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

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### LCO

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

## BASES

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### LCO (continued)

In MODES 1, 2, and 3, an ECCS train consists of a centrifugal charging subsystem, an SI subsystem, and an RHR subsystem. Each ECCS train includes the piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the RWST upon an SI signal and automatically transferring RHR suction to the containment sump.

During an event requiring ECCS actuation, a flow path is required to provide an abundant supply of water from the RWST to the RCS via the ECCS pumps and their respective supply headers to each of the four cold leg injection nozzles. In the long term, this flow path may be switched to take its supply from the containment sump and to supply its flow to the RCS hot and cold legs.

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

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

As indicated in Note 2, operation in MODE 3 with ECCS trains made incapable of injecting in order to facilitate entry into or exit from the Applicability of LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," is necessary for plants with an LTOP arming temperature at or near the MODE 3 boundary temperature of 350°F. LCO 3.4.12 requires that certain pumps be rendered incapable of injecting at and below the LTOP arming temperature. When this temperature is at or near the MODE 3 boundary temperature, time is needed to make pumps incapable of injecting prior to entering the LTOP Applicability, and provide time to restore the inoperable pumps to OPERABLE status on exiting the LTOP Applicability.

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### APPLICABILITY

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

## BASES

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### APPLICABILITY (continued)

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

In MODES 5 and 6, plant conditions are such that the probability of an event requiring ECCS injection is extremely low.

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### ACTIONS

#### A.1

With one or more ECCS trains inoperable and at least 100% of the ECCS flow equivalent to a single OPERABLE ECCS train available, the inoperable components must be returned to OPERABLE status within 72 hours. The 72 hour Completion Time is based on an NRC reliability evaluation (Ref. 5) and is a reasonable time for repair of many ECCS components.

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

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

An event accompanied by a loss of offsite power and the failure of an EDG can disable one ECCS train until power is restored. A reliability analysis (Ref. 5) has shown that the impact of having one full ECCS train inoperable is sufficiently small to justify continued operation for 72 hours.

Reference 6 describes situations in which one component, such as an RHR crossover valve, can disable both ECCS trains. With one or more component(s) inoperable such that 100% of the flow equivalent to a single OPERABLE ECCS train is not available, the facility is in a condition outside the accident analysis. Therefore, LCO 3.0.3 must be immediately entered.

## BASES

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### ACTIONS (continued)

#### B.1 and B.2

If the inoperable trains cannot be returned to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

#### C.1

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

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### SURVEILLANCE REQUIREMENTS

#### SR 3.5.2.1

Verification of proper valve position ensures that the flow path from the ECCS pumps to the RCS is maintained. Misalignment of these valves could render both ECCS trains inoperable. Securing these valves in position by removal of power ensures that they cannot change position as a result of an active failure or be inadvertently misaligned. These valves are of the type, described in Reference 6, that can disable the function of both ECCS trains and invalidate the accident analyses.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.2.2

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

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.5.2.3

With the exception of the operating centrifugal charging pump, 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. ECCS piping is verified full of water by venting and/or ultrasonic testing (UT) pump casings and accessible high point vents. This will also prevent water hammer, pump cavitation, and pumping of noncondensable gas (e.g., air, nitrogen, or hydrogen) into the reactor vessel following an SI signal or during shutdown cooling.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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 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 plant safety analysis. SRs are specified in the Inservice Testing Program of the ASME Code.

## BASES

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### SURVEILLANCE REQUIREMENTS (continued)

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

#### SR 3.5.2.5 and SR 3.5.2.6

These Surveillances demonstrate that each automatic ECCS valve actuates to the required position on an actual or simulated SI signal and that each ECCS pump starts on receipt of an actual or simulated SI signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

#### SR 3.5.2.7

Realignment of valves in the flow path on an SI signal is necessary for proper ECCS performance. These valves have stops to allow proper positioning for restricted flow to a ruptured cold leg, ensuring that the other cold legs receive at least the required minimum flow.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

#### SR 3.5.2.8

Periodic inspections of the containment sump suction inlet ensure that it is unrestricted and stays in proper operating condition.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

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- REFERENCES
1. 10 CFR 50, Appendix A, GDC 35.
  2. 10 CFR 50.46.
  3. UFSAR, Section 6.3.
  4. UFSAR, Chapter 15, "Accident Analysis."
  5. NRC Memorandum to V. Stello, Jr., from R.L. Baer, "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975.
  6. IE Information Notice No. 87-01.
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## B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

### B 3.5.3 ECCS - Shutdown

#### BASES

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BACKGROUND	<p>The Background section for Bases 3.5.2, "ECCS - Operating," is applicable to these Bases, with the following modifications.</p> <p>In MODE 4, the required ECCS train consists of two separate subsystems: centrifugal charging (high head) and residual heat removal (RHR) (low head).</p> <p>The ECCS flow paths consist of piping, valves, RHR heat exchangers, and pumps such that water from the refueling water storage tank (RWST) can be injected into the Reactor Coolant System (RCS) following the accidents described in Bases 3.5.2.</p>
APPLICABLE SAFETY ANALYSES	<p>The Applicable Safety Analyses section of Bases 3.5.2 also applies to this Bases section.</p> <p>Due to the stable conditions associated with operation in MODE 4 and the reduced probability of occurrence of a Design Basis Accident (DBA), the ECCS operational requirements are reduced. It is understood in these reductions that automatic safety injection (SI) actuation is not available. In this MODE, sufficient time exists for manual actuation of the required ECCS to mitigate the consequences of a DBA.</p> <p>Only one train of ECCS is required for MODE 4. This requirement dictates that single failures are not considered during this MODE of operation. The ECCS trains satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).</p>
LCO	<p>In MODE 4, one of the two independent (and redundant) ECCS trains is required to be OPERABLE to ensure that sufficient ECCS flow is available to the core and adequate core cooling is maintained following a DBA.</p> <p>In MODE 4, an ECCS train consists of a centrifugal charging subsystem and an RHR subsystem. Each train includes the piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the RWST and transferring suction to the containment sump.</p> <p>During an event requiring ECCS actuation, a flow path is required to provide an abundant supply of water from the RWST to the RCS via the ECCS pumps and their respective supply headers to each of the four cold leg injection nozzles. In the long term, this flow path may be switched to take its supply from the containment sump and to deliver its flow to the RCS hot and cold legs.</p>

BASES

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LCO (continued)

This LCO is modified by two Notes. Note 1 allows an RHR train to be considered OPERABLE during alignment and operation for decay heat removal, if capable of being manually realigned (remote or local) to the ECCS mode of operation and not otherwise inoperable. A second Note allows the required ECCS RHR subsystem to be inoperable because of surveillance testing of RCS pressure isolation valve leakage (FCV-74-1 and FCV-74-2). This allows testing while RCS pressure is high enough to obtain valid leakage data and following valve closure for RHR decay heat removal path. The condition requiring manual realignment capability (FCV-74-1 and FCV-74-2 can be opened from the main control room) ensures that in the unlikely event of a DBA during the one hour of surveillance testing, the RHR subsystem can be placed in ECCS recirculation mode when required to mitigate the event. This allows operation in the RHR mode during MODE 4.

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APPLICABILITY

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

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

In MODES 5 and 6, plant conditions are such that the probability of an event requiring ECCS injection is extremely low.

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ACTIONS

A Note prohibits the application of LCO 3.0.4.b to an inoperable ECCS centrifugal charging subsystem when entering MODE 4. There is an increased risk associated with entering MODE 4 from MODE 5 with an inoperable ECCS centrifugal charging subsystem and the provisions of LCO 3.0.4.b, which allow entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, should not be applied in this circumstance.

A.1

With no ECCS RHR subsystem OPERABLE, the plant is not prepared to respond to a loss of coolant accident or to continue a cooldown using the RHR pumps and heat exchangers. The Completion Time of immediately to initiate actions that would restore at least one ECCS RHR subsystem to OPERABLE status ensures that prompt action is taken to restore the required cooling capacity.

BASES

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ACTIONS (continued)

With both RHR subsystems inoperable, it would be unwise to require the plant to go to MODE 5, where the only available heat removal system is the RHR. Therefore, the appropriate action is to initiate measures to restore one ECCS RHR subsystem and to continue the actions until the subsystem is restored to OPERABLE status.

B.1

With no ECCS centrifugal charging subsystem OPERABLE, the plant is not prepared to provide high pressure response to Design Basis Events requiring SI. The 1 hour Completion Time to restore at least one ECCS centrifugal charging subsystem to OPERABLE status ensures that prompt action is taken to provide the required cooling capacity.

C.1

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

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SURVEILLANCE  
REQUIREMENTS

SR 3.5.3.1

The applicable Surveillance descriptions from Bases 3.5.2 apply.

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REFERENCES

The applicable references from Bases 3.5.2 apply.

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## B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

### B 3.5.4 Refueling Water Storage Tank (RWST)

#### BASES

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##### BACKGROUND

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

The RWST supplies both trains of the ECCS and the Containment Spray System through a common supply header during the injection phase of a loss of coolant accident (LOCA) recovery. Motor operated isolation valves are provided in the header to isolate the RWST from the ECCS once the system has been transferred to the recirculation mode. The recirculation mode is entered when ECCS pump suction is transferred to the containment sump following receipt of the RWST – Low Level signal coincident with Containment Sump Level – High signal. The transfer of the containment spray pump suction to the containment sump is manually initiated upon receipt of a high level in the containment sump or the RWST Low-Low (Level) alarm. Use of a single RWST to supply both trains of the ECCS and Containment Spray System is acceptable since the RWST is a passive component, and passive failures are not required to be assumed to occur coincidentally with Design Basis Events.

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

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

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

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

BASES

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## BACKGROUND (continued)

This LCO ensures that:

- a. The RWST contains sufficient borated water to support the ECCS during the injection phase;
- b. Sufficient water volume exists in the containment sump to support continued operation of the ECCS and Containment Spray System pumps at the time of transfer to the recirculation mode of cooling; and
- c. The reactor remains subcritical following a LOCA.

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

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APPLICABLE  
SAFETY  
ANALYSES

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

The RWST must also meet volume, boron concentration, and temperature requirements for non-LOCA events. The volume is not an explicit assumption in non-LOCA events since the required volume is a small fraction of the available volume. The deliverable volume limit is set by the LOCA and containment analyses. For the RWST, the deliverable volume is different from the total volume contained since, due to the design of the tank, more water can be contained than can be delivered. The minimum boron concentration is an explicit assumption in the main steam line break (MSLB) analysis to ensure the required shutdown capability.

## BASES

## APPLICABLE SAFETY ANALYSES (continued)

The maximum RWST temperature ensures that the amount of cooling provided from the RWST during the heatup phase of a feedline break is consistent with safety analysis assumptions; the minimum RWST temperature is an assumption in the MSLB analysis.

The MSLB analysis has considered a delay associated with the interlock between the VCT and RWST isolation valves, and the results show that the departure from nucleate boiling design basis is met. The delay has been established as 28 seconds, with offsite power available, or 58 seconds without offsite power.

For a large break LOCA analysis, the minimum water volume limit of 370,000 gallons and the lower boron concentration limit of 2500 ppm are used to compute the post LOCA sump boron concentration necessary to assure subcriticality. The large break LOCA is the limiting case since the safety analysis assumes that all control rods are out of the core.

The upper limit on boron concentration of 2700 ppm is used to determine the maximum allowable time to switch to hot leg recirculation following a LOCA. The purpose of switching from cold leg to hot leg recirculation is to minimize the potential for boron precipitation in the core following the accident.

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

## LCO

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

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

## APPLICABILITY

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

BASES

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## ACTIONS

A.1

With RWST boron concentration or borated water temperature not within limits, they must be returned to within limits within 8 hours. Under these conditions neither the ECCS nor the Containment Spray System can perform its design function. Therefore, prompt action must be taken to restore the RWST to OPERABLE condition. The 8 hour limit to restore the RWST temperature or boron concentration to within limits was developed considering the time required to change either the boron concentration or temperature and the fact that the contents of the tank are still available for injection.

B.1

With the RWST inoperable for reasons other than Condition A (e.g., water volume), it must be restored to OPERABLE status within 1 hour. In this condition, neither the ECCS nor the Containment Spray System can perform its design function. Therefore, prompt action must be taken to restore the RWST to OPERABLE status. The short time limit of 1 hour to restore the RWST to OPERABLE status is based on this condition simultaneously affecting redundant trains.

C.1 and C.2

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

BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.5.4.1

The RWST borated water temperature should be verified to be within the limits assumed in the accident analyses band.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.5.4.2

The RWST water volume should be verified to be above the required minimum level in order to ensure that a sufficient initial supply is available for injection and to support continued ECCS and Containment Spray System pump operation on recirculation.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.5.4.3

The boron concentration of the RWST should be verified to be within the required limits. This SR ensures that the reactor will remain subcritical following a LOCA. Further, it assures that the resulting sump pH will be maintained in an acceptable range so that boron precipitation in the core will not occur and the effect of chloride and caustic stress corrosion on mechanical systems and components will be minimized.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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REFERENCES

1. UFSAR, Chapter 6 and Chapter 15.
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## B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

### B 3.5.5 Seal Injection Flow

#### BASES

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BACKGROUND	<p>The function of the seal injection throttle valves during an accident is similar to the function of the ECCS throttle valves in that each restricts flow from the centrifugal charging pump header to the Reactor Coolant System (RCS).</p> <p>The restriction on reactor coolant pump (RCP) seal injection flow limits the amount of ECCS flow that would be diverted from the injection path following an accident. This limit is based on safety analysis assumptions that are required because RCP seal injection flow is not isolated during safety injection (SI).</p> <p>The RCP seal injection flow is restricted by the seal injection line flow which is adjusted through positioning of the manual RCP seal injection throttle valves. The RCP seal injection flow is determined by measuring the pressurizer pressure, the centrifugal charging pump discharge header pressure, and the RCP seal injection flow rate.</p> <p>The charging flow control valve throttles the centrifugal charging pump discharge header flow as necessary to maintain the programmed level in the pressurizer. The charging flow control valve fails open to ensure that, in the event of either loss of air or loss of control signal to the valve, when the centrifugal charging pumps are supplying charging flow, seal injection flow to the RCP seals is maintained. Positioning of the charging flow control valve may vary during normal plant operating conditions, resulting in a proportional change to RCP seal injection flow. The flow provided by RCP seal injection throttle valves will remain fixed when the charging flow control valve is repositioned provided the throttle valve(s) position are not adjusted.</p>
APPLICABLE SAFETY ANALYSES	<p>All ECCS subsystems are taken credit for in the large break loss of coolant accident (LOCA) at full power (Ref. 1). The LOCA analysis establishes the minimum flow for the ECCS pumps. The centrifugal charging pumps are also credited in the small break LOCA analysis. This analysis establishes the flow and discharge head at the design point for the centrifugal charging pumps. The steam generator tube rupture and main steam line break event analyses also credit the centrifugal charging pumps, but are not limiting in their design. Reference to these analyses is made in assessing changes to the Seal Injection System for evaluation of their effects in relation to the acceptance limits in these analyses.</p>

BASES

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APPLICABLE SAFETY ANALYSES (continued)

This LCO ensures that seal injection flow will be sufficient for RCP seal integrity but limited so that the ECCS trains will be capable of delivering sufficient water to match boiloff rates soon enough to minimize uncovering of the core following a large LOCA. It also ensures that the centrifugal charging pumps will deliver sufficient water for a small LOCA and sufficient boron to maintain the core subcritical. For smaller LOCAs, the charging pumps alone deliver sufficient fluid to overcome the loss and maintain RCS inventory. Seal injection flow satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

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LCO

The intent of the LCO limit on seal injection flow is to make sure that flow through the RCP seal water injection line is low enough to ensure that sufficient centrifugal charging pump injection flow is directed to the RCS via the injection points (Ref. 2).

The LCO is not strictly a flow limit, but rather a flow limit based on a flow line resistance. In order to establish the proper flow line resistance, a pressure and flow must be known. The flow line resistance is established by adjusting the RCP seal injection throttle valves (needle valves) to provide a total seal injection flow in the acceptable region of Figure 3.5.5-1 at a given pressure differential between the charging header and the RCS. The centrifugal charging pump discharge header pressure remains essentially constant through all the applicable MODES of this LCO. A reduction in RCS pressure would result in more flow being diverted to the RCP seal injection line than at normal operating pressure. The valve settings established at the prescribed centrifugal charging pump discharge header pressure result in a conservative valve position should RCS pressure decrease. The flow limits established by Figure 3.5.5-1 ensures that the minimum ECCS flow assumed in the safety analyses is maintained.

The limit on seal injection flow must be met to render the ECCS OPERABLE. If these conditions are not met, the ECCS flow will not be as assumed in the accident analyses.

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APPLICABILITY

In MODES 1, 2, and 3, the seal injection flow limit is dictated by ECCS flow requirements, which are specified for MODES 1, 2, 3, and 4. The seal injection flow limit is not applicable for MODE 4 and lower, however, because high seal injection flow is less critical as a result of the lower initial RCS pressure and decay heat removal requirements in these MODES. Therefore, RCP seal injection flow must be limited in MODES 1, 2, and 3 to ensure adequate ECCS performance.

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BASES

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ACTIONS

A.1

With the seal injection flow not within its limit, the amount of charging flow available to the RCS may be reduced. Under this Condition, action must be taken to restore the flow to within its limit. The operator has 4 hours from the time the flow is known to not be within the limit to correctly position the manual valves and thus be in compliance with the accident analysis. The Completion Time minimizes the potential exposure of the plant to a LOCA with insufficient injection flow and provides a reasonable time to restore seal injection flow within limits. This time is conservative with respect to the Completion Times of other ECCS LCOs; it is based on operating experience and is sufficient for taking corrective actions by operations personnel.

B.1 and B.2

When the Required Actions cannot be completed within the required Completion Time, a controlled shutdown must be initiated. The Completion Time of 6 hours for reaching MODE 3 from MODE 1 is a reasonable time for a controlled shutdown, based on operating experience and normal cooldown rates, and does not challenge plant safety systems or operators. Continuing the plant shutdown begun in Required Action B.1, an additional 6 hours is a reasonable time, based on operating experience and normal cooldown rates, to reach MODE 4, where this LCO is no longer applicable.

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SURVEILLANCE  
REQUIREMENTS

SR 3.5.5.1

Verification that the manual seal injection throttle valves are adjusted to give a flow within the limit ensures that the ECCS injection flows stay within the safety analysis. A differential pressure is established between the charging header and the RCS, and the total seal injection flow is verified to within the limit determined in accordance with the ECCS safety analysis.

The flow shall be verified by confirming seal injection flow and differential pressure within the acceptable region of Figure 3.5.5-1.

Control valves in the flow path between the charging header and the RCS pressure sensing points must be in their post accident position (e.g., charging flow control valve open) during this Surveillance to correlate with the acceptance criteria.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

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SURVEILLANCE REQUIREMENTS (continued)

As noted, the Surveillance is not required to be performed until 4 hours after the RCS pressure has stabilized at  $\geq 2215$  psig and  $\leq 2255$  psig. The RCS pressure requirement is specified since this configuration will produce the required pressure conditions necessary to assure that the manual valves are set correctly. The exception is limited to 4 hours to ensure that the Surveillance is timely.

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- REFERENCES
1. UFSAR, Chapter 6 and Chapter 15.
  2. 10 CFR 50.46.
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## B 3.6 CONTAINMENT SYSTEMS

### B 3.6.1 Containment

#### BASES

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**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 the radioactive material that may be released from the reactor core following a design basis loss of coolant accident (LOCA). Additionally, the containment and shield building provide shielding from the fission products that may be present in the containment atmosphere following accident conditions.

The containment vessel is a vertical cylindrical steel pressure vessel with hemispherical dome and a concrete base mat with steel membrane. It is completely enclosed by a reinforced concrete shield building. An annular space exists between the walls and domes of the steel containment vessel and the concrete shield building to provide for the collection, mixing, holdup, and controlled release of containment out leakage. Ice condenser containments utilize an outer concrete building for shielding and an inner steel containment for leak tightness.

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

The inner steel containment and its penetrations establish the leakage limiting boundary of the containment. Maintaining the containment OPERABLE limits the leakage of fission product radioactivity from the containment to the environment. SR 3.6.1.1 leakage rate requirements comply with 10 CFR 50, Appendix J, Option B (Ref. 1), as modified by approved exemptions.

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

BASES

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BACKGROUND (continued)

- 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 sealing mechanism associated with each containment penetration (e.g., welds, bellows, or O-rings) is OPERABLE (i.e., OPERABLE such that the containment leakage limits are met).
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APPLICABLE  
SAFETY  
ANALYSES

The safety design basis for the containment is that the containment must withstand the pressures and temperatures of the limiting Design Basis Accident (DBA) without exceeding the design leakage rates.

The DBAs that result in a challenge to containment OPERABILITY from high pressures and temperatures are a LOCA, a steam line break, and a rod ejection accident (REA) (Ref. 2). In addition, release of significant fission product radioactivity within containment can occur from a LOCA or REA. In the DBA analyses, it is assumed that the containment is OPERABLE such that, for the DBAs involving release of fission product radioactivity, release to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of 0.25% of containment air weight per day (Ref. 3). This leakage rate, used in the evaluation of offsite doses resulting from accidents, is defined in 10 CFR 50, Appendix J, Option B (Ref. 1), as  $L_a$ : the maximum allowable containment leakage rate at the calculated peak containment internal pressure ( $P_a$ ) resulting from the limiting design basis LOCA. The allowable leakage rate represented by  $L_a$  forms the basis for the acceptance criteria imposed on all containment leakage rate testing.  $L_a$  is assumed to be 0.25% per day in the safety analysis at  $P_a$  (Ref. 3).

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

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

## BASES

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LCO Containment OPERABILITY is maintained by limiting leakage to  $\leq 1.0 L_a$ , except prior to the first startup after performing a required Containment Leakage Rate Testing Program leakage test. At this time the applicable leakage limits must be met.

Compliance with this LCO will ensure a containment configuration, including equipment hatches, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analysis.

Individual leakage rates specified for the containment air lock (LCO 3.6.2), 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 overall acceptance criteria of the Containment Leakage Rate Testing Program.

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APPLICABILITY In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material into containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, containment is not required to be OPERABLE in MODE 5 to prevent leakage of radioactive material from containment. The requirements for containment during MODE 6 are addressed in LCO 3.9.4, "Containment Penetrations."

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## ACTIONS

### A.1

In the event containment is inoperable, containment must be restored to OPERABLE status within 1 hour. The 1 hour Completion Time provides a period of time to correct the problem commensurate with the importance of maintaining containment 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.

BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.6.1.1

Maintaining the containment OPERABLE requires compliance with the visual examinations and leakage rate test requirements of the Containment Leakage Rate Testing Program. The containment concrete visual examinations may be performed during either power operation or during a maintenance or refueling outage. The visual examinations of the steel liner plate inside containment are performed during maintenance or refueling outages since this is the only time the liner plate is fully accessible.

Failure to meet air lock, shield building bypass leakage path, 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 Containment Leakage Rate Testing Program leakage test is required to be  $\leq 0.6 L_a$  for combined Type B and C leakage, and  $\leq 0.75 L_a$  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 the Containment Leakage Rate Testing Program. These periodic testing requirements verify that the containment leakage rate does not exceed the leakage rate assumed in the safety analysis.

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REFERENCES

1. 10 CFR 50, Appendix J, Option B.
  2. UFSAR, Chapter 15.
  3. UFSAR, Section 6.2.
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## B 3.6 CONTAINMENT SYSTEMS

### B 3.6.2 Containment Air Locks

#### BASES

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##### BACKGROUND

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

Each air lock is nominally a right circular cylinder, approximately 8 feet 7 inches 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.

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 analyses.

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##### APPLICABLE SAFETY ANALYSES

The DBAs that result in a release of radioactive material within containment are a loss of coolant accident (LOCA) and a rod ejection accident (Ref. 2). In the analysis of each of these accidents, it is assumed that containment is OPERABLE such that release of fission products to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of 0.25% of containment air weight per day (Ref. 3). This leakage rate is defined in 10 CFR 50, Appendix J, Option B (Ref. 1), as  $L_a = 0.25\%$  of containment air weight per day, the maximum allowable containment leakage rate at the calculated peak containment internal pressure  $P_a$  following a design basis LOCA. This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air locks.

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

## BASES

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### LCO

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

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

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### APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the containment air locks are not required in MODE 5 to prevent leakage of radioactive material from containment. The requirements for the containment air locks during MODE 6 are addressed in LCO 3.9.4, "Containment Penetrations."

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### ACTIONS

The ACTIONS are modified by a Note that allows entry and exit to perform repairs on the affected air lock component. If the outer door is inoperable, then it may be easily accessed for most repairs. It is preferred that the air lock be accessed from inside primary containment by entering through the other OPERABLE air lock. However, if this 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 due to the low probability of an event that could pressurize the containment during the short time in which the OPERABLE door is expected to be open. After each entry and exit, the OPERABLE door must be immediately closed. If ALARA conditions permit, entry and exit should be via an OPERABLE air lock.

## BASES

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### ACTIONS (continued)

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

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

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

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

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

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

## BASES

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### ACTIONS (continued)

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 on a periodic basis 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 is 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 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.

BASES

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ACTIONS (continued)

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 within the 1 hour Completion Time. This specified time period is consistent with the ACTIONS of LCO 3.6.1, which requires that containment be restored to OPERABLE status within 1 hour.

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

D.1 and D.2

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

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SURVEILLANCE  
REQUIREMENTS

SR 3.6.2.1

Maintaining containment air locks OPERABLE requires compliance with the leakage rate test requirements of the Containment Leakage Rate Testing Program. This SR reflects the leakage rate testing requirements with regard to air lock leakage (Type B leakage tests). The acceptance criteria were established during initial air lock and containment OPERABILITY testing. The periodic testing requirements verify that the

## BASES

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### SURVEILLANCE REQUIREMENTS (continued)

air lock leakage does not exceed the allowed fraction of the overall containment leakage rate. The Frequency is required by the Containment Leakage Rate Testing Program.

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

#### SR 3.6.2.2

The air lock interlock is designed to prevent simultaneous opening of both doors in a single air lock. Since both the inner and outer doors of an air lock are designed to withstand the maximum expected post accident containment pressure, closure of either door will support containment OPERABILITY. Thus, the door interlock feature supports containment OPERABILITY while the air lock is being used for personnel transit in and out of the containment. Periodic testing of this interlock demonstrates that the interlock will function as designed and that simultaneous opening of the inner and outer doors will not inadvertently occur.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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- |            |                                     |
|------------|-------------------------------------|
| REFERENCES | 1. 10 CFR 50, Appendix J, Option B. |
|            | 2. UFSAR, Section 15.4.             |
|            | 3. UFSAR, Section 6.2.4.1.          |
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## B 3.6 CONTAINMENT SYSTEMS

### B 3.6.3 Containment Isolation Valves

#### BASES

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**BACKGROUND** The containment isolation valves form part of the containment pressure boundary and provide a means for fluid penetrations not serving accident consequence limiting systems to be provided with two isolation barriers that are closed on a containment 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, or an approved exemption is provided, so that no single credible failure or malfunction of an active component can result in a loss of isolation or leakage that exceeds limits assumed in the safety analyses. One of these barriers may be a closed system. These barriers (typically containment isolation valves) make up the Containment Isolation System.

Automatic isolation signals are produced during accident conditions. Containment Phase "A" isolation occurs upon receipt of a safety injection signal. The Phase "A" isolation signal isolates nonessential process lines in order to minimize leakage of fission product radioactivity. Containment Phase "B" isolation occurs upon receipt of a containment pressure High-High signal and isolates the remaining process lines, except systems required for accident mitigation. In addition to the isolation signals listed above, the purge and exhaust valves receive an isolation signal on a containment high radiation condition. As a result, the containment isolation valves (and blind flanges) help ensure that the containment atmosphere will be isolated from the environment in the event of a release of fission product radioactivity to the containment atmosphere as a result of a Design Basis Accident (DBA).

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

BASES

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BACKGROUND (continued)

Reactor Building Purge Ventilation (RBPV) System

The RBPV System operates to supply outside air into the containment for ventilation and cooling or heating and may also be used to reduce the concentration of noble gases within containment prior to and during personnel access. The RBPV System provides for mechanical ventilation of the primary containment, the instrument room located within the containment, and the annulus secondary containment located between primary containment and the Shield Building.

The RBPV System includes one supply duct penetration through the Shield Building wall into the annulus area. There are four purge air supply penetrations through the containment vessel, two to the upper compartment and two to the lower containment. Two normally closed 24-inch purge supply isolation valves at each penetration through the containment vessel provide containment isolation.

The RBPV System includes one exhaust duct penetration through the Shield Building wall from the annulus area. There are three purge air exhaust penetrations through the containment vessel, two from the upper compartment and one from the lower containment. There is one pressure relief penetration through the containment vessel. Two normally closed 24-inch purge exhaust isolation valves at each penetration through the containment vessel provide containment isolation. Two normally closed 8-inch pressure relief isolation valves through the containment vessel provide containment isolation.

The RBPV System includes one supply and one exhaust duct penetration through the Shield Building wall and one supply and one exhaust duct penetration through the containment vessel wall for ventilation of the instrument room inside containment. Two normally closed 12-inch purge isolation valves at each supply and exhaust penetration through the containment vessel provide containment isolation.

Since the valves used in the RBPV System are designed to meet the requirements for automatic containment isolation valves, these valves may be opened as needed in MODES 1, 2, 3, and 4.

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APPLICABLE  
SAFETY  
ANALYSES

The containment isolation valve LCO was derived from the assumptions related to minimizing the loss of reactor coolant inventory and establishing 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 analyses of any event requiring isolation of containment is applicable to this LCO.

BASES

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APPLICABLE SAFETY ANALYSES (continued)

The DBAs that result in a release of radioactive material within containment are a loss of coolant accident (LOCA) and a rod ejection accident (Ref. 1). In the analyses 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 bounding safety analyses assume that one pair of containment purge system lines are open at event initiation. The open purge system lines include one set of supply valves (i.e., inboard and outboard) and one set of exhaust valves (i.e., inboard and outboard).

The DBA analysis assumes that, within 85 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 85 seconds includes signal delay, diesel generator startup (for loss of offsite power), and containment isolation valve stroke times.

The single failure criterion required to be imposed in the conduct of plant safety analyses was considered in the original design of the containment purge valves. Two valves (i.e., one set) 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, pneumatically operated to open and spring closed. This arrangement was designed to preclude common mode failures from disabling both valves on a purge line.

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

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LCO

Containment isolation valves form a part of the containment boundary. The containment isolation valves' 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 containment isolation purge valves have blocks installed to prevent full opening. Blocked purge valves also actuate on an automatic signal. The valves covered by this LCO are listed along with their associated stroke times in the UFSAR (Ref. 2).

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

BASES

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LCO (continued)

intact. These passive isolation valves/devices are those listed in Reference 2.

Purge valves with resilient seals and shield building bypass leakage paths 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.

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APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the containment isolation valves are not required to be OPERABLE in MODE 5. The requirements for containment isolation valves during MODE 6 are addressed in LCO 3.9.4, "Containment Penetrations."

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ACTIONS

The ACTIONS are modified by a Note allowing 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.

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

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

## BASES

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### ACTIONS (continued)

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

Note 5 limits the number of open containment purge lines to no more than one set of supply valves and one set of exhaust valves.

#### A.1 and A.2

Condition A is applicable to penetration flow paths with two containment isolation valves, and penetration flow paths with only one containment isolation valve and a closed system. The closed system must meet the requirements of Reference 3.

In the event one containment isolation valve in one or more penetration flow paths is inoperable, except for inoperable containment vacuum relief isolation valve(s), or shield building bypass or containment purge isolation valve leakage not within limit, the affected penetration flow path must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and deactivated automatic containment isolation valve, a closed manual valve, a blind flange, and a check valve with flow through the valve secured. For a penetration flow path 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 Completion Time specified for each Category of containment isolation valve identified in Table B 3.6.3-1, Containment Isolation Valve Completion Times. The Completion Time is justified in Reference 4.

For affected penetration flow paths that cannot be restored to OPERABLE status within the specified 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 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

## BASES

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### ACTIONS (continued)

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.

Required Action A.2 is modified by two Notes. Note 1 applies to isolation devices located in high radiation areas and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Note 2 applies to isolation devices that are locked, sealed, or otherwise secured in position and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since the function of locking, sealing, or securing components is to ensure that these devices are not inadvertently repositioned. 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 inoperable containment vacuum relief isolation valve(s), or shield building bypass or containment purge isolation valve leakage not within limit, the affected penetration flow path must be isolated within 1 hour. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, a closed manual valve, and a blind flange. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1. In the event the affected penetration is isolated in accordance with Required Action B.1, the affected penetration must be verified to be isolated on a periodic basis per Required Action A.2, which remains in effect. This periodic verification is necessary to assure leak tightness of containment and that penetrations requiring isolation following an accident are isolated. The Completion Time of once per 31 days for verifying each affected penetration flow path is isolated is appropriate considering the fact that the valves are operated under administrative control 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.

BASES

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ACTIONS (continued)

C.1

In the event one containment isolation valve in two or more penetration flow paths is inoperable except for inoperable containment vacuum relief isolation valve(s), or shield building bypass or containment purge isolation valve leakage not within limit, all but one of the affected penetration flow path(s) must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic containment isolation valve, a closed manual valve, a blind flange, and a check valve with flow through the valve secured. For a penetration flow path isolated in accordance with Required Action C.1, the device used to isolate the penetration should be the closest available one to containment. Required Action C.1 must be completed within 4 hours. For subsequent containment isolation valve inoperabilities, the Required Action and Completion Time continue to apply to each additional containment isolation valve inoperability, with the Completion Time based on each subsequent entry into the Condition consistent with Note 2 to the ACTIONS Table (e.g., for each entry into the Condition). Each containment isolation valve(s) that is (are) declared inoperable for subsequent Condition C entries shall meet the Required Action and Completion Time. For the penetration flow paths isolated in accordance with Required Action C.1, the affected penetration(s) 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 that the penetrations requiring isolation following an accident are isolated. 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.

D.1 and D.2

In the event two or more pairs of containment purge lines are open, all but one penetration flow paths 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, and a blind flange. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1. The 1 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.

## BASES

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### ACTIONS (continued)

#### E.1

With one or more containment vacuum relief isolation valve(s) inoperable, the inoperable valve(s) must be restored to OPERABLE status or the affected penetration flow path must be isolated.

In the containment vacuum relief lines, containment vacuum relief valves 30-571, 30-572, and 30-573 are qualified to perform a containment isolation function. These valves are self-actuated, swing disk (check) valves with an elastomer seat. The valves are normally closed and are equipped with limit switches that provide fully open and fully closed indication in the main control room (MCR). Therefore, a 72 hour Completion Time is appropriate while actions are taken to return the containment vacuum relief isolation valves to service.

#### F.1

With the shield building bypass leakage rate (SR 3.6.3.8) not within limit, the assumptions of the safety analyses are not met. Therefore, the leakage must be restored to within limit. 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.

#### G.1, G.2, and G.3

In the event one or more containment purge supply or exhaust valves in one or more penetration flow paths are not within the purge supply and exhaust valve leakage limits, purge supply and exhaust valve leakage must be restored to within limits, or the affected penetration flow path must be isolated. The method of isolation must be by the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, closed manual valve, or blind flange. A purge valve with resilient seals utilized to satisfy Required Action G.1

## BASES

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### ACTIONS (continued)

must have been demonstrated to meet the leakage requirements of SR 3.6.3.5. 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 G.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 that those 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 a resilient seal that is isolated in accordance with Required Action G.1, SR 3.6.3.5 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. Since more reliance is placed on a single valve while in this Condition, a Frequency of once per 92 days was chosen and has been shown to be acceptable based on operating experience.

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

BASES

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ACTIONS (continued)

H.1 and H.2

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

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SURVEILLANCE  
REQUIREMENTS

SR 3.6.3.1

This SR ensures that the containment purge 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 containment purge valves are open for the reasons stated. The valves may be opened for pressure control, ALARA or air quality considerations for personnel entry, or for Surveillances that require the valves to be open. The number of valves open during MODES 1, 2, 3, and 4, is limited to no more than one pair of containment purge lines, that includes one set of supply valves and one set of exhaust valves. The containment purge valves are capable of closing in the environment following a LOCA. Therefore, these valves are allowed to be open for limited periods of time.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.6.3.2

This SR requires verification that each containment isolation manual valve and blind flange located outside containment and not locked, sealed, or otherwise secured and required to be closed during accident conditions is closed. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside of the containment boundary is within design limits. This SR does not require any testing or valve manipulation. Rather, it involves verification that those containment isolation valves outside containment and capable of being mispositioned are in the correct position.

## BASES

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### SURVEILLANCE REQUIREMENTS (continued)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

The SR specifies that containment isolation valves that are open under administrative controls are not required to meet the SR during the time the valves are open. This SR does not apply to valves that are locked, sealed, or otherwise secured in the closed position, since these were verified to be in the correct position upon locking, sealing, or securing.

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

#### SR 3.6.3.3

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

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

BASES

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.3.4

Verifying that the isolation time of each automatic power operated containment isolation valve is within limits is required to demonstrate OPERABILITY. The isolation time test ensures the valve will isolate in a time period less than or equal to that assumed in the safety analyses.

The isolation time and Frequency of this SR is in accordance with the Inservice Testing Program.

SR 3.6.3.5

For containment purge valves with resilient seals, additional leakage rate testing beyond the test requirements of 10 CFR 50, Appendix J, Option B, is required to ensure OPERABILITY. Operating experience has demonstrated that this type of seal has the potential to degrade in a shorter time period than do other seal types.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.6.3.6

Automatic containment isolation valves close on a containment isolation signal to prevent leakage of radioactive material from containment following a DBA. This SR ensures that each automatic containment isolation valve will actuate to its isolation position on a containment isolation signal. This surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.6.3.7

Verifying that each containment purge valve is blocked to restrict opening to < 50 degrees is required to ensure that the valves can close under DBA conditions within the times assumed in the analyses of References 1 and 2. If a LOCA occurs, the purge valves must close to maintain

BASES

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SURVEILLANCE REQUIREMENTS (continued)

containment leakage within the values assumed in the accident analysis. At other times when purge valves are required to be capable of closing (e.g., during movement of irradiated fuel assemblies), pressurization concerns are not present, thus the purge valves can be fully open.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.6.3.8

This SR ensures that the combined leakage rate of all shield building bypass leakage paths (those paths that would potentially allow leakage from the primary containment to circumvent the annulus secondary containment enclosure and escape to the auxiliary building secondary enclosure) is less than or equal to the specified leakage rate. This provides assurance that the assumptions in the safety analysis are met. The as-left bypass leakage rate prior to the first startup after performing a leakage test requires calculation using maximum pathway leakage (leakage through the worse of the two isolation valves). If the penetration is isolated by use of one closed and de-activated automatic valve, closed manual valve, or blind flange, then 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. Following startup, the as-found leakage rate will be calculated using the minimum pathway leakage. The Frequency is required by the Containment Leakage Rate Testing Program. This SR simply imposes additional acceptance criteria.

Bypass leakage is considered part of  $L_a$ .

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REFERENCES

1. UFSAR, Chapter 15.
  2. UFSAR, Section 6.2.4 and Table 6.2.4-1.
  3. Standard Review Plan 6.2.4.
  4. WCAP-15791-A, Rev. 2, "Risk-Informed Evaluation of Extensions to Containment Isolation Valve Completion Times," June 2008.
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Table B 3.6.3-1 (Page 1 of 17)  
Containment Isolation Valve Completion Times

UNID	Penetration	Pressure Boundary Maintained		Pressure Boundary Compromised	
		Category	Completion Time	Category	Completion Time
FCV-01-7	X-14D	2	8 hours	9	8 hours
FCV-01-14	X-14A	2	8 hours	9	8 hours
FCV-01-25	X-14C	2	8 hours	9	8 hours
FCV-01-32	X-14B	2	8 hours	9	8 hours
FCV-01-147	X-13A	2	8 hours	9	8 hours
FCV-01-148	X-13B	2	8 hours	9	8 hours
FCV-01-149	X-13C	2	8 hours	9	8 hours
FCV-01-150	X-13D	2	8 hours	9	8 hours
FCV-01-181	X-14D	2	8 hours	9	8 hours
FCV-01-182	X-14A	2	8 hours	9	8 hours
FCV-01-183	X-14C	2	8 hours	9	8 hours
FCV-01-184	X-14B	2	8 hours	9	8 hours
VLV-01-532	X-13C	2	8 hours	9	8 hours
VLV-01-534	X-13B	2	8 hours	9	8 hours
VLV-01-536	X-13A	2	8 hours	9	8 hours
VLV-01-538	X-13D	2	8 hours	9	8 hours
VLV-01-824	X-14D	2	8 hours	9	8 hours
VLV-01-825	X-14A	2	8 hours	9	8 hours
VLV-01-826	X-14C	2	8 hours	9	8 hours
VLV-01-827	X-14B	2	8 hours	9	8 hours
VLV-01-922	X-13A	2	8 hours	9	8 hours
VLV-01-923	X-13B	2	8 hours	9	8 hours
VLV-01-924	X-13C	2	8 hours	9	8 hours
VLV-01-925	X-13D	2	8 hours	9	8 hours
VLV-03-351C	X-104	2	8 hours	9	8 hours
VLV-03-352C	X-102	2	8 hours	9	8 hours
VLV-03-500	X-12C	2	8 hours	9	8 hours
VLV-03-502	X-12B	2	8 hours	9	8 hours
VLV-03-842	X-40B	1	4 hours	8	4 Hours
VLV-03-847	X-40B	1	4 hours	8	4 Hours
VLV-03-848	X-40A	1	4 hours	8	4 Hours
VLV-03-849	X-12A	2	8 hours	9	8 hours

Table B 3.6.3-1 (Page 2 of 17)  
Containment Isolation Valve Completion Times

UNID	Penetration	Pressure Boundary Maintained		Pressure Boundary Compromised	
		Category	Completion Time	Category	Completion Time
VLV-03-850	X-12D	2	8 hours	9	8 hours
VLV-03-851	X-40B	1	4 hours	8	4 Hours
VLV-03-852	X-40A	1	4 hours	8	4 Hours
VLV-03-853	X-12A	2	8 hours	9	8 hours
VLV-03-854	X-12D	2	8 hours	9	8 hours
VLV-03-855	X-40B	1	4 hours	8	4 Hours
VLV-03-857	X-12A	2	8 hours	9	8 hours
VLV-03-858	X-12D	2	8 hours	9	8 hours
VLV-03-859	X-40B	1	4 hours	8	4 Hours
VLV-03-860	X-40A	1	4 hours	8	4 Hours
VLV-03-887	X-40B	1	4 hours	8	4 Hours
VLV-03-888	X-40A	1	4 hours	8	4 Hours
VLV-03-889	X-12A	2	8 hours	9	8 hours
VLV-03-890	X-12D	2	8 hours	9	8 hours
VLV-03-896	X-40B	1	4 hours	8	4 Hours
VLV-03-897	X-40B	1	4 hours	8	4 Hours
VLV-03-899	X-40A	1	4 hours	8	4 Hours
VLV-03-900	X-40A	1	4 hours	8	4 Hours
VLV-03-901	X-40A	1	4 hours	8	4 Hours
VLV-03-903	X-12A	2	8 hours	9	8 hours
VLV-03-904	X-12A	2	8 hours	9	8 hours
VLV-03-906	X-12D	2	8 hours	9	8 hours
VLV-03-907	X-12D	2	8 hours	9	8 hours
FCV-26-240	X-51	7	7 days	14	7 days
FCV-26-243	X-78	7	7 days	14	7 days
VLV-26-1258	X-51	7	7 days	14	7 days
VLV-26-1260	X-51	7	7 days	14	7 days
VLV-26-1293	X-78	7	7 days	14	7 days
VLV-26-1296	X-78	7	7 days	14	7 days
DRIV-30-30CX	X-27A	7	7 days	14	7 days
DRIV-30-30CY	X-27A	7	7 days	14	7 days
DRIV-30-42X	X-27B	7	7 days	14	7 days

Table B 3.6.3-1 (Page 3 of 17)  
Containment Isolation Valve Completion Times

UNID	Penetration	Pressure Boundary Maintained		Pressure Boundary Compromised	
		Category	Completion Time	Category	Completion Time
DRIV-30-42Y	X-27B	7	7 days	14	7 days
DRIV-30-43X	X-26A	7	7 days	14	7 days
DRIV-30-43Y	X-26A	7	7 days	14	7 days
DRIV-30-44X	X-25B	7	7 days	14	7 days
DRIV-30-44Y	X-25B	7	7 days	14	7 days
DRIV-30-45X	X-85B	7	7 days	14	7 days
DRIV-30-45Y	X-85B	7	7 days	14	7 days
DRIV-30-46AX	X-111	7	7 days	14	7 days
DRIV-30-46AY	X-111	7	7 days	14	7 days
DRIV-30-46BY	X-111	7	7 days	14	7 days
DRIV-30-47AX	X-112	7	7 days	14	7 days
DRIV-30-47AY	X-112	7	7 days	14	7 days
DRIV-30-47BY	X-112	7	7 days	14	7 days
DRIV-30-48AX	X-113	7	7 days	14	7 days
DRIV-30-48AY	X-113	7	7 days	14	7 days
DRIV-30-48BY	X-113	7	7 days	14	7 days
DRIV-30-310X	X-26A	7	7 days	14	7 days
DRIV-30-310Y	X-26A	7	7 days	14	7 days
DRIV-30-311X	X-25B	7	7 days	14	7 days
DRIV-30-311Y	X-25B	7	7 days	14	7 days
FCV-30-7	X-9A	7	7 days	14	7 days
FCV-30-8	X-9A	7	7 days	14	7 days
FCV-30-9	X-9B	7	7 days	14	7 days
FCV-30-10	X-9B	7	7 days	14	7 days
FCV-30-14	X-10A	7	7 days	14	7 days
FCV-30-15	X-10A	7	7 days	14	7 days
FCV-30-16	X-10B	7	7 days	14	7 days
FCV-30-17	X-10B	7	7 days	14	7 days
FCV-30-19	X-11	7	7 days	14	7 days
FCV-30-20	X-11	7	7 days	14	7 days
FCV-30-37	X-80	7	7 days	14	7 days
FCV-30-40	X-80	7	7 days	14	7 days

Table B 3.6.3-1 (Page 4 of 17)  
Containment Isolation Valve Completion Times

UNID	Penetration	Pressure Boundary Maintained		Pressure Boundary Compromised	
		Category	Completion Time	Category	Completion Time
FCV-30-46	X-111	7	7 days	14	7 days
FCV-30-47	X-112	7	7 days	14	7 days
FCV-30-48	X-113	7	7 days	14	7 days
FCV-30-50	X-6	7	7 days	14	7 days
FCV-30-51	X-6	7	7 days	14	7 days
FCV-30-52	X-7	7	7 days	14	7 days
FCV-30-53	X-7	7	7 days	14	7 days
FCV-30-56	X-4	7	7 days	14	7 days
FCV-30-57	X-4	7	7 days	14	7 days
FCV-30-58	X-5	7	7 days	14	7 days
FCV-30-59	X-5	7	7 days	14	7 days
FSV-30-134	X-97	7	7 days	14	7 days
FSV-30-135	X-97	7	7 days	14	7 days
PDT-30-30C	X-27A	7	7 days	14	7 days
PDT-30-42	X-27B	7	7 days	14	7 days
PDT-30-43	X-26A	7	7 days	14	7 days
PDT-30-44	X-25B	7	7 days	14	7 days
PDT-30-45	X-85B	7	7 days	14	7 days
PDT-30-310	X-26A	7	7 days	14	7 days
PDT-30-311	X-25B	7	7 days	14	7 days
VLV-30-554TP	X-5	7	7 days	14	7 days
VLV-30-555TP	X-4	7	7 days	14	7 days
VLV-30-556TP	X-80	7	7 days	14	7 days
VLV-30-557TP	X-7	7	7 days	14	7 days
VLV-30-558TP	X-6	7	7 days	14	7 days
VLV-30-559TP	X-11	7	7 days	14	7 days
VLV-30-560TP	X-10B	7	7 days	14	7 days
VLV-30-561TP	X-10A	7	7 days	14	7 days
VLV-30-562TP	X-9B	7	7 days	14	7 days
VLV-30-563TP	X-9A	7	7 days	14	7 days
VLV-30-571	X-111	7	7 days	14	7 days
VLV-30-572	X-112	7	7 days	14	7 days

Table B 3.6.3-1 (Page 5 of 17)  
Containment Isolation Valve Completion Times

UNID	Penetration	Pressure Boundary Maintained		Pressure Boundary Compromised	
		Category	Completion Time	Category	Completion Time
VLV-30-573	X-113	7	7 days	14	7 days
FCV-31C-222	X-64	7	7 days	14	7 days
FCV-31C-223	X-64	7	7 days	14	7 days
FCV-31C-224	X-65	7	7 days	14	7 days
FCV-31C-225	X-65	7	7 days	14	7 days
FCV-31C-229	X-66	7	7 days	14	7 days
FCV-31C-230	X-66	7	7 days	14	7 days
FCV-31C-231	X-67	7	7 days	14	7 days
FCV-31C-232	X-67	7	7 days	14	7 days
VLV-31C-697	X-67	7	7 days	14	7 days
VLV-31C-715	X-66	7	7 days	14	7 days
VLV-31C-734	X-65	7	7 days	14	7 days
VLV-31C-752	X-64	7	7 days	14	7 days
FCV-32-80	X-90	7	7 days	14	7 days
FCV-32-102	X-26B	7	7 days	14	7 days
FCV-32-110	X-34	7	7 days	14	7 days
VLV-32-281	X-90	7	7 days	14	7 days
VLV-32-285	X-90	7	7 days	14	7 days
VLV-32-287	X-90	7	7 days	14	7 days
VLV-32-292	X-26B	7	7 days	14	7 days
VLV-32-295	X-26B	7	7 days	14	7 days
VLV-32-297	X-26B	7	7 days	14	7 days
VLV-32-373	X-34	7	7 days	14	7 days
VLV-32-375	X-34	7	7 days	14	7 days
VLV-32-377	X-34	7	7 days	14	7 days
BLF-Sys 33	X-40D	7	7 days	14	7 days
VLV-33-212	X-76	7	7 days	14	7 days
VLV-33-704	X-76	7	7 days	14	7 days
VLV-33-740	X-76	7	7 days	14	7 days
CKV-43-460	X-106	7	7 days	14	7 days
CKV-43-461	X-103	7	7 days	14	7 days
FSV-43-2	X-25A	7	7 days	14	7 days

Table B 3.6.3-1 (Page 6 of 17)  
Containment Isolation Valve Completion Times

UNID	Penetration	Pressure Boundary Maintained		Pressure Boundary Compromised	
		Category	Completion Time	Category	Completion Time
FSV-43-3	X-25A	7	7 days	14	7 days
FSV-43-11	X-25D	7	7 days	14	7 days
FSV-43-12	X-25D	7	7 days	14	7 days
FSV-43-22	X-96C	7	7 days	14	7 days
FSV-43-23	X-96C	7	7 days	14	7 days
FSV-43-34	X-93	7	7 days	14	7 days
FSV-43-35	X-93	7	7 days	14	7 days
FCV-43-55	X-14D	2	8 hours	9	8 hours
FCV-43-58	X-14A	2	8 hours	9	8 hours
FCV-43-61	X-14C	2	8 hours	9	8 hours
FCV-43-64	X-14B	2	8 hours	9	8 hours
FSV-43-201	X-99, X-100	7	7 days	14	7 days
FSV-43-202	X-99, X-100	7	7 days	14	7 days
FSV-43-207	X-92A, X-92B	7	7 days	14	7 days
FSV-43-208	X-92A, X-92B	7	7 days	14	7 days
FSV-43-250	X-91	7	7 days	14	7 days
FSV-43-251	X-91	7	7 days	14	7 days
FSV-43-287	X-116A	7	7 days	14	7 days
FSV-43-288	X-116A	7	7 days	14	7 days
FSV-43-307	X-106	7	7 days	14	7 days
FSV-43-309	X-23	7	7 days	14	7 days
FSV-43-310	X-23	7	7 days	14	7 days
FSV-43-317	X-103	7	7 days	14	7 days
FSV-43-318	X-101	7	7 days	14	7 days
FSV-43-319	X-101	7	7 days	14	7 days
FSV-43-325	X-106	7	7 days	14	7 days
FSV-43-341	X-103	7	7 days	14	7 days
FSV-43-450	X-99, X-100	7	7 days	14	7 days
FSV-43-451	X-99, X-100	7	7 days	14	7 days
FSV-43-452	X-92A, X-92B	7	7 days	14	7 days

Table B 3.6.3-1 (Page 7 of 17)  
Containment Isolation Valve Completion Times

UNID	Penetration	Pressure Boundary Maintained		Pressure Boundary Compromised	
		Category	Completion Time	Category	Completion Time
FSV-43-453	X-92A, X-92B	7	7 days	14	7 days
TV-43-464	X-103	7	7 days	14	7 days
TV-43-469	X-106	7	7 days	14	7 days
TV-43-474	X-101	7	7 days	14	7 days
TV-43-477	X-116A	7	7 days	14	7 days
VLV-43-423	X-92A, X-92B	7	7 days	14	7 days
VLV-43-424	X-92A, X-92B	7	7 days	14	7 days
VLV-43-425	X-99, X-100	7	7 days	14	7 days
VLV-43-426	X-99, X-100	7	7 days	14	7 days
VLV-43-492	X-23	7	7 days	14	7 days
VLV-43-497	X-91	7	7 days	14	7 days
TTIV-52-508	X-98	7	7 days	14	7 days
TTIV-52-510	X-27C	7	7 days	14	7 days
VLV-52-500	X-87D	7	7 days	14	7 days
VLV-52-501	X-87D	7	7 days	14	7 days
VLV-52-502	X-87B	7	7 days	14	7 days
VLV-52-503	X-87B	7	7 days	14	7 days
VLV-52-504	X-27C	7	7 days	14	7 days
VLV-52-505	X-27C	7	7 days	14	7 days
VLV-52-506	X-98	7	7 days	14	7 days
VLV-52-507	X-98	7	7 days	14	7 days
VLV-59-522	X-77	7	7 days	14	7 days
VLV-59-529	X-77	7	7 days	14	7 days
VLV-59-633	X-77	7	7 days	14	7 days
VLV-59-651	X-77	7	7 days	14	7 days
VLV-59-704	X-77	7	7 days	14	7 days
BLF-Sys 61	X-79A	7	7 days	13	72 hours
BLF-Sys 61	X-79B	7	7 days	13	72 hours
FCV-61-96	X-115	7	7 days	14	7 days
FCV-61-97	X-115	7	7 days	14	7 days

Table B 3.6.3-1 (Page 8 of 17)  
Containment Isolation Valve Completion Times

UNID	Penetration	Pressure Boundary Maintained		Pressure Boundary Compromised	
		Category	Completion Time	Category	Completion Time
FCV-61-110	X-114	7	7 days	14	7 days
FCV-61-122	X-114	7	7 days	14	7 days
FCV-61-191	X-47A	7	7 days	14	7 days
FCV-61-192	X-47A	7	7 days	14	7 days
FCV-61-193	X-47B	7	7 days	14	7 days
FCV-61-194	X-47B	7	7 days	14	7 days
VLV-61-532	X-47A	7	7 days	14	7 days
VLV-61-533	X-47A	7	7 days	14	7 days
VLV-61-680	X-47B	7	7 days	14	7 days
VLV-61-681	X-47B	7	7 days	14	7 days
VLV-61-691	X-115	7	7 days	14	7 days
VLV-61-692	X-115	7	7 days	14	7 days
VLV-61-745	X-114	7	7 days	14	7 days
VLV-61-746	X-114	7	7 days	14	7 days
FCV-62-61	X-44	7	7 days	14	7 days
FCV-62-63	X-44	7	7 days	14	7 days
FCV-62-72	X-15	7	7 days	14	7 days
FCV-62-73	X-15	7	7 days	14	7 days
FCV-62-74	X-15	7	7 days	14	7 days
FCV-62-77	X-15	7	7 days	14	7 days
FCV-62-90	X-16	7	7 days	14	7 days
VLV-62-505	X-24	7	7 days	14	7 days
VLV-62-543	X-16	7	7 days	14	7 days
VLV-62-544	X-16	7	7 days	14	7 days
VLV-62-546	X-43A, X-43B, X-43C, X43D	7	7 days	14	7 days
VLV-62-549	X-43A, X-43B, X-43C, X43D	7	7 days	14	7 days
VLV-62-550	X-43A, X-43B, X-43C, X43D	7	7 days	14	7 days

Table B 3.6.3-1 (Page 9 of 17)  
Containment Isolation Valve Completion Times

UNID	Penetration	Pressure Boundary Maintained		Pressure Boundary Compromised	
		Category	Completion Time	Category	Completion Time
VLV-62-555	X-43A, X-43B, X-43C, X43D	7	7 days	14	7 days
VLV-62-560	X-43D	7	7 days	14	7 days
VLV-62-561	X-43B	7	7 days	14	7 days
VLV-62-562	X-43C	7	7 days	14	7 days
VLV-62-563	X-43A	7	7 days	14	7 days
VLV-62-568	X-43D	7	7 days	14	7 days
VLV-62-569	X-43B	7	7 days	14	7 days
VLV-62-570	X-43C	7	7 days	14	7 days
VLV-62-571	X-43A	7	7 days	14	7 days
VLV-62-572	X-43D	7	7 days	14	7 days
VLV-62-573	X-43B	7	7 days	14	7 days
VLV-62-574	X-43C	7	7 days	14	7 days
VLV-62-575	X-43A	7	7 days	14	7 days
VLV-62-576	X-43D	7	7 days	14	7 days
VLV-62-577	X-43B	7	7 days	14	7 days
VLV-62-578	X-43A	7	7 days	14	7 days
VLV-62-579	X-43C	7	7 days	14	7 days
VLV-62-639	X-44	7	7 days	14	7 days
VLV-62-662	X-15	7	7 days	14	7 days
VLV-62-707	X-15	7	7 days	14	7 days
VLV-62-709	X-16	7	7 days	14	7 days
FCV-63-21	X-32	7	7 days	14	7 days
FCV-63-22	X-33	1	4 hours	8	4 hours
FCV-63-23	X-30	7	7 days	14	7 days
FCV-63-25	X-22	1	4 hours	8	4 hours
FCV-63-26	X-22	1	4 hours	8	4 hours
FCV-63-64	X-39A	7	7 days	14	7 days
FCV-63-71	X-30	7	7 days	14	7 days
FCV-63-84	X-30	7	7 days	14	7 days
FCV-63-93	X-20B	7	7 days	14	7 days

Table B 3.6.3-1 (Page 10 of 17)  
Containment Isolation Valve Completion Times

UNID	Penetration	Pressure Boundary Maintained		Pressure Boundary Compromised	
		Category	Completion Time	Category	Completion Time
FCV-63-94	X-20A	7	7 days	14	7 days
FCV-63-111	X-20B	7	7 days	14	7 days
FCV-63-112	X-20A	7	7 days	14	7 days
FCV-63-121	X-33	7	7 days	14	7 days
FCV-63-156	X-32	7	7 days	8	4 hours
FCV-63-157	X-21	7	7 days	8	4 hours
FCV-63-158	X-17	7	7 days	14	7 days
FCV-63-167	X-21	7	7 days	14	7 days
FCV-63-172	X-17	1	4 hours	8	4 hours
FCV-63-174	X-22	7	7 days	14	7 days
FSV-63-25	X-22	7	7 days	14	7 days
FSV-63-26	X-22	7	7 days	14	7 days
VLV-63-311A	X-32	7	7 days	14	7 days
VLV-63-313A	X-21	7	7 days	14	7 days
VLV-63-314A	X-21	7	7 days	14	7 days
VLV-63-315A	X-32	7	7 days	14	7 days
VLV-63-316A	X-32	7	7 days	14	7 days
VLV-63-317A	X-21	7	7 days	14	7 days
VLV-63-318A	X-21	7	7 days	14	7 days
VLV-63-319A	X-33	7	7 days	14	7 days
VLV-63-320A	X-33	7	7 days	14	7 days
VLV-63-321A	X-33	7	7 days	14	7 days
VLV-63-322A	X-33	7	7 days	14	7 days
VLV-63-323A	X-33	7	7 days	14	7 days
VLV-63-324A	X-33	7	7 days	14	7 days
VLV-63-325A	X-33	7	7 days	14	7 days
VLV-63-326A	X-33	7	7 days	14	7 days
VLV-63-344A	X-30	7	7 days	14	7 days
VLV-63-413	X-20B	7	7 days	14	7 days
VLV-63-511	X-24	7	7 days	14	7 days
VLV-63-534	X-24	7	7 days	14	7 days
VLV-63-535	X-24	7	7 days	14	7 days

Table B 3.6.3-1 (Page 11 of 17)  
Containment Isolation Valve Completion Times

UNID	Penetration	Pressure Boundary Maintained		Pressure Boundary Compromised	
		Category	Completion Time	Category	Completion Time
VLV-63-536	X-24	7	7 days	14	7 days
VLV-63-537	X-30	7	7 days	14	7 days
VLV-63-541	X-32	7	7 days	14	7 days
VLV-63-543	X-32	7	7 days	8	4 hours
VLV-63-545	X-32	7	7 days	8	4 hours
VLV-63-547	X-21	7	7 days	8	4 hours
VLV-63-549	X-21	7	7 days	8	4 hours
VLV-63-551	X-33	1	4 hours	8	4 hours
VLV-63-553	X-33	1	4 hours	8	4 hours
VLV-63-555	X-33	1	4 hours	8	4 hours
VLV-63-557	X-33	1	4 hours	8	4 hours
VLV-63-581	X-22	1	4 hours	8	4 hours
VLV-63-590	X-17	7	7 days	14	7 days
VLV-63-591	X-17	7	7 days	14	7 days
VLV-63-592	X-17	7	7 days	14	7 days
VLV-63-593	X-17	7	7 days	14	7 days
VLV-63-612A	X-32	7	7 days	14	7 days
VLV-63-626	X-24	7	7 days	14	7 days
VLV-63-627	X-24	7	7 days	14	7 days
VLV-63-630	X-20B	7	7 days	14	7 days
VLV-63-631	X-20A	7	7 days	14	7 days
VLV-63-632	X-20B	7	7 days	14	7 days
VLV-63-633	X-20A	7	7 days	14	7 days
VLV-63-634	X-20B	7	7 days	14	7 days
VLV-63-635	X-20A	7	7 days	14	7 days
VLV-63-636	X-17	7	7 days	14	7 days
VLV-63-637	X-17	7	7 days	14	7 days
VLV-63-638	X-24	7	7 days	14	7 days
VLV-63-640	X-17	1	4 hours	8	4 hours
VLV-63-642	X-17	7	7 days	14	7 days
VLV-63-643	X-17	1	4 hours	8	4 hours
VLV-63-648	X-21	7	7 days	14	7 days

Table B 3.6.3-1 (Page 12 of 17)  
Containment Isolation Valve Completion Times

UNID	Penetration	Pressure Boundary Maintained		Pressure Boundary Compromised	
		Category	Completion Time	Category	Completion Time
VLV-63-649	X-21	7	7 days	14	7 days
VLV-63-650	X-21	7	7 days	14	7 days
VLV-63-653	X-33	7	7 days	14	7 days
VLV-63-654	X-33	7	7 days	14	7 days
VLV-63-655	X-33	7	7 days	14	7 days
VLV-63-656	X-33	7	7 days	14	7 days
VLV-63-657	X-32	7	7 days	14	7 days
VLV-63-658	X-32	7	7 days	14	7 days
VLV-63-659	X-20B	7	7 days	14	7 days
VLV-63-660	X-20B	7	7 days	14	7 days
VLV-63-661	X-20A	7	7 days	14	7 days
VLV-63-667	X-20A	7	7 days	14	7 days
VLV-63-823	X-32	7	7 days	14	7 days
VLV-63-831	X-33	7	7 days	14	7 days
VLV-63-833	X-20A	7	7 days	14	7 days
VLV-63-836	X-33	7	7 days	14	7 days
VLV-63-862	X-21	7	7 days	14	7 days
VLV-63-864	X-32	7	7 days	14	7 days
VLV-63-870	X-17	7	7 days	14	7 days
CKV-67-1523A	X-58	7	7 days	14	7 days
CKV-67-1523B	X-60	7	7 days	14	7 days
CKV-67-1523C	X-62	7	7 days	14	7 days
CKV-67-1523D	X-56	7	7 days	14	7 days
FCV-67-83	X-56	7	7 days	14	7 days
FCV-67-87	X-59	7	7 days	14	7 days
FCV-67-88	X-59	7	7 days	14	7 days
FCV-67-89	X-56	7	7 days	14	7 days
FCV-67-90	X-60	7	7 days	14	7 days
FCV-67-91	X-60	7	7 days	14	7 days
FCV-67-95	X-63	7	7 days	14	7 days
FCV-67-96	X-63	7	7 days	14	7 days
FCV-67-99	X-62	7	7 days	14	7 days

Table B 3.6.3-1 (Page 13 of 17)  
Containment Isolation Valve Completion Times

UNID	Penetration	Pressure Boundary Maintained		Pressure Boundary Compromised	
		Category	Completion Time	Category	Completion Time
FCV-67-103	X-61	7	7 days	14	7 days
FCV-67-104	X-61	7	7 days	14	7 days
FCV-67-105	X-62	7	7 days	14	7 days
FCV-67-106	X-58	7	7 days	14	7 days
FCV-67-107	X-58	7	7 days	14	7 days
FCV-67-111	X-57	7	7 days	14	7 days
FCV-67-112	X-57	7	7 days	14	7 days
VLV-67-561A	X-58	7	7 days	14	7 days
VLV-67-561B	X-60	7	7 days	14	7 days
VLV-67-561C	X-62	7	7 days	14	7 days
VLV-67-561D	X-56	7	7 days	14	7 days
VLV-67-575A	X-59	7	7 days	14	7 days
VLV-67-575B	X-61	7	7 days	14	7 days
VLV-67-575C	X-63	7	7 days	14	7 days
VLV-67-575D	X-57	7	7 days	14	7 days
VLV-67-772	X-56	7	7 days	14	7 days
VLV-67-774	X-60	7	7 days	14	7 days
VLV-67-776	X-62	7	7 days	14	7 days
VLV-67-778	X-58	7	7 days	14	7 days
FCV-68-305	X-39B	7	7 days	14	7 days
FCV-68-307	X-84A	7	7 days	14	7 days
FCV-68-308	X-84A	7	7 days	14	7 days
RVLIS-Sys 68	X-85C	7	7 days	14	7 days
RVLIS-Sys 68	X-86A	7	7 days	14	7 days
RVLIS-Sys 68	X-86B	7	7 days	14	7 days
RVLIS-Sys 68	X-86C	7	7 days	14	7 days
RVLIS-Sys 68	X-25C	7	7 days	14	7 days
RVLIS-Sys 68	X-27D	7	7 days	14	7 days
RVLIS-Sys 68	X-26C	7	7 days	14	7 days
VLV-68-559	X-24	7	7 days	14	7 days
VLV-68-560	X-24	7	7 days	14	7 days
VLV-68-561	X-24	7	7 days	14	7 days

Table B 3.6.3-1 (Page 14 of 17)  
Containment Isolation Valve Completion Times

UNID	Penetration	Pressure Boundary Maintained		Pressure Boundary Compromised	
		Category	Completion Time	Category	Completion Time
FCV-70-85	X-35	7	7 days	14	7 days
FCV-70-87	X-50A	7	7 days	14	7 days
FCV-70-89	X-29	7	7 days	14	7 days
FCV-70-90	X-50A	7	7 days	14	7 days
FCV-70-92	X-29	7	7 days	14	7 days
FCV-70-134	X-50B	7	7 days	14	7 days
FCV-70-140	X-52	7	7 days	14	7 days
FCV-70-141	X-52	7	7 days	14	7 days
FCV-70-143	X-53	7	7 days	14	7 days
VLV-70-678B	X-50B	7	7 days	14	7 days
VLV-70-679	X-50B	7	7 days	14	7 days
VLV-70-687	X-50A	7	7 days	14	7 days
VLV-70-691B	X-52	7	7 days	14	7 days
VLV-70-698	X-29	7	7 days	14	7 days
VLV-70-702B	X-53	7	7 days	14	7 days
VLV-70-702C	X-35	7	7 days	14	7 days
VLV-70-702E	X-53	7	7 days	14	7 days
VLV-70-702F	X-35	7	7 days	14	7 days
VLV-70-703	X-35, X-53	7	7 days	14	7 days
VLV-70-735	X-29	7	7 days	14	7 days
VLV-70-737	X-50A	7	7 days	14	7 days
VLV-70-760	X-53	7	7 days	14	7 days
VLV-70-762	X-35	7	7 days	14	7 days
VLV-70-764	X-35	7	7 days	14	7 days
VLV-70-765	X-53	7	7 days	14	7 days
VLV-70-791	X-52	7	7 days	14	7 days
DRIV-72-215F	X-49A	7	7 days	14	7 days
DRIV-72-216F	X-49A	7	7 days	14	7 days
DRIV-72-217F	X-49B	7	7 days	14	7 days
DRIV-72-218F	X-49B	7	7 days	14	7 days
FCV-72-2	X-48B	7	7 days	14	7 days
FCV-72-39	X-48A	7	7 days	14	7 days

Table B 3.6.3-1 (Page 15 of 17)  
Containment Isolation Valve Completion Times

UNID	Penetration	Pressure Boundary Maintained		Pressure Boundary Compromised	
		Category	Completion Time	Category	Completion Time
FCV-72-40	X-49A	7	7 days	14	7 days
FCV-72-41	X-49B	7	7 days	14	7 days
FCV-72-546	X-48B	7	7 days	14	7 days
RFV-72-40	X-49A	7	7 days	14	7 days
RFV-72-41	X-49B	7	7 days	14	7 days
TTIV-72-215E	X-49A	7	7 days	14	7 days
TTIV-72-216E	X-49A	7	7 days	14	7 days
TTIV-72-217E	X-49B	7	7 days	14	7 days
TTIV-72-218E	X-49B	7	7 days	14	7 days
VLV-72-512	X-24	7	7 days	14	7 days
VLV-72-513	X-24	7	7 days	14	7 days
VLV-72-517	X-24	7	7 days	14	7 days
VLV-72-518	X-24	7	7 days	14	7 days
VLV-72-543	X-48A	7	7 days	14	7 days
VLV-72-544	X-48B	7	7 days	14	7 days
VLV-72-545	X-48A	7	7 days	14	7 days
VLV-72-547	X-48A	7	7 days	14	7 days
VLV-72-548	X-48B	7	7 days	14	7 days
VLV-72-551	X-49B	7	7 days	14	7 days
VLV-72-552	X-49A	7	7 days	14	7 days
VLV-72-555	X-49B	7	7 days	14	7 days
VLV-72-556	X-49A	7	7 days	14	7 days
FCV-74-1	X-107	1	4 hours	8	4 hours
FCV-74-2	X-107	1	4 hours	8	4 hours
VLV-74-503	X-107	7	7 days	14	7 days
VLV-74-504	X-107	7	7 days	14	7 days
VLV-74-505	X-107	1	4 hours	8	4 hours
VLV-74-549	X-107	7	7 days	14	7 days
FCV-77-9	X-46	7	7 days	14	7 days
FCV-77-10	X-46	7	7 days	14	7 days
FCV-77-18	X-45	7	7 days	14	7 days
FCV-77-19	X-45	7	7 days	14	7 days

Table B 3.6.3-1 (Page 16 of 17)  
Containment Isolation Valve Completion Times

UNID	Penetration	Pressure Boundary Maintained		Pressure Boundary Compromised	
		Category	Completion Time	Category	Completion Time
FCV-77-20	X-45	7	7 days	14	7 days
FCV-77-127	X-41	7	7 days	14	7 days
FCV-77-128	X-41	7	7 days	14	7 days
VLV-77-848	X-39B	7	7 days	14	7 days
VLV-77-849	X-39B	7	7 days	14	7 days
VLV-77-867	X-39A	7	7 days	14	7 days
VLV-77-868	X-39A	7	7 days	14	7 days
VLV-77-984	X-45	7	7 days	14	7 days
VLV-78-226A	X-83	7	7 days	14	7 days
VLV-78-228A	X-82	7	7 days	14	7 days
VLV-78-557	X-83	6	72 hours	13	72 hours
VLV-78-558	X-83	6	72 hours	13	72 hours
VLV-78-560	X-82	6	72 hours	13	72 hours
VLV-78-561	X-82	6	72 hours	13	72 hours
FCV-81-12	X-42	7	7 days	14	7 days
VLV-81-502	X-42	7	7 days	14	7 days
VLV-81-529	X-42	7	7 days	14	7 days
VLV-84-511	X-46	7	7 days	14	7 days
BLF-Sys 88	X-54	7	7 days	13	72 hours
BLF-Sys 88	X-88	7	7 days	14	7 days
BLF-Sys 88	X-108	1	4 hours	8	4 hours
BLF-Sys 88	X-109	1	4 hours	8	4 hours
BLF-Sys 88	X-117	6	72 hours	13	72 hours
BLF-Sys 88	X-118	6	72 hours	13	72 hours
FCV-90-107	X-94A	7	7 days	14	7 days
FCV-90-107	X-94B	7	7 days	14	7 days
FCV-90-108	X-94B	7	7 days	14	7 days
FCV-90-109	X-94A	7	7 days	14	7 days
FCV-90-110	X-94C	7	7 days	14	7 days
FCV-90-111	X-94C	7	7 days	14	7 days
FCV-90-113	X-95A	7	7 days	14	7 days
FCV-90-113	X-95B	7	7 days	14	7 days

Table B 3.6.3-1 (Page 17 of 17)  
Containment Isolation Valve Completion Times

UNID	Penetration	Pressure Boundary Maintained		Pressure Boundary Compromised	
		Category	Completion Time	Category	Completion Time
FCV-90-114	X-95B	7	7 days	14	7 days
FCV-90-115	X-95A	7	7 days	14	7 days
FCV-90-116	X-95C	7	7 days	14	7 days
FCV-90-117	X-95C	7	7 days	14	7 days

## B 3.6 CONTAINMENT SYSTEMS

### B 3.6.4 Containment Pressure

#### BASES

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**BACKGROUND** The containment pressure is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a loss of coolant accident (LOCA) or steam line break (SLB). These limits also prevent the containment pressure from exceeding the containment design negative pressure differential with respect to the annulus pressure 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. Should operation occur outside these limits coincident with a Design Basis Accident (DBA), post accident containment pressures could exceed calculated values.

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**APPLICABLE SAFETY ANALYSES** Containment internal pressure is an initial condition used in the DBA analyses to establish the maximum peak containment internal pressure. The limiting DBAs considered, relative to containment pressure, are the LOCA and SLB, which are analyzed using computer pressure transients. The worst case LOCA generates larger mass and energy release than the worst case SLB. Thus, the LOCA event bounds the SLB event from the containment peak pressure standpoint (Ref. 1).

The initial pressure condition used in the containment analysis was 0.3 psig. This resulted in a maximum peak compression pressure of 7.18 psig in the upper containment from a LOCA. The containment analysis (Ref. 1) shows that the maximum peak calculated containment pressure,  $P_a$ , results from the limiting LOCA. The maximum containment pressure resulting from the worst case LOCA, 11.44 psig, does not exceed the containment design pressure, 12 psig.

The containment was also designed for an external pressure load equivalent to 0.5 psig. The inadvertent actuation of the Containment Spray System and Air Return System was analyzed to determine the resulting reduction in containment pressure. The initial pressure condition used in this analysis was 0.1 psi less than annulus pressure. This resulted in a minimum pressure inside containment of 0.49 psi less than annulus pressure, which is less than the design load.

BASES

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APPLICABLE SAFETY ANALYSES (continued)

For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the cooling effectiveness of the Emergency Core Cooling System during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. Therefore, for the reflood phase, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the containment pressure response in accordance with 10 CFR 50, Appendix K (Ref. 2).

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

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LCO

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

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APPLICABILITY

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

---

ACTIONS

A.1

When containment pressure is not within the limits of the LCO, it 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.

BASES

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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.

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SURVEILLANCE  
REQUIREMENTS

SR 3.6.4.1

Verifying that containment pressure is within limits ensures that unit operation remains within the limits assumed in the containment analysis.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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REFERENCES

1. UFSAR, Section 6.2.
  2. 10 CFR 50, Appendix K.
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## B 3.6 CONTAINMENT SYSTEMS

### B 3.6.5 Containment Air Temperature

#### BASES

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**BACKGROUND** The containment structure serves to contain radioactive material 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 steam line break (SLB).

The containment average air temperature limit is derived from the input conditions used in the containment functional analyses and the containment structure external pressure analyses. This LCO ensures that initial conditions assumed in the analysis of containment response to a DBA are not violated during 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 upon the energy released to the containment due to the event, as well as the initial containment temperature and pressure. The higher the initial temperature, the more energy that must be removed, resulting in a higher peak containment pressure and temperature. Exceeding containment design pressure may result in leakage greater than that assumed in the accident analysis. Operation with containment temperature in excess of the LCO limit violates an initial condition assumed in the accident analysis.

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**APPLICABLE SAFETY ANALYSES** Containment average air temperature is an initial condition used in the DBA analyses that establishes the containment environmental qualification operating envelope for both pressure and temperature. The limit for containment average air temperature ensures that operation is maintained within the assumptions used in the DBA analyses for containment (Ref. 1).

The limiting DBAs considered relative to containment OPERABILITY are the LOCA and SLB. The DBA LOCA and SLB are analyzed using computer codes designed to predict the resultant containment pressure transients. No two DBAs are assumed to occur simultaneously or consecutively. The postulated DBAs are analyzed with regard to Engineered Safety Feature (ESF) systems, assuming the loss of one ESF bus, which is the worst case single active failure, resulting in one train each of Containment Spray System, Residual Heat Removal System, and Air Return System being rendered inoperable.

The limiting DBA for the maximum peak containment air temperature is an SLB. For the upper compartment, the initial containment average air temperature assumed in the design basis analyses (Ref. 1) is 85°F.

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BASES

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APPLICABLE SAFETY ANALYSES (continued)

For the lower compartment, the initial average containment air temperature assumed in the design basis analyses is 120°F. However, a sensitivity analysis performed on the steam line break containment response analysis (Ref. 2) indicates that an increase of 5°F in the initial lower containment temperature to 125°F would net a 0.1°F increase in the calculated peak temperature in the lower containment. This resulted in a maximum containment air temperature of 325.6°F. The design temperature is 327°F.

The temperature upper limits are used to establish the environmental qualification operating envelope for both containment compartments. The maximum peak containment air temperature for both containment compartments was calculated to not exceed the containment design temperature. The basis of the containment design temperature, however, is to ensure the performance of safety related equipment inside containment (Ref. 3). Therefore, it is concluded that the calculated transient containment air temperatures are acceptable for the DBA SLB.

The temperature upper limits are also used in the depressurization analyses to ensure that the minimum pressure limit is maintained following an inadvertent actuation of the Containment Spray System for both containment compartments.

The containment pressure transient is sensitive to the initial air mass in containment and, therefore, to the initial containment air temperature. The limiting DBA for establishing the maximum peak containment internal pressure is a LOCA. The temperature lower limits, 85°F for the upper compartment and 100°F for the lower compartment, are used in this analysis to ensure that, in the event of an accident, the maximum containment internal pressure will not be exceeded in either containment compartment.

Containment average air temperature satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

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LCO

During a DBA, with an initial containment average air temperature within the LCO temperature limits, the resultant accident temperature profile assures that the containment structural temperature is maintained below its design temperature and that required safety related equipment will continue to perform its function. In MODES 2, 3, and 4, containment air temperature may be as low as 60°F because the resultant calculated peak containment accident pressure would not exceed the design pressure due to a lesser amount of energy released from the pipe break in these MODES.

BASES

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

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ACTIONS

A.1

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

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SURVEILLANCE  
REQUIREMENTS

SR 3.6.5.1 and SR 3.6.5.2

Verifying that containment average air temperature is within the LCO limits ensures that containment operation remains within the limits assumed for the containment analyses. In order to determine the containment average air temperature, the primary containment upper and lower compartment average air temperatures are the weighted average of the ambient air temperature monitoring stations located in the upper and lower compartment, respectively. The weighted average is the sum of each temperature multiplied by its respective containment volume fraction. In the event of inoperable temperature sensor(s), the weighted average shall be taken as the reduced total divided by one minus the volume fraction represented by the sensor(s) out of service. As a minimum, the temperature readings for the upper compartment average air temperature shall be obtained from Elevation 743 feet (ft), Elevation 786 ft, and either Elevation 786 or 845 ft. Additionally, as a minimum, the temperature readings for the lower compartment average air temperature shall be obtained from Elevation 722 ft, Elevation 700 ft, and either Elevation 685 or 703 ft.

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BASES

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SURVEILLANCE REQUIREMENTS (continued)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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REFERENCES

1. UFSAR, Section 6.2.
  2. LTR-CRA-11-201, Westinghouse Memo – Sequoyah Units 1 and 2 Steamline Break Containment Response Sensitivity Analysis Addressing an Increase in the Initial Temperature in the Lower Containment, August 5, 2011.
  3. 10 CFR 50.49.
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## B 3.6 CONTAINMENT SYSTEMS

### B 3.6.6 Containment Spray System

#### BASES

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**BACKGROUND** The Containment Spray System provides 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 System is designed to meet the requirements of 10 CFR 50, Appendix A, GDC 38, "Containment Heat Removal," GDC 39, "Inspection of Containment Heat Removal Systems," and GDC 40, "Testing of Containment Heat Removal Systems" (Ref. 1).

The Containment Spray System consists of two separate subsystems of equal capacity, each capable of meeting the system design basis spray coverage. A containment spray subsystem contains one containment spray train and one RHR spray train. Each containment spray train includes a containment spray pump, one containment spray heat exchanger, spray headers, nozzles, valves, and piping. Each containment spray train is powered from a separate Engineered Safety Feature (ESF) bus. The refueling water storage tank (RWST) supplies borated water to the Containment Spray System during the injection phase of operation. In the recirculation mode of operation, containment spray pump suction is transferred from the RWST to the containment recirculation sump(s).

The diversion of a portion of the recirculation flow from each train of the Residual Heat Removal (RHR) System to additional redundant spray headers completes the Containment Spray System heat removal capability. Each RHR spray train is capable of supplying spray coverage, if required, to supplement the Containment Spray System.

The containment spray train and RHR spray train provide a spray of cold or subcooled borated water into the upper region of containment to limit the containment pressure and temperature during a DBA. The RWST solution temperature is an important factor in determining the heat removal capability of the Containment Spray System during the injection phase. In the recirculation mode of operation, heat is removed from the containment sump water by the Containment Spray System heat exchangers. Each containment spray subsystem provides adequate spray coverage to meet the system design requirements for containment heat removal.

## BASES

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### BACKGROUND (continued)

The containment spray trains are actuated either automatically by a High – High containment pressure signal or manually. An automatic actuation opens the containment spray pump discharge valves, starts the two containment spray pumps, and begins the injection phase. A manual actuation of the containment spray trains requires the operator to actuate two separate switches on the main control board to begin the same sequence. The Low-Low alarm for the RWST or a high level in the containment sump signals the operator to manually align the system to the recirculation mode. Operation of the Containment Spray System in the recirculation mode is controlled by the operator in accordance with the emergency operation procedures.

The RHR spray trains are initiated manually, when required by the emergency operating procedures, after the Emergency Core Cooling System (ECCS) is operating in the recirculation mode. The RHR spray trains are available to supplement the containment spray trains, if required, in limiting containment pressure. This additional spray capacity would typically be used after the ice bed has been depleted and in the event that containment pressure rises above a predetermined limit. The Containment Spray System is an ESF system. It is designed to ensure that the heat removal capability required during the post accident period can be attained.

The operation of the Containment Spray System, together with the ice condenser, is adequate to assure pressure suppression during the initial blowdown of steam and water from a DBA. During the post blowdown period, the Air Return System (ARS) is automatically started. The ARS returns upper compartment air through the divider barrier to the lower compartment. This serves to continue circulating heated air and steam through the ice condenser, where heat is removed by the remaining ice.

The Containment Spray System limits the temperature and pressure that could be expected following a DBA. Protection of containment integrity limits leakage of fission product radioactivity from containment to the environment.

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### APPLICABLE SAFETY ANALYSES

The limiting DBAs considered relative to containment OPERABILITY are the loss of coolant accident (LOCA) and the steam line break (SLB). The DBA LOCA and SLB are analyzed using computer codes designed to predict the resultant containment pressure and temperature transients. No two DBAs are assumed to occur simultaneously or consecutively. The postulated DBAs are analyzed, in regard to containment ESF systems, assuming the loss of one ESF bus, which is the worst case single active failure, resulting in one train of containment spray, RHR, and an ARS fan being rendered inoperable (Ref. 2).

## BASES

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### APPLICABLE SAFETY ANALYSES (continued)

The DBA analyses show that the maximum peak containment pressure of 11.44 psig results from the LOCA analysis, and is calculated to be less than the containment design pressure. The basis of the containment design temperature (327°F) is to ensure the OPERABILITY of safety related equipment inside containment (Ref. 3). The maximum peak containment atmosphere temperature of 325.6°F results from the SLB analysis. Therefore, the calculated peak containment atmosphere temperature is acceptable for the DBA SLB.

The modeled containment spray trains actuation from the containment analysis is based on a response time associated with exceeding the High—High containment pressure signal setpoint to achieving full flow through the containment spray train nozzles. A delayed response time initiation provides conservative analyses of peak calculated containment temperature and pressure responses. The containment spray trains total response time of 250 seconds is composed of signal delay, diesel generator startup, and system startup time to full flow through the containment spray train nozzles.

For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the ECCS cooling effectiveness during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. For these calculations, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the calculated transient containment pressures in accordance with 10 CFR 50, Appendix K (Ref. 4).

Inadvertent actuation of the Containment Spray System is evaluated in the analysis, and the resultant reduction in containment pressure is calculated. The maximum calculated reduction in containment pressure resulted in a containment external pressure load of 0.49 psid, which is below the containment design external pressure load.

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

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### LCO

During a DBA, one subsystem of Containment Spray System is required to provide the heat removal capability assumed in the safety analyses. To ensure that these requirements are met, two containment spray subsystems must be OPERABLE with power from two safety related, independent power supplies.

A containment spray subsystem shall be compromised of one containment spray train and one RHR spray train.

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BASES

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LCO (continued)

Each containment spray train includes a containment spray pump, header, valves, heat exchanger, nozzles, piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the RWST upon an ESF actuation signal and manually transferring suction to the containment sump.

Each RHR spray train includes an RHR pump, header, valves, heat exchanger, nozzles, piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the containment sump.

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APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment and an increase in containment pressure and temperature requiring the operation of the Containment Spray System. Furthermore, as stated in the Applicability Note, the RHR spray trains are not required to be OPERABLE in MODE 4.

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

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ACTIONS

A.1

With one containment spray subsystem inoperable, the affected subsystem must be restored to OPERABLE status within 72 hours. The components in this degraded condition are capable of providing 100% of the heat removal needs after an accident. The 72 hour Completion Time was developed taking into account the redundant heat removal afforded by the OPERABLE train and the low probability of a DBA occurring during this period.

B.1 and B.2

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

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BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.6.6.1

Verifying the correct alignment of manual, power operated, and automatic valves, excluding check valves, in the containment spray train provides assurance that the proper flow path exists for containment spray train operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position since they were verified in the correct position prior to being secured. This SR does not require any testing or valve manipulation. Rather, it involves verification that those valves outside containment and capable of potentially being mispositioned, are in the correct position.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.6.6.2

Verifying that each containment spray train pump's developed head at the flow test point is greater than or equal to the required developed head ensures that spray pump performance has not degraded during the cycle. Flow and differential head are normal tests of centrifugal pump performance required by the ASME Code (Ref. 5). Since the containment spray train pumps cannot be tested with flow through the spray headers, they are tested on bypass flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice inspections confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of this SR is in accordance with the Inservice Testing Program.

SR 3.6.6.3 and SR 3.6.6.4

These SRs require verification that each automatic containment spray train valve actuates to its correct position and each containment spray train pump starts upon receipt of an actual or simulated containment spray actuation signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

The surveillance of containment sump isolation valves is also required by SR 3.6.6.3. A single surveillance may be used to satisfy both requirements.

BASES

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.6.5

With the containment spray train inlet valves closed and the containment spray header drained of any solution, low pressure air or smoke can be blown through test connections. This SR ensures that each spray nozzle is unobstructed and that spray coverage of the containment during an accident is not degraded.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.6.6.6

Verifying the correct alignment of manual, power operated, and automatic valves, excluding check valves, in the RHR spray train provides assurance that the proper flow path exists for system operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position since they were verified in the correct position prior to locking, sealing, or securing. A valve is also allowed to be in the nonaccident position provided it can be aligned to the accident position within the time assumed in the accident analysis. This is acceptable since the RHR spray mode is manually initiated. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned, are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.6.6.7

Verifying that each RHR spray train pump's developed head at the flow test point is greater than or equal to the required developed head ensures that spray pump performance has not degraded during the cycle. Flow and differential head are normal tests of centrifugal pump performance required by the ASME Code (Ref. 5). Since the RHR spray train pumps cannot be tested with flow through the spray headers, they are tested on bypass flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice inspections confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of this SR is in accordance with the Inservice Testing Program.

BASES

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.6.8

With the RHR spray train inlet valves closed and the RHR spray header drained of any solution, low pressure air or smoke can be blown through test connections. This SR ensures that each RHR spray nozzle is unobstructed and that spray coverage of the containment during an accident is not degraded.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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REFERENCES

1. 10 CFR 50, Appendix A, GDC 38, GDC 39, and GDC 40.
  2. UFSAR, Section 6.2.
  3. 10 CFR 50.49.
  4. 10 CFR 50, Appendix K.
  5. ASME Code for Operation and Maintenance of Nuclear Power Plants.
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## B 3.6 CONTAINMENT SYSTEMS

### B 3.6.7 Shield Building

#### BASES

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##### BACKGROUND

The shield building is a concrete structure that surrounds the steel containment vessel. Between the containment vessel and the shield building inner wall is an annular space that collects containment leakage that may occur following a loss of coolant accident (LOCA). This space also allows for periodic inspection of the outer surface of the steel containment vessel.

The Emergency Gas Treatment System (EGTS) is a system consisting of two subsystems:

- a. annulus vacuum control subsystem, and
- b. air cleanup subsystem.

The annulus vacuum control subsystem is used during normal operation to establish and maintain a negative pressure in the annulus space. The annulus vacuum control subsystem does not perform any safety function.

The air cleanup subsystem operates during a LOCA to establish and maintain a negative annulus pressure of at least 0.5 inches water gauge. Filters in the subsystem then control the release of radioactive contaminants to the environment. The EGTS air cleanup subsystem OPERABILITY requirements are specified in LCO 3.6.10, "Emergency Gas Treatment System (EGTS) Air Cleanup Subsystem." The shield building is required to be OPERABLE to ensure retention of containment leakage and proper operation of the EGTS Air Cleanup Subsystem.

The isolation devices for the penetrations in the shield building boundary are a part of the shield building leak tight barrier. To maintain the shield building boundary leak tight, the sealing mechanism associated with each penetration (e.g., welds, bellows, or O-rings) are required to be OPERABLE. Access to the annulus area of the shield building is provided via the reactor building access room door and the water tight annulus access door located on 690 ft. elevation. During normal operation, these doors provide personnel and equipment access to the shield building annulus area and are equipped with electrical interlocks to assure that one door is always closed.

BASES

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APPLICABLE SAFETY ANALYSES      The design basis for shield building OPERABILITY is a LOCA. Maintaining shield building OPERABILITY ensures that the release of radioactive material from the containment atmosphere is restricted to those leakage paths and associated leakage rates assumed in the accident analyses.

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

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LCO      Shield building OPERABILITY must be maintained to ensure proper operation of the EGTS Air Cleanup Subsystem and to limit radioactive leakage from the containment to those paths and leakage rates assumed in the accident analyses.

The LCO is modified by a Note to allow the annulus access door to be opened to allow normal transit entry and exit. The basis of this exception is the assumption that, for normal transit, the time which the door is open will be short (i.e., shorter than the Completion Time for Condition A).

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APPLICABILITY      Maintaining shield building OPERABILITY prevents leakage of radioactive material from the shield building. Radioactive material may enter the shield building from the containment following a LOCA. Therefore, shield building OPERABILITY is required in MODES 1, 2, 3, and 4 when a steam line break, LOCA, or rod ejection accident could release radioactive material to the 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.

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ACTIONS      A.1

In the event shield building OPERABILITY is not maintained, shield building OPERABILITY must be restored within 1 hour. One hour is a reasonable Completion Time considering the limited leakage design of containment and the low probability of a Design Basis Accident occurring during this time period. This specified time period is also consistent with the ACTIONS of LCO 3.6.1, "Containment," which requires the containment be restored to OPERABLE status within 1 hour.

BASES

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ACTIONS (continued)

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.

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SURVEILLANCE  
REQUIREMENTS

SR 3.6.7.1

Maintaining shield building OPERABILITY requires verifying the door in the access opening closed. The access opening contains one door. The annulus access door is normally kept closed, except when the access opening is being used for entry and exit.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.6.7.2

A visual inspection of the accessible shield building interior and exterior surfaces and verification that no apparent changes in the concrete surface appearance or other abnormal degradation will 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.

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REFERENCES

None.

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## B 3.6 CONTAINMENT SYSTEMS

### B 3.6.8 Hydrogen Mitigation System (HMS)

#### BASES

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**BACKGROUND** The HMS reduces the potential for breach of primary containment due to a hydrogen oxygen reaction in post accident environments. The HMS is required by 10 CFR 50.44, "Standards for Combustible Gas Control Systems in Light-Water-Cooled Reactors" (Ref. 1), to reduce the hydrogen concentration in the primary containment following a degraded core accident. The HMS must be capable of handling an amount of hydrogen equivalent to that generated from a metal water reaction involving 75% of the fuel cladding surrounding the active fuel region (excluding the plenum volume).

10 CFR 50.44 (Ref. 1) requires units with ice condenser containments to install suitable hydrogen control systems that would accommodate an amount of hydrogen equivalent to that generated from the reaction of 75% of the fuel cladding with water. The HMS provides this required capability. This requirement was placed on ice condenser units because of their small containment volume and low design pressure (compared with pressurized water reactor dry containments). Calculations indicate that if hydrogen equivalent to that generated from the reaction of 75% of the fuel cladding with water were to collect in the primary containment, the resulting hydrogen concentration would be far above the lower flammability limit such that, if ignited from a random ignition source, the resulting hydrogen burn would seriously challenge the containment and safety systems in the containment.

The HMS is based on the concept of controlled ignition using thermal ignitors, designed to be capable of functioning in a post accident environment, seismically supported, and capable of actuation from the control room. A total of 68 ignitors are distributed throughout the various regions of containment in which hydrogen could be released or to which it could flow in significant quantities. The ignitors are arranged in two independent trains such that each containment region has at least two ignitors, one from each train, controlled and powered redundantly so that ignition would occur in each region even if one train failed to energize. Additional information regarding containment regions and the distribution of hydrogen ignitors within each region is contained in UFSAR Section 6.2.5A.

When the HMS is initiated, the ignitor elements are energized and heat up to a surface temperature  $\geq 1700^{\circ}\text{F}$ . At this temperature, they ignite the hydrogen gas that is present in the airspace in the vicinity of the ignitor. The HMS depends on the dispersed location of the ignitors so that local pockets of hydrogen at increased concentrations would burn before

BASES

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BACKGROUND (continued)

reaching a hydrogen concentration significantly higher than the lower flammability limit. Hydrogen ignition in the vicinity of the ignitors is assumed to occur when the local hydrogen concentration reaches 8.0 volume percent and results in 85% of the hydrogen present being consumed.

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APPLICABLE  
SAFETY  
ANALYSES

The HMS causes hydrogen in containment to burn in a controlled manner as it accumulates following a degraded core accident (Ref. 2). Burning occurs at the lower flammability concentration, where the resulting temperatures and pressures are relatively benign. Without the system, hydrogen could build up to higher concentrations that could result in a violent reaction if ignited by a random ignition source after such a buildup.

The hydrogen ignitors are not included for mitigation of a Design Basis Accident (DBA) because an amount of hydrogen equivalent to that generated from the reaction of 75% of the fuel cladding with water is far in excess of the hydrogen calculated for the limiting DBA loss of coolant accident (LOCA). The hydrogen ignitors have been shown by probabilistic risk analysis to be a significant contributor to limiting the severity of accident sequences that are commonly found to dominate risk for units with ice condenser containments. The Hydrogen Mitigation System satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

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LCO

Two HMS trains must be OPERABLE with power from two independent, safety related power supplies.

For this unit, an OPERABLE HMS train consists of 33 of 34 ignitors energized on the train.

Operation with at least one HMS train ensures that the hydrogen in containment can be burned in a controlled manner. Unavailability of both HMS trains could lead to hydrogen buildup to higher concentrations, which could result in a violent reaction if ignited. The reaction could take place fast enough to lead to high temperatures and overpressurization of containment and, as a result, breach containment or cause containment leakage rates above those assumed in the safety analyses. Damage to safety related equipment located in containment could also occur.

Each containment region must contain at least one OPERABLE hydrogen ignitor. This ensures that, assuming a single failure, there is still ignition capability in an adjacent region.

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APPLICABILITY

Requiring OPERABILITY in MODES 1 and 2 for the HMS ensures its immediate availability after safety injection and scram actuated on a LOCA initiation. In the post accident environment, the two HMS trains are required to control the hydrogen concentration within containment to near

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## BASES

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### APPLICABILITY (continued)

its flammability limit of 4.1 volume percent assuming a worst case single failure. This prevents overpressurization of containment and damage to safety related equipment and instruments located within containment.

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

In MODES 5 and 6, the probability and consequences of a LOCA are reduced due to the pressure and temperature limitations of these MODES. Therefore, the HMS is not required to be OPERABLE in MODES 5 and 6.

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### ACTIONS

#### A.1 and A.2

With one HMS train inoperable, the inoperable train must be restored to OPERABLE status within 7 days or the OPERABLE train must be verified OPERABLE frequently by performance of SR 3.6.8.1. The 7 day Completion Time is based on the low probability of the occurrence of a degraded core event that would generate hydrogen in amounts equivalent to a metal water reaction of 75% of the core cladding, the length of time after the event that operator action would be required to prevent hydrogen accumulation from exceeding this limit, and the low probability of failure of the OPERABLE HMS train. Alternative Required Action A.2, by frequent surveillances, provides assurance that the OPERABLE train continues to be OPERABLE.

#### B.1

Condition B is one containment region with no OPERABLE hydrogen ignitor. Thus, while in Condition B, or in Conditions A and B simultaneously, there would always be ignition capability in the adjacent containment regions that would provide redundant capability by flame propagation to the region with no OPERABLE ignitors.

Required Action B.1 calls for the restoration of one hydrogen ignitor in each region to OPERABLE status within 7 days. The 7 day Completion Time is based on the same reasons given under Required Action A.1.

BASES

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## ACTIONS (continued)

C.1

The unit must be placed in a MODE in which the LCO does not apply if the HMS subsystem(s) cannot be restored to OPERABLE status within the associated Completion Time. This is done by placing the unit in 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.

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SURVEILLANCE  
REQUIREMENTSSR 3.6.8.1

This SR confirms that  $\geq 33$  of 34 hydrogen ignitors can be successfully energized in each train. The ignitors are simple resistance elements. Therefore, energizing provides assurance of OPERABILITY. The allowance of one inoperable hydrogen ignitor is acceptable because, although one inoperable hydrogen ignitor in a region would compromise redundancy in that region, the containment regions are interconnected so that ignition in one region would cause burning to progress to the others (i.e., there is overlap in each hydrogen ignitor's effectiveness between regions).

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.6.8.2

This SR confirms that the two inoperable hydrogen ignitors allowed by SR 3.6.8.1 (i.e., one in each train) are not in the same containment region which ensures that each containment region contains at least one OPERABLE hydrogen ignitor.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.6.8.3

A more detailed functional test is performed to verify system OPERABILITY. All ignitors (glow plugs), including normally inaccessible ignitors, are visually checked for a glow to verify that they are energized. Additionally, the surface temperature of each glow plug is measured to be  $\geq 1700^{\circ}\text{F}$  to demonstrate that a temperature sufficient for ignition is achieved.

BASES

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SURVEILLANCE REQUIREMENTS (continued)

The Surveillance Frequency is controlled under the Surveillance  
Frequency Control Program

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REFERENCES

1. 10 CFR 50.44.
  2. UFSAR, Section 6.2.
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## B 3.6 CONTAINMENT SYSTEMS

### B 3.6.9 Vacuum Relief Valves

#### BASES

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**BACKGROUND** The purpose of the vacuum relief lines is to 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 containment cooling features, such as the Containment Spray System, the Air Return System, or both. Multiple equipment failures or human errors are necessary to cause inadvertent actuation of these systems.

The containment pressure vessel contains three vacuum relief lines that protect the containment from excessive external loading.

The vacuum relief system has three identical lines located on the dome, at the same elevation, and 120° apart. Each line contains a containment vessel vacuum relief valve in series with a containment vessel vacuum relief isolation valve, the vacuum relief valve being outside of the isolation valve. The lines are installed such that there is sufficient space between the vacuum relief system and the Shield Building to prevent contact during seismic or pressure transient motion and to allow for an adequate airflow path.

Each containment vessel vacuum relief valve is a 24 inch, self-actuated, horizontally installed, swing-disc valve, with an elastomer seat. The seat material will withstand post-LOCA temperature, pressure, and radiation conditions. Each line has a design airflow rate of 28 pounds per second at a pressure differential of 0.5 psid across the entire line. Each normally closed vacuum relief valve is equipped with limit switches so that open and closed positions of the valve are indicated in the main control room. The opening of any of these valves is indicated in the main control room. The valves begin opening at a containment external pressure differential of 0.1 psid and will be fully open in 2.2 seconds for a vacuum relief system design basis event.

Each containment vessel vacuum relief isolation valve is a pneumatically operated butterfly valve with an elastomer seat. The valve, including seat material, will withstand post-LOCA temperature, pressure, and radiation conditions. Two separate trains of control air supplies are available to the two independent solenoid valves which power the isolation valve. The isolation valve, which is normally open, fails open, and will close when containment high pressure reaches the set pressure of 1.5 psid. The high pressure signal is developed from either of two independent sets of three pressure sensors and is completely independent of other containment isolation signals for other systems. Each isolation valve is equipped with

BASES

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BACKGROUND (continued)

a limit switch so that open and closed positions are indicated in the main control room.

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APPLICABLE  
SAFETY  
ANALYSES

Design of the vacuum relief lines involves calculating the effect of inadvertent actuation of containment cooling features, which can reduce the atmospheric temperature (and hence pressure) inside containment (Ref. 1). Conservative assumptions are used for all the relevant parameters in the calculation; for example, for the Containment Spray System, the minimum spray water temperature, maximum initial containment temperature, maximum spray flow, all spray trains operating, etc. The resulting containment pressure versus time is calculated, including the effect of the opening of the vacuum relief lines when their negative pressure setpoint is reached. It is also assumed that one valve fails to open.

The containment was designed for an external pressure load equivalent to 0.5 psig. The inadvertent actuation of the containment cooling features was analyzed to determine the resulting reduction in containment pressure. The initial pressure condition used in this analysis was 0.1 psi less than annulus pressure. This resulted in a minimum pressure inside containment of 0.49 psi less than annulus pressure, which is less than the design load.

The vacuum relief valves must also perform the containment isolation function in a containment high pressure event. For this reason, the system is designed to take the full containment positive design pressure and the environmental conditions (temperature, pressure, humidity, radiation, chemical attack, etc.) associated with the containment DBA.

The vacuum relief valves satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

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LCO

The LCO establishes the minimum equipment required to accomplish the vacuum relief function following the inadvertent actuation of containment cooling features. Three vacuum relief lines are required to be OPERABLE to ensure that at least two are available, assuming one vacuum relief valve fails to open.

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APPLICABILITY

In MODES 1, 2, 3, and 4, the containment cooling features, such as the Containment Spray System and the Air Return System, are required to be OPERABLE to mitigate the effects of a DBA. Excessive negative pressure inside containment could occur 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, Air Return System, or both.

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BASES

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APPLICABILITY (continued)

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 Air Return System are not required to be OPERABLE in MODES 5 and 6. Therefore, maintaining OPERABLE vacuum relief valves is not required in MODE 5 or 6.

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ACTIONS

A.1

When one of the required vacuum relief lines is 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.

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SURVEILLANCE  
REQUIREMENTS

SR 3.6.9.1

This SR cites 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 the ASME Code (Ref. 2). Therefore, SR Frequency is governed by the Inservice Testing Program.

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REFERENCES

1. UFSAR, Section 6.2.
  2. ASME Code for Operation and Maintenance of Nuclear Power Plants.
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## B 3.6 CONTAINMENT SYSTEMS

### B 3.6.10 Emergency Gas Treatment System (EGTS) Air Cleanup Subsystem

#### BASES

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**BACKGROUND** The EGTS Air Cleanup Subsystem is required by 10 CFR 50, Appendix A, GDC 41, "Containment Atmosphere Cleanup" (Ref. 1), to ensure that radioactive materials that leak from the primary containment into the shield building (secondary containment) following a Design Basis Accident (DBA) are filtered and adsorbed prior to exhausting to the environment.

The containment has a secondary containment called the shield building, which is a concrete structure that surrounds the steel primary containment vessel. Between the containment vessel and the shield building inner wall is an annular space that collects any containment leakage that may occur following a loss of coolant accident (LOCA). This space also allows for periodic inspection of the outer surface of the steel containment vessel.

The EGTS design consists of two subsystems. The annulus vacuum control subsystem is used to establish and maintain a negative pressure within the secondary containment annulus during normal plant operation. The annulus vacuum control subsystem does not perform any safety function. The air cleanup subsystem is actuated following a LOCA to maintain a negative pressure in the annulus area between the shield building and the steel containment. Filters in the air cleanup subsystem then control the release of radioactive contaminants to the environment. The air cleanup subsystem is the portion of EGTS that performs a safety function and is required to be OPERABLE. OPERABILITY requirements associated with the shield building are specified in LCO 3.6.7, "Shield Building." Shield building OPERABILITY is required to ensure retention of primary containment leakage and proper operation of the EGTS Air Cleanup Subsystem.

The EGTS Air Cleanup Subsystem consists of two separate and redundant trains. Each train includes a heater, a prefilter, moisture separators, 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

BASES

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BACKGROUND (continued)

Phase A containment isolation signal. The system is described in Reference 2.

The prefilters remove large particles in the air, and the moisture separators remove entrained water droplets present, to prevent excessive loading of the HEPA filters and charcoal absorbers. Heaters may be included to reduce the relative humidity of the airstream on systems that operate in high humidity.

The EGTS Air Cleanup Subsystem reduces the radioactive content in the shield building atmosphere following a DBA. Loss of the EGTS Air Cleanup Subsystem could cause site boundary doses, in the event of a DBA, to exceed the values given in licensing basis.

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APPLICABLE  
SAFETY  
ANALYSES

The EGTS Air Cleanup Subsystem 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 EGTS Air Cleanup Subsystem 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 EGTS Air Cleanup Subsystem actuation in the safety analyses is based upon a worst case response time following a Phase A containment isolation initiated at the limiting setpoint. The total response time, from exceeding the signal setpoint to attaining the negative pressure of 0.5 inch water gauge in the shield building, is 60 seconds. This response time is composed of signal delay, diesel generator startup and sequencing time, system startup time, and time for the system to attain the required pressure after starting.

The EGTS Air Cleanup Subsystem satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

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LCO

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

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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

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BASES

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APPLICABILITY (continued)

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 (although one or more trains may be operating for other reasons, such as habitability during maintenance in the shield building annulus).

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ACTIONS

A.1

With one EGTS Air Cleanup Subsystem 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 EGTS Air Cleanup Subsystem train and the low probability of a DBA occurring during this period. The Completion Time is adequate to make most repairs.

B.1 and B.2

If the EGTS Air Cleanup Subsystem 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.

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SURVEILLANCE  
REQUIREMENTS

SR 3.6.10.1

Operating each EGTS Air Cleanup Subsystem train from the Control Room with flow through the HEPA filters and charcoal adsorbers 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 Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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BASES

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.10.2

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

SR 3.6.10.3

The automatic startup ensures that each EGTS Air Cleanup Subsystem train responds properly.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.6.10.4

The EGTS Air Cleanup Subsystem filter cooling bypass valves are tested to verify OPERABILITY. The ability to cool the filters and adsorbers in an inactive air cleanup unit is accomplished with two crossover flow ducts that draw a small stream of air from the active air cleanup unit through the inactive air cleanup unit. The valves in the inactive train automatically receive a signal to open. The capability to manually open the suction valve for the inactive train and align to the affected unit is provided in the main control room to complete the flow path through the inactive unit.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.6.10.5

The proper functioning of the fans, dampers, filters, adsorbers, etc., as a system is verified by the ability of each train to produce the required system flow rate.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.10.6

The EGTS Air Cleanup Subsystem produces a negative pressure to prevent leakage from the shield building. This Surveillance verifies that the shield building can be rapidly drawn down to - 0.5 inch water gauge in the annulus. This test is used to ensure shield building boundary integrity. Establishment of this pressure is confirmed by this SR, which demonstrates that the shield building can be drawn down to a negative pressure of  $\geq 0.5$  inches of water gauge in the annulus in  $\leq 60$  seconds using one EGTS Air Cleanup Subsystem train. The time limit ensures that no significant quantity of radioactive material leaks from the shield building prior to developing the negative pressure. Since this Surveillance is a shield building boundary integrity test, it does not need to be performed with each EGTS Air Cleanup Subsystem train; thus, this Surveillance is performed on a STAGGERED TEST BASIS. The primary purpose of this SR is to ensure shield building integrity. The secondary purpose of this SR is to ensure that the EGTS Air Cleanup Subsystem train being tested functions as designed. Upon failure to meet this SR, the leak tightness of the shield building must be immediately assessed to determine the impact on the OPERABILITY of the shield building. If a negative pressure of  $\geq 0.5$  inch water gauge cannot be maintained in the annulus by either EGTS Air Cleanup Subsystem train (i.e., loss of shield building safety function), the shield building must be declared inoperable and ACTIONS of LCO 3.6.7 performed in accordance with LCO 3.0.6 and Specification 5.5.13, "Safety Function Determination Program (SFDP)."

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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- REFERENCES
1. 10 CFR 50, Appendix A, GDC 41.
  2. UFSAR, Section 6.2.
  3. UFSAR, Chapter 15.
  4. Regulatory Guide 1.52, Revision 2.
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## B 3.6 CONTAINMENT SYSTEMS

### B 3.6.11 Air Return System (ARS)

#### BASES

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**BACKGROUND** The ARS is designed to assure the rapid return of air from the upper to the lower containment compartment after the initial blowdown following a Design Basis Accident (DBA). The return of this air to the lower compartment and subsequent recirculation back up through the ice condenser assists in cooling the containment atmosphere and limiting post accident pressure and temperature in containment to less than design values. Limiting pressure and temperature reduces the release of fission product radioactivity from containment to the environment in the event of a DBA.

The ARS provides post accident hydrogen mixing in selected areas of containment. Hydrogen collection headers are routed to potential hydrogen pockets in containment, terminating on the suction side of either of the two ARS fans at the header isolation valves. The minimum design flow from each potential hydrogen pocket is sufficient to limit the local concentration of hydrogen.

The ARS consists of two separate trains of equal capacity, each capable of meeting the design bases. Each train includes a 100% capacity air return fan, associated damper, and hydrogen collection headers with isolation valves. Each train is powered from a separate Engineered Safety Features (ESF) bus.

The ARS fans are automatically started by the Phase B containment isolation signal approximately 10 minutes after the containment pressure reaches the pressure setpoint. The fan backdraft dampers ensure that no energy released during the initial phase of a DBA will bypass the ice bed through the ARS fans.

After starting, the fans displace air from the upper compartment to the lower compartment, thereby returning the air that was displaced by the high energy line break blowdown from the lower compartment. After discharge into the lower compartment, air flows with steam produced by residual heat through the ice condenser doors into the ice condenser compartment where the steam portion of the flow is condensed. The air flow returns to the upper compartment through the top deck doors in the upper portion of the ice condenser compartment. The ARS fans operate continuously after actuation, circulating air through the containment volume and purging all potential hydrogen pockets in containment.

The ARS also functions, after all the ice has melted, to circulate any steam still entering the lower compartment to the upper compartment where the Containment Spray System can cool it.

BASES

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BACKGROUND (continued)

The ARS is an ESF system. It is designed to ensure that the heat removal capability required during the post accident period can be attained. The operation of the ARS, in conjunction with the ice bed, the Containment Spray System, and the Residual Heat Removal (RHR) System spray, provides the required heat removal capability to limit post accident conditions to less than the containment design values.

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APPLICABLE  
SAFETY  
ANALYSES

The limiting DBAs considered relative to containment temperature and pressure are the loss of coolant accident (LOCA) and the steam line break (SLB). The LOCA and SLB are analyzed using computer codes designed to predict the resultant containment pressure and temperature transients. DBAs are assumed not to occur simultaneously or consecutively. The postulated DBAs are analyzed, in regard to ESF systems, assuming the loss of one ESF bus, which is the worst case single active failure and results in one train each of the Containment Spray System, RHR System, and ARS being inoperable (Ref. 1). The DBA analyses show that the maximum peak containment pressure results from the LOCA analysis and is calculated to be less than the containment design pressure.

For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the cooling effectiveness of the Emergency Core Cooling System during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. For these calculations, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the calculated transient containment pressures, in accordance with 10 CFR 50, Appendix K (Ref. 2).

The analysis for minimum internal containment pressure (i.e., maximum external differential containment pressure) assumes inadvertent simultaneous actuation of both the ARS and the Containment Spray System. The containment vacuum relief valves are designed to accommodate inadvertent actuation of either or both systems.

The modeled ARS actuation from the containment analysis is based upon a response time associated with exceeding the containment pressure High-High signal setpoint to achieving full ARS air flow. A delayed response time initiation ensures that no energy released during the initial phase of a DBA will bypass the ice bed through the ARS fans. The ARS total response time of 600 seconds consists of the built in signal delay.

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

BASES

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LCO In the event of a DBA, one train of the ARS is required to provide the minimum air recirculation for heat removal and hydrogen mixing assumed in the safety analyses. To ensure this requirement is met, two trains of the ARS must be OPERABLE. This will ensure that at least one train will operate, assuming the worst case single failure occurs, which is in the loss of ESF power supply.

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APPLICABILITY In MODES 1, 2, 3, and 4, a DBA could cause an increase in containment pressure and temperature requiring the operation of the ARS. Therefore, the LCO is applicable in MODES 1, 2, 3, and 4.

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

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ACTIONS A.1

If one of the required trains of the ARS is inoperable, it must be restored to OPERABLE status within 72 hours. The components in this degraded condition are capable of providing 100% of the flow needs after an accident. The 72 hour Completion Time was developed taking into account the redundant flow capability of the OPERABLE ARS train and the low probability of a DBA occurring in this period.

B.1 and B.2

If the ARS 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.

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SURVEILLANCE REQUIREMENTS SR 3.6.11.1

Verifying that each ARS fan starts on an actual or simulated actuation signal, after a delay  $\geq 9.0$  minutes and  $\leq 11.0$  minutes, and operates for  $\geq 15$  minutes is sufficient to ensure that the fans are OPERABLE and that the associated controls and time delays are functioning properly. It also ensures that blockage, fan and/or motor failure, or excessive vibration can be detected for corrective action.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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BASES

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.11.2

Verifying ARS fan motor current with the return air dampers closed confirms one operating condition of the fan. This test is indicative of overall fan motor performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.6.11.3

Verifying the OPERABILITY of the return air damper provides assurance that the proper flow path will exist when the fan is started. By applying the correct counterweight, the damper operation can be confirmed.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program

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REFERENCES

1. UFSAR, Section 6.2.
  2. 10 CFR 50, Appendix K.
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## B 3.6 CONTAINMENT SYSTEMS

### B 3.6.12 Ice Bed

#### BASES

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**BACKGROUND** The ice bed consists of a minimum of 1,916,000 lb of ice stored within the ice condenser. The primary purpose of the ice bed is to provide a large heat sink in the event of a release of energy from a Design Basis Accident (DBA) in containment. The ice would absorb energy and limit containment peak pressure and temperature during the accident transient. Limiting the pressure and temperature reduces the release of fission product radioactivity from containment to the environment in the event of a DBA.

The ice condenser is an annular compartment enclosing approximately 300 degrees of the perimeter of the upper containment compartment, but penetrating the operating deck so that a portion extends into the lower containment compartment. The lower portion has a series of hinged doors exposed to the atmosphere of the lower containment compartment, which, for normal unit operation, are designed to remain closed. At the top of the ice condenser is another set of doors exposed to the atmosphere of the upper compartment, which also remain closed during normal unit operation. Intermediate deck doors, located below the top deck doors, form the floor of a plenum at the upper part of the ice condenser. These doors also remain closed during normal unit operation. The upper plenum area is used to facilitate surveillance and maintenance of the ice bed.

1944 ice baskets contain the ice within the ice condenser. The ice bed is considered to consist of the total volume from the bottom elevation of the ice baskets to the top elevation of the ice baskets. The ice baskets position the ice within the ice bed in an arrangement to promote heat transfer from steam to ice. This arrangement enhances the ice condenser's primary function of condensing steam and absorbing heat energy released to the containment during a DBA.

In the event of a DBA, the ice condenser inlet doors (located below the operating deck) open due to the pressure rise in the lower compartment. This allows air and steam to flow from the lower compartment into the ice condenser. The resulting pressure increase within the ice condenser causes the intermediate deck doors and the top deck doors to open, which allows the air to flow out of the ice condenser into the upper compartment. Steam condensation within the ice condenser limits the pressure and temperature buildup in containment. A divider barrier (i.e., operating deck and extensions thereof) separates the upper and lower compartments and ensures that the steam is directed into the ice condenser.

## BASES

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### BACKGROUND (continued)

The ice, together with the containment spray, is adequate to absorb the initial blowdown of steam and water from a DBA and the additional heat loads that would enter containment during several hours following the initial blowdown. The additional heat loads would come from the residual heat in the reactor core, the hot piping and components, and the secondary system, including the steam generators. During the post blowdown period, the Air Return System (ARS) returns upper compartment air through the divider barrier to the lower compartment. This serves to continue circulating heated air and steam from the lower compartment through the ice condenser where the heat is removed by the remaining ice.

As ice melts, the water passes through the ice condenser floor drains into the lower compartment. Thus, a second function of the ice bed is to be a large source of borated water (via the containment sump) for long term Emergency Core Cooling System (ECCS) and Containment Spray System heat removal functions in the recirculation mode.

A third function of the ice bed and melted ice is to remove fission product iodine that may be released from the core during a DBA. Iodine removal occurs during the ice melt phase of the accident and continues as the melted ice is sprayed into the containment atmosphere by the Containment Spray System. The ice is adjusted to an alkaline pH that facilitates removal of radioactive iodine from the containment atmosphere.

It is important for ice to exist in the ice baskets, the ice to be appropriately distributed around the 24 ice condenser bays, and for open flow paths to exist around ice baskets. This is especially important during the initial blowdown so that the steam and water mixture entering the lower compartment do not pass through only part of the ice condenser, depleting the ice there while bypassing the ice in other bays.

Two phenomena that can degrade the ice bed during the long service period are:

- a. Loss of ice by melting or sublimation; and
- b. Obstruction of flow passages through the ice bed due to buildup of ice.

## BASES

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### BACKGROUND (continued)

Both of these degrading phenomena are reduced by minimizing air leakage into and out of the ice condenser.

The ice bed limits the temperature and pressure that could be expected following a DBA, thus limiting leakage of fission product radioactivity from containment to the environment.

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### APPLICABLE SAFETY ANALYSES

The limiting DBAs considered relative to containment temperature and pressure are the loss of coolant accident (LOCA) and the steam line break (SLB). The LOCA and SLB are analyzed using computer codes designed to predict the resultant containment pressure and temperature transients. DBAs are not assumed to occur simultaneously or consecutively.

Although the ice condenser is a passive system that requires no electrical power to perform its function, the Containment Spray System and the ARS also function to assist the ice bed in limiting pressures and temperatures. Therefore, the postulated DBAs are analyzed in regards to containment Engineered Safety Feature (ESF) systems, assuming the loss of one ESF bus, which is the worst case single active failure and results in one train of the Containment Spray System and one ARS fan being inoperable.

The limiting DBA analyses (Ref. 1) show that the maximum peak containment pressure results from the LOCA analysis and is calculated to be less than the containment design pressure. For certain aspects of the transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the cooling effectiveness of the ECCS during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. For these calculations, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the calculated transient containment pressures, in accordance with 10 CFR 50, Appendix K (Ref. 2).

The maximum peak containment atmosphere temperature results from the SLB analysis and is discussed in the Bases for LCO 3.6.5, "Containment Air Temperature."

In addition to calculating the overall peak containment pressures, the DBA analyses include calculation of the transient differential pressures that occur across subcompartment walls during the initial blowdown phase of the accident transient. The internal containment walls and structures are designed to withstand these local transient pressure differentials for the limiting DBAs.

BASES

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APPLICABLE SAFETY ANALYSES (continued)

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

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LCO The ice bed LCO requires the existence of the required quantity of stored ice, appropriate distribution of the ice within the ice bed, open flow paths through the ice bed, and appropriate chemical content and pH of the stored ice. The stored ice functions to absorb heat during the blowdown phase and long term phase of a DBA, thereby limiting containment air temperature and pressure. The chemical content and pH of the stored ice provide core SDM (boron content) and remove radioactive iodine from the containment atmosphere when the melted ice is recirculated through the ECCS and the Containment Spray System, respectively.

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APPLICABILITY In MODES 1, 2, 3, and 4, a DBA could cause an increase in containment pressure and temperature requiring the operation of the ice bed. Therefore, the LCO is applicable in MODES 1, 2, 3, and 4.

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the ice bed is not required to be OPERABLE in these MODES.

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ACTIONS A.1

If the ice bed is inoperable, it must be restored to OPERABLE status within 48 hours. The Completion Time was developed based on operating experience, which confirms that due to the very large mass of stored ice, the parameters comprising OPERABILITY do not change appreciably in this time period. If a degraded condition is identified, even for temperature, with such a large mass of ice it is not possible for the degraded condition to significantly degrade further in a 48 hour period. Therefore, 48 hours is a reasonable amount of time to correct a degraded condition before initiating a shutdown.

B.1 and B.2

If the ice bed 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.

BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.6.12.1

Verifying that the maximum temperature of the ice bed is  $\leq 27^{\circ}\text{F}$  ensures that the ice is kept well below the melting point. The 12 hour Frequency was based on operating experience, which confirmed that, due to the large mass of stored ice, it is not possible for the ice bed temperature to degrade significantly within a 12 hour period and was also based on assessing the proximity of the LCO limit to the melting temperature.

Furthermore, the 12 hour Frequency is considered adequate in view of indications in the control room, including the alarm, to alert the operator to an abnormal ice bed temperature condition. This SR may be satisfied by use of the Ice Bed Temperature Monitoring System.

SR 3.6.12.2

The minimum weight figure of 1145 pounds of ice per basket contains a 15% conservative allowance for ice loss through sublimation which is a factor of 15 higher than assumed for the ice condenser design. The minimum weight figure of 2,225,880 pounds of ice also contains an additional 1% conservative allowance to account for systematic error in weighing instruments.

The Frequency of 18 months was based on ice storage tests and the allowance built into the required ice mass over and above the mass assumed in the safety analyses. Operating experience has verified that, with the 18-month Frequency, the weight requirements are maintained with no significant degradation between surveillances.

SR 3.6.12.3

This SR ensures that the flow channels through the ice bed have not accumulated ice blockage that exceeds 15 percent of the total flow area through the ice bed region. The allowable 15 percent buildup of ice is based on the analysis of the sub-compartment response to a design basis LOCA with partial blockage of the ice condenser flow channels. The analysis did not perform detailed flow area modeling, but lumped the ice condenser bays into six sections ranging from 2.75 bays to 6.5 bays. Individual bays are acceptable with greater than 15 percent blockage, as long as 15 percent blockage is not exceeded for any analysis section.

To provide a 95 percent confidence that flow blockage does not exceed the allowed 15 percent, the visual inspection must be made for at least 54 (33 percent) of the 162 flow channels per ice condenser bay. The visual inspection of the ice bed flow channels is to inspect the flow area, by looking down from the top of the ice bed, and where view is achievable up

## BASES

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### SURVEILLANCE REQUIREMENTS (continued)

from the bottom of the ice bed. Flow channels to be inspected are determined by random sample. As the most restrictive ice bed flow passage is found at a lattice frame elevation, the 15 percent blockage criteria only applies to "flow channels" that comprise the area:

- a. between ice baskets, and
- b. past lattice frames and wall panels.

Due to a significantly larger flow area in the regions of the upper deck grating and the lower inlet plenum support structures and turning vanes, a gross buildup of ice on these structures would be required to degrade air and steam flow. Therefore, these structures are excluded as part of a flow channel for application of the 15 percent blockage criteria. Industry experience has shown that removal of ice from the excluded structures during the refueling outage is sufficient to ensure they remain OPERABLE throughout the operating cycle. Removal of any gross ice buildup on the excluded structures is performed following outage maintenance activities.

Operating experience has demonstrated that the ice bed is the region that is the most flow restrictive, due to the normal presence of ice accumulation on lattice frames and wall panels. The flow area through the ice basket support platform is not a more restrictive flow area because it is easily accessible from the lower plenum and is maintained clear of ice accumulation. There is no mechanistically credible method for ice to accumulate on the ice basket support platform during plant operation. Plant and industry experience has shown that the vertical flow area through the ice basket support platform remains clear of ice accumulation that could produce blockage. Normally only a glaze may develop or exist on the ice basket support platform which is not significant to blockage of flow area. Additionally, outage maintenance practices provide measures to clear the ice basket support platform following maintenance activities of any accumulation of ice that could block flow areas.

Frost buildup or loose ice is not to be considered as flow channel blockage, whereas attached ice is considered blockage of a flow channel. Frost is the solid form of water that is loosely adherent, and can be brushed off with the open hand.

#### SR 3.6.12.4

Verifying the chemical composition of the stored ice ensures that the stored ice has a boron concentration  $\geq 1800$  ppm and  $\leq 2500$  ppm as sodium tetraborate and a high pH,  $\geq 9.0$  and  $\leq 9.5$ , in order to meet the requirement for borated water when the melted ice is used in the ECCS

BASES

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## SURVEILLANCE REQUIREMENTS (continued)

recirculation mode of operation. Additionally, the minimum boron concentration value is used to assure reactor subcriticality in a post LOCA environment, while the maximum boron concentration is used as the bounding value in the hot leg switchover timing calculation (Ref. 3). This is accomplished by obtaining at least 24 ice samples. Each sample is taken approximately one foot from the top of the ice of each randomly selected ice basket in each ice condenser bay. The SR is modified by a Note that allows the boron concentration and pH value obtained from averaging the individual samples' analysis results to satisfy the requirements of the SR. If either the average boron concentration or average pH value is outside their prescribed limit, then entry into Condition A is required. Sodium tetraborate has been proven effective in maintaining the boron content for long storage periods, and it also enhances the ability of the solution to remove and retain fission product iodine. The high pH is required to enhance the effectiveness of the ice and the melted ice in removing iodine from the containment atmosphere. This pH range also minimizes the occurrence of chloride and caustic stress corrosion on mechanical systems and components exposed to ECCS and Containment Spray System fluids in the recirculation mode of operation.

The Frequency of 54 months is intended to be consistent with the expected length of three fuel cycles, and was developed considering these facts:

- a. Long term ice storage tests have determined that the chemical composition of the stored ice is extremely stable,
- b. There are no normal operating mechanisms that decrease the boron concentration of the stored ice, and pH remains within a 9.0-9.5 range when boron concentrations are above approximately 1200 ppm.
- c. Operating experience has demonstrated that meeting the boron concentration and pH requirements has never been a problem, and
- d. Someone would have to enter the containment to take the sample, and, if the unit is at power, that person would receive a radiation dose.

SR 3.6.12.5

This SR ensures that a representative sampling of ice baskets, which are relatively thin walled, perforated cylinders, have not been degraded by wear, cracks, corrosion, or other damage. The Frequency of 40 months for a visual inspection of the structural soundness of the ice baskets is based on engineering judgment and considers such factors as the

## BASES

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### SURVEILLANCE REQUIREMENTS (continued)

thickness of the basket walls relative to corrosion rates expected in their services environment and the results of the long term ice storage testing.

#### SR 3.6.12.6

This SR ensures that initial ice fill and any subsequent ice additions meet the boron concentration and pH requirements of SR 3.6.12.4. The SR is modified by a Note that allows the chemical analysis to be performed on either the liquid or resulting ice of each sodium tetraborate solution prepared. If ice is obtained from offsite sources, then chemical analysis data must be obtained for the ice supplied.

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### REFERENCES

1. UFSAR, Section 6.2.
  2. 10 CFR 50, Appendix K.
  3. Sequoyah Nuclear Plant Units 1 and 2 – Nuclear Steam Supply System Engineering Support Services – Contract 99NAN-251787 – Letter N9873, Contract Work Authorization N20000 020 – Tritium Production Core – Post Loss of Coolant Accident (LOCA) Long Term Core Cooling Analysis – N2N 058, dated August 13, 2001.
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## B 3.6 CONTAINMENT SYSTEMS

### B 3.6.13 Ice Condenser Doors

#### BASES

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- BACKGROUND** The ice condenser doors consist of the inlet doors, the intermediate deck doors, and the top deck doors. The functions of the doors are to:
- a. Seal the ice condenser from air leakage during the lifetime of the unit; and
  - b. Open in the event of a Design Basis Accident (DBA) to direct the hot steam air mixture from the DBA into the ice bed, where the ice would absorb energy and limit containment peak pressure and temperature during the accident transient.

Limiting the pressure and temperature following a DBA reduces the release of fission product radioactivity from containment to the environment.

The ice condenser is an annular compartment enclosing approximately 300 degrees of the perimeter of the upper containment compartment, but penetrating the operating deck so that a portion extends into the lower containment compartment. The inlet doors separate the atmosphere of the lower compartment from the ice bed inside the ice condenser. The top deck doors are above the ice bed and are exposed to the atmosphere of the upper compartment. The intermediate deck doors, located below the top deck doors, form the floor of a plenum at the upper part of the ice condenser. This plenum area is used to facilitate surveillance and maintenance of the ice bed.

The ice baskets held in the ice bed within the ice condenser are arranged to promote heat transfer from steam to ice. This arrangement enhances the ice condenser's primary function of condensing steam and absorbing heat energy released to the containment during a DBA.

In the event of a DBA, the ice condenser inlet doors (located below the operating deck) open due to the pressure rise in the lower compartment. This allows air and steam to flow from the lower compartment into the ice condenser. The resulting pressure increase within the ice condenser causes the intermediate deck doors and the top deck doors to open, which allows the air to flow out of the ice condenser into the upper compartment. Steam condensation within the ice condensers limits the pressure and temperature buildup in containment. A divider barrier separates the upper and lower compartments and ensures that the steam is directed into the ice condenser.

## BASES

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### BACKGROUND (continued)

The ice, together with the containment spray, serves as a containment heat removal system and is adequate to absorb the initial blowdown of steam and water from a DBA as well as the additional heat loads that would enter containment during the several hours following the initial blowdown. The additional heat loads would come from the residual heat in the reactor core, the hot piping and components, and the secondary system, including the steam generators. During the post blowdown period, the Air Return System (ARS) returns upper compartment air through the divider barrier to the lower compartment. This serves to continue circulating heated air and steam from the lower compartment through the ice condenser, where the heat is removed by the remaining ice.

The water from the melted ice drains into the lower compartment where it serves as a source of borated water (via the containment sump) for the Emergency Core Cooling System (ECCS) and the Containment Spray System heat removal functions in the recirculation mode. The ice (via the Containment Spray System) and the recirculated ice melt also serve to clean up the containment atmosphere.

The ice condenser doors ensure that the ice stored in the ice bed is preserved during normal operation (doors closed) and that the ice condenser functions as designed if called upon to act as a passive heat sink following a DBA.

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### APPLICABLE SAFETY ANALYSES

The limiting DBAs considered relative to containment pressure and temperature are the loss of coolant accident (LOCA) and the steam line break (SLB). The LOCA and SLB are analyzed using computer codes designed to predict the resultant containment pressure and temperature transients. DBAs are assumed not to occur simultaneously or consecutively.

Although the ice condenser is a passive system that requires no electrical power to perform its function, the Containment Spray System and ARS also function to assist the ice bed in limiting pressures and temperatures. Therefore, the postulated DBAs are analyzed with respect to Engineered Safety Feature (ESF) systems, assuming the loss of one ESF bus, which is the worst case single active failure and results in one train of the Containment Spray System and one ARS fan being rendered inoperable.

The limiting DBA analyses (Ref. 1) show that the maximum peak containment pressure results from the LOCA analysis and is calculated to be less than the containment design pressure. For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the cooling effectiveness of

## BASES

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### APPLICABLE SAFETY ANALYSES (continued)

the ECCS during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. For these calculations, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the calculated transient containment pressures, in accordance with 10 CFR 50, Appendix K (Ref. 2).

The maximum peak containment atmosphere temperature results from the SLB analysis and is discussed in the Bases for LCO 3.6.5, "Containment Air Temperature."

An additional design requirement was imposed on the ice condenser door design for a small break accident in which the flow of heated air and steam is not sufficient to fully open the doors.

For this situation, the doors are designed so that all of the doors would partially open by approximately the same amount. Thus, the partially opened doors would modulate the flow so that each ice bay would receive an approximately equal fraction of the total flow.

This design feature ensures that the heated air and steam will not flow preferentially to some ice bays and deplete the ice there without utilizing the ice in the other bays.

In addition to calculating the overall peak containment pressures, the DBA analyses include the calculation of the transient differential pressures that would occur across subcompartment walls during the initial blowdown phase of the accident transient. The internal containment walls and structures are designed to withstand the local transient pressure differentials for the limiting DBAs.

The ice condenser doors satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

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### LCO

This LCO establishes the minimum equipment requirements to assure that the ice condenser doors perform their safety function. The ice condenser inlet doors, intermediate deck doors, and top deck doors must be closed to minimize air leakage into and out of the ice condenser, with its attendant leakage of heat into the ice condenser and loss of ice through melting and sublimation. The doors must be OPERABLE to ensure the proper opening of the ice condenser in the event of a DBA. OPERABILITY includes being free of any obstructions that would limit their opening, and for the inlet doors, being adjusted such that the opening and closing torques are within limits. The ice condenser doors function with the ice condenser to limit the pressure and temperature that could be expected following a DBA.

## BASES

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**APPLICABILITY** In MODES 1, 2, 3, and 4, a DBA could cause an increase in containment pressure and temperature requiring the operation of the ice condenser doors. Therefore, the LCO is applicable in MODES 1, 2, 3, and 4.

The probability and consequences of these events in MODES 5 and 6 are reduced due to the pressure and temperature limitations of these MODES. Therefore, the ice condenser doors are not required to be OPERABLE in these MODES.

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**ACTIONS** Note 1 provides clarification that, for this LCO, separate Condition entry is allowed for each ice condenser door.

Note 2 has been added to allow an intermediate deck or top deck door to be inoperable for a short duration solely due to personnel standing on or opening the door to perform required Surveillances, minor preventative maintenance, or system walkdowns, and does not require entry into Condition B. This is acceptable since the ice bed temperature is normally continuously monitored using an alarm in the control room, which alarms on an increasing ice bed temperature.

### A.1

If one or more ice condenser inlet doors are inoperable due to being physically restrained from opening, the door(s) must be restored to OPERABLE status within 1 hour. The Required Action is necessary to return operation to within the bounds of the containment analysis. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1, "Containment," which requires containment to be restored to OPERABLE status within 1 hour.

### B.1 and B.2

If one or more ice condenser doors are determined to be partially open or otherwise inoperable for reasons other than Condition A or if a door is found that is not closed, it is acceptable to continue unit operation for up to 14 days, provided the ice bed temperature is monitored once per 4 hours to ensure that the open or inoperable door is not allowing enough air leakage to cause the maximum ice bed temperature to approach the melting point. The Completion Time of once per 4 hours is based on the fact that temperature changes cannot occur rapidly in the ice bed because of the large mass of ice involved. The 14 day Completion Time is based on long term ice storage tests that indicate that if the temperature is maintained at or below 27°F, there would not be a significant loss of ice from sublimation. If the maximum ice bed temperature is > 27°F at any time, the situation reverts to Condition C and a Completion Time of 48 hours is allowed to restore the inoperable door

## BASES

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### ACTIONS (continued)

to OPERABLE status or enter into Required Actions D.1 and D.2. Ice bed temperature must be verified to be within the specified Frequency as augmented by the provisions of SR 3.0.2. If this verification is not made, Required Actions D.1 and D.2, not Required Action C.1, must be taken. Entry into Condition B is not required due to personnel standing on or opening an intermediate deck or upper deck door for short durations to perform required surveillances, minor maintenance such as ice removal, or routine tasks such as system walkdowns.

#### C.1

If Required Actions B.1 or B.2 are not met, the doors must be restored to OPERABLE status and closed positions within 48 hours. The 48 hour Completion Time is based on the fact that, with the very large mass of ice involved, it would not be possible for the temperature to decrease to the melting point and a significant amount of ice to melt in a 48 hour period. Condition C is entered from Condition B only when the Completion Time of Required Action B.2 is not met or when the ice bed temperature has not been verified at the required frequency.

#### D.1 and D.2

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

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### SURVEILLANCE REQUIREMENTS

#### SR 3.6.13.1

Verifying, by means of the Inlet Door Position Monitoring System, that the inlet doors are in their closed positions makes the operator aware of an inadvertent opening of one or more doors.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

#### SR 3.6.13.2

Verifying, by visual inspection, that each intermediate deck door is closed and not impaired by ice, frost, or debris provides assurance that the

## BASES

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### SURVEILLANCE REQUIREMENTS (continued)

intermediate deck doors (which form the floor of the upper plenum where frequent maintenance on the ice bed is performed) have not been left open or obstructed.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

#### SR 3.6.13.3

Verifying, by visual inspection, that the ice condenser inlet doors are not impaired by ice, frost, or debris provides assurance that the doors are free to open in the event of a DBA.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

#### SR 3.6.13.4

Verifying the opening torque of the inlet doors provides assurance that no doors have become stuck in the closed position. The value of 675 in-lb is based on the design opening pressure on the doors of 1.0 lb/ft<sup>2</sup>.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

#### SR 3.6.13.5

The torque test Surveillance ensures that the inlet doors have not developed excessive friction and that the return springs are producing a door return torque within limits. The torque test consists of the following:

1. Verify that the torque, T(OPEN), required to cause opening motion at the 40° open position is < 195 in-lb,
2. Verify that the torque, T(CLOSE), required to hold the door stationary (i.e., keep it from closing) at the 40° open position is > 78 in-lb, and
3. Calculate the frictional torque,  $T(\text{FRICT}) = 0.5 \{T(\text{OPEN}) - T(\text{CLOSE})\}$ , and verify that the T(FRICT) is  $\leq 40$  in-lb.

The purpose of the friction and return torque Specifications is to ensure that, in the event of a small break LOCA or SLB, all of the 24 door pairs open uniformly. This assures that, during the initial blowdown phase, the

BASES

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SURVEILLANCE REQUIREMENTS (continued)

steam and water mixture entering the lower compartment does not pass through part of the ice condenser, depleting the ice there, while bypassing the ice in other bays.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.6.13.6

Verifying the OPERABILITY of the intermediate deck doors provides assurance that the intermediate deck doors are free to open in the event of a DBA. The verification consists of visually inspecting the intermediate doors for structural deterioration, verifying free movement of the vent assemblies, and ascertaining free movement of each door when lifted with the applicable force shown below:

	<u>Door</u>	<u>Lifting Force</u>
a.	0-1, 0-5	≤ 37.4 lb
b.	0-2, 0-6	≤ 33.8 lb
c.	0-3, 0-7	≤ 31.0 lb
d.	0-4, 0-8	≤ 31.8 lb

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.6.13.7

Verifying, by visual inspection, that the top deck doors are in place, closed, and not obstructed provides assurance that the doors are performing their function of keeping warm air out of the ice condenser during normal operation, and would not be obstructed if called upon to open in response to a DBA.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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REFERENCES

1. UFSAR, Chapter 6.
  2. 10 CFR 50, Appendix K.
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## B 3.6 CONTAINMENT SYSTEMS

### B 3.6.14 Divider Barrier Integrity

#### BASES

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**BACKGROUND** The divider barrier consists of the operating deck and associated seals, personnel access doors, and equipment hatches that separate the upper and lower containment compartments. Divider barrier integrity is necessary to minimize bypassing of the ice condenser by the hot steam and air mixture released into the lower compartment during a Design Basis Accident (DBA). This ensures that most of the gases pass through the ice bed, which condenses the steam and limits pressure and temperature during the accident transient. Limiting the pressure and temperature reduces the release of fission product radioactivity from containment to the environment in the event of a DBA.

In the event of a DBA, the ice condenser inlet doors (located below the operating deck) open due to the pressure rise in the lower compartment. This allows air and steam to flow from the lower compartment into the ice condenser. The resulting pressure increase within the ice condenser causes the intermediate deck doors and the door panels at the top of the condenser to open, which allows the air to flow out of the ice condenser into the upper compartment. The ice condenses the steam as it enters, thus limiting the pressure and temperature buildup in containment. The divider barrier separates the upper and lower compartments and ensures that the steam is directed into the ice condenser. The ice, together with the containment spray, is adequate to absorb the initial blowdown of steam and water from a DBA as well as the additional heat loads that would enter containment over several hours following the initial blowdown. The additional heat loads would come from the residual heat in the reactor core, the hot piping and components, and the secondary system, including the steam generators. During the post blowdown period, the Air Return System (ARS) returns upper compartment air through the divider barrier to the lower compartment. This serves to continue circulating heated air and steam from the lower compartment through the ice condenser, where the heat is removed by the remaining ice.

Divider barrier integrity ensures that the high energy fluids released during a DBA would be directed through the ice condenser and that the ice condenser would function as designed if called upon to act as a passive heat sink following a DBA.

## BASES

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### APPLICABLE SAFETY ANALYSES

Divider barrier integrity ensures the functioning of the ice condenser to the limiting containment pressure and temperature that could be experienced following a DBA. The limiting DBAs considered relative to containment temperature and pressure are the loss of coolant accident (LOCA) and the steam line break (SLB). The LOCA and SLB are analyzed using computer codes designed to predict the resultant containment pressure and temperature transients. DBAs are assumed not to occur simultaneously or consecutively.

Although the ice condenser is a passive system that requires no electrical power to perform its function, the Containment Spray System and the ARS also function to assist the ice bed in limiting pressures and temperatures. Therefore, the postulated DBAs are analyzed, with respect to containment Engineered Safety Feature (ESF) systems, assuming the loss of one ESF bus, which is the worst case single active failure and results in the inoperability of one train in the Containment Spray System and one ARS fan.

The limiting DBA analyses (Ref. 1) show that the maximum peak containment pressure results from the LOCA analysis and is calculated to be less than the containment design pressure. The maximum peak containment temperature results from the SLB analysis and is discussed in the Bases for LCO 3.6.5, "Containment Air Temperature."

In addition to calculating the overall peak containment pressures, the DBA analyses include calculation of the transient differential pressures that occur across subcompartment walls during the initial blowdown phase of the accident transient. The internal containment walls and structures are designed to withstand these local transient pressure differentials for the limiting DBAs.

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

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### LCO

This LCO establishes the minimum equipment requirements to ensure that the divider barrier performs its safety function of ensuring that bypass leakage, in the event of a DBA, does not exceed the bypass leakage assumed in the accident analysis. Included are the requirements that the personnel access doors and equipment hatches in the divider barrier are OPERABLE and closed and that the divider barrier seal is properly installed and has not degraded with time. An exception to the requirement that the doors be closed is made to allow personnel transit through the divider barrier. The basis of this exception is the assumption that, for personnel transit, the time during which a door is open will be short (i.e., shorter than the Completion Time of 1 hour for Condition A). The divider barrier functions with the ice condenser to limit the pressure and temperature that could be expected following a DBA.

BASES

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APPLICABILITY In MODES 1, 2, 3, and 4, a DBA could cause an increase in containment pressure and temperature requiring the integrity of the divider barrier. Therefore, the LCO is applicable in MODES 1, 2, 3, and 4.

The probability and consequences of these events in MODES 5 and 6 are low due to the pressure and temperature limitations of these MODES. As such, divider barrier integrity is not required in these MODES.

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ACTIONS

A.1

If one or more personnel access doors or equipment hatches are inoperable or open, 1 hour is allowed to restore the door(s) and equipment hatches to OPERABLE status and the closed position. The 1 hour Completion Time is consistent with LCO 3.6.1, "Containment," which requires that containment be restored to OPERABLE status within 1 hour.

Condition A has been modified by a Note to provide clarification that separate Condition entry is allowed for each personnel access door or equipment hatch.

B.1

If the divider barrier seal is inoperable, 1 hour is allowed to restore the seal to OPERABLE status. The 1 hour Completion Time is consistent with LCO 3.6.1, which requires that containment be restored to OPERABLE status within 1 hour.

C.1 and C.2

If divider barrier integrity 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.

BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.6.14.1

Verification, by visual inspection, that the personnel access doors and equipment hatches between the upper and lower containment compartments are closed provides assurance that divider barrier integrity is maintained prior to the reactor being taken from MODE 5 to MODE 4. This SR is necessary because many of the doors and hatches may have been opened for maintenance during the shutdown.

SR 3.6.14.2

Verification, by visual inspection, that the personnel access door and equipment hatch seals, sealing surfaces, and alignments are acceptable provides assurance that divider barrier integrity is maintained. This inspection cannot be made when the door or hatch is closed. Therefore, SR 3.6.14.2 is required for each door or hatch that has been opened, prior to the final closure. Some doors and hatches may not be opened for long periods of time. Those that use resilient materials in the seals must be opened and inspected periodically to provide assurance that the seal material has not aged to the point of degraded performance.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.6.14.3

Verification, by visual inspection, after each opening of a personnel access door or equipment hatch that it has been closed makes the operator aware of the importance of closing it and thereby provides additional assurance that divider barrier integrity is maintained while in applicable MODES.

SR 3.6.14.4

Conducting periodic physical property tests on divider barrier seal test coupons provides assurance that the seal material has not degraded in the containment environment, including the effects of irradiation with the reactor at power. The required test consists of a differential pressure test. The test sequence will be as follows: two coupons will be tested to 60 psid; with no failures, the results are acceptable. If a failure occurs at 60 psid, four coupons will be tested to 30 psid; with no failures, the results are acceptable. If a failure occurs at 30 psid, five coupons will be sent to the manufacturer for LOCA environment simulation (radiation, humidity, temperature) and testing to 15 psid.

BASES

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SURVEILLANCE REQUIREMENTS (continued)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.6.14.5

Visual inspection of the seal around the perimeter provides assurance that the seal is properly secured in place.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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REFERENCES

1. UFSAR, Section 6.2.
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## B 3.6 CONTAINMENT SYSTEMS

### B 3.6.15 Containment Recirculation Drains

#### BASES

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BACKGROUND	<p>The containment recirculation drains consist of the ice condenser drains and the refueling canal drains. The ice condenser is partitioned into 24 bays, each having a pair of inlet doors that open from the bottom plenum to allow the hot steam-air mixture from a Design Basis Accident (DBA) to enter the ice condenser. The drains shall provide a flow area out of the ice condenser of at least 15 square feet. No more than two adjacent bays shall be without drains. Each drain leads to a drain pipe that drops down several feet, then makes one or more 90° bends and exits into the lower compartment. A check (flapper) valve at the end of each pipe keeps warm air from entering during normal operation, but when the water exerts pressure, it opens to allow the water to spill into the lower compartment. This prevents water from backing up and interfering with the ice condenser inlet doors. The water delivered to the lower containment serves to cool the atmosphere as it falls through to the floor and provides a source of borated water at the containment sump for long term use by the Emergency Core Cooling System (ECCS) and the Containment Spray System during the recirculation mode of operation.</p> <p>The two refueling canal drains are at low points in the refueling canal. During a refueling, plugs are installed in the drains and the canal is flooded to facilitate the refueling process. The water acts to shield and cool the spent fuel as it is transferred from the reactor vessel to storage. After refueling, the canal is drained and the plugs removed. In the event of a DBA, the refueling canal drains are the main return path to the lower compartment for Containment Spray System water sprayed into the upper compartment.</p> <p>The ice condenser drains and the refueling canal drains function with the ice bed, the Containment Spray System, and the ECCS to limit the pressure and temperature that could be expected following a DBA.</p>
APPLICABLE SAFETY ANALYSES	<p>The limiting DBAs considered relative to containment temperature and pressure are the loss of coolant accident (LOCA) and the steam line break (SLB). The LOCA and SLB are analyzed using computer codes designed to predict the resultant containment pressure and temperature transients. DBAs are assumed not to occur simultaneously or consecutively. Although the ice condenser is a passive system that requires no electrical power to perform its function, the Containment Spray System and the Air Return System (ARS) also function to assist the ice bed in limiting pressures and temperatures. Therefore, the</p>

BASES

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APPLICABLE SAFETY ANALYSES (continued)

analysis of the postulated DBAs, with respect to Engineered Safety Feature (ESF) systems, assumes the loss of one ESF bus, which is the worst case single active failure and results in one train of the Containment Spray System and one ARS fan being rendered inoperable.

The limiting DBA analyses (Ref. 1) show that the maximum peak containment pressure results from the LOCA analysis and is calculated to be less than the containment design pressure. The maximum peak containment atmosphere temperature results from the SLB analysis and is discussed in the Bases for LCO 3.6.5, "Containment Air Temperature." In addition to calculating the overall peak containment pressures, the DBA analyses include calculation of the transient differential pressures that occur across subcompartment walls during the initial blowdown phase of the accident transient. The internal containment walls and structures are designed to withstand these local transient pressure differentials for the limiting DBAs.

The containment recirculation drains satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

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LCO

This LCO establishes the minimum requirements to ensure that the containment recirculation drains perform their safety functions. The ice condenser floor drain valve disks must be closed to minimize air leakage into and out of the ice condenser during normal operation and must open in the event of a DBA when water begins to drain out. The refueling canal drains must have their plugs removed and remain clear to ensure the return of Containment Spray System water to the lower containment in the event of a DBA. The containment recirculation drains function with the ice condenser, ECCS, and Containment Spray System to limit the pressure and temperature that could be expected following a DBA.

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APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause an increase in containment pressure and temperature, which would require the operation of the containment recirculation drains. Therefore, the LCO is applicable in MODES 1, 2, 3, and 4.

The probability and consequences of these events in MODES 5 and 6 are low due to the pressure and temperature limitations of these MODES. As such, the containment recirculation drains are not required to be OPERABLE in these MODES.

BASES

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ACTIONS

A.1

If one ice condenser floor drain is inoperable, 1 hour is allowed to restore the drain to OPERABLE status. The Required Action is necessary to return operation to within the bounds of the containment analysis. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1, "Containment," which requires that containment be restored to OPERABLE status within 1 hour.

B.1

If one refueling canal drain is inoperable, 1 hour is allowed to restore the drain to OPERABLE status. The Required Action is necessary to return operation to within the bounds of the containment analysis. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1, which requires that containment be restored to OPERABLE status in 1 hour.

C.1 and C.2

If the affected drain(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 and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

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SURVEILLANCE  
REQUIREMENTS

SR 3.6.15.1

Verifying the OPERABILITY of the refueling canal drains ensures that they will be able to perform their functions in the event of a DBA. This Surveillance confirms that the refueling canal drain plugs have been removed and that the drains are clear of any obstructions that could impair their functioning. In addition to debris near the drains, attention must be given to any debris that is located where it could be moved to the drains in the event that the Containment Spray System is in operation and water is flowing to the drains. SR 3.6.15.1 must be performed before entering MODE 4 from MODE 5 after every filling of the canal to ensure that the plugs have been removed and that no debris that could impair the drains was deposited during the time the canal was filled.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.15.2

Verifying the OPERABILITY of the ice condenser floor drains ensures that they will be able to perform their functions in the event of a DBA. Inspecting the drain valve disk ensures that the valve is performing its function of sealing the drain line from warm air leakage into the ice condenser during normal operation, yet will open if melted ice fills the line following a DBA. Verifying that the drain lines are not obstructed ensures their readiness to drain water from the ice condenser.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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REFERENCES      1. UFSAR, Sections 6.2 and 6.5.

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## B 3.7 PLANT SYSTEMS

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

#### BASES

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BACKGROUND	<p>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 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.</p> <p>Five MSSVs are located on each main steam header, outside containment, upstream of the main steam isolation valves, as described in UFSAR, Section 10.3.2 (Ref. 1). The MSSVs must have sufficient capacity to limit the secondary system pressure to <math>\leq 110\%</math> of the steam generator design pressure in order to meet the requirements of the ASME Code, Section III (Ref. 2). The MSSV design includes staggered setpoints, according to Table 3.7.1-2 in the accompanying LCO, so that only the needed valves will actuate. Staggered setpoints reduce the potential for valve chattering that is due to steam pressure insufficient to fully open all valves following a turbine reactor trip.</p>
APPLICABLE SAFETY ANALYSES	<p>The design basis for the MSSVs comes from Reference 2 and its purpose is to limit the secondary system pressure to <math>\leq 110\%</math> of design pressure for any anticipated operational occurrence (AOO) or accident considered in the Design Basis Accident (DBA) and transient analysis.</p> <p>The events that challenge the relieving capacity of the MSSVs, and thus RCS pressure, are those characterized as decreased heat removal events, which are presented in UFSAR, Section 15.2.7 (Ref. 3). Of these, the full power turbine trip without steam dump is the limiting AOO. This event also terminates normal feedwater flow to the steam generators.</p> <p>The safety analysis demonstrates that the transient response for turbine trip occurring from full power without a direct reactor trip presents no hazard to the integrity of the RCS or the Main Steam System. One turbine trip analysis is performed assuming primary system pressure control via operation of the pressurizer relief valves and spray. This analysis demonstrates that the DNB (Departure from Nucleate Boiling) design basis is met. Another analysis is performed assuming no primary system pressure control, but crediting reactor trip on high pressurizer pressure and operation of the pressurizer safety valves. This analysis demonstrates that RCS integrity is maintained by showing that the maximum RCS pressure does not exceed 110% of the design pressure. All cases analyzed demonstrate that the MSSVs maintain Main Steam System integrity by limiting the maximum steam pressure to less than or equal to 110% of the steam generator design pressure.</p>

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## BASES

## APPLICABLE SAFETY ANALYSES (continued)

In addition to the decreased heat removal events, reactivity insertion events may also challenge the relieving capacity of the MSSVs. The UFSAR, Section 15.2.2 (Ref. 7) safety analysis of the uncontrolled rod cluster control assembly (RCCA) bank withdrawal at power event is characterized by an increase in core power and steam generation rate until reactor trip occurs when the Overtemperature  $\Delta T$ , Overpower  $\Delta T$ , High Pressurizer Pressure, High Pressurizer Water Level, or Power Range Neutron Flux-High setpoint is reached. Steam flow to the turbine will not increase from its initial value for this event. The increased heat transfer to the secondary side causes an increase in steam pressure and may result in opening of the MSSVs prior to reactor trip, assuming no credit for operation of the atmospheric relief or condenser steam dump valves. The analysis of the RCCA bank withdrawal (Reference 8) at power slow event demonstrates that the MSSVs are capable of preventing secondary side overpressurization for this AOO.

The UFSAR safety analyses discussed above assume that all of the MSSVs for each steam generator are OPERABLE. If there are inoperable MSSV(s), it is necessary to limit the primary system power during steady-state operation and AOOs to a value that does not result in exceeding the combined steam flow capacity of the turbine (if available) and the remaining OPERABLE MSSVs. The required limitation on primary system power necessary to prevent secondary system overpressurization may be determined by system transient analyses or conservatively arrived at by a simple heat balance calculation. In some circumstances it is necessary to limit the primary side heat generation that can be achieved during an AOO by reducing the setpoint of the Power Range Neutron Flux-High reactor trip function.

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

## LCO

The accident analysis requires that five MSSVs per steam generator be OPERABLE to provide overpressure protection for design basis transients occurring at 102% RTP. The LCO requires that five MSSVs per steam generator be OPERABLE in compliance with Reference 2, and the DBA analysis.

The OPERABILITY of the MSSVs is defined as the ability to open upon demand within the setpoint tolerances, to relieve steam generator overpressure, and reseal when pressure has been reduced. The OPERABILITY of the MSSVs is determined by periodic surveillance testing in accordance with the Inservice Testing Program.

This LCO provides assurance that the MSSVs will perform their designed safety functions to mitigate the consequences of accidents that could result in a challenge to the RCPB, or Main Steam System integrity.

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**BASES**

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**APPLICABILITY** In MODES 1, 2, and 3, five MSSVs per steam generator are required to be OPERABLE to prevent Main Steam System overpressurization.

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.

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

With one or more MSSVs inoperable, action must be taken so that the available MSSV relieving capacity meets Reference 2 requirements. Operation with less than all five MSSVs OPERABLE for each steam generator is permissible, if THERMAL POWER is limited to the relief capacity of the remaining MSSVs. This is accomplished by restricting THERMAL POWER so that the energy transfer to the most limiting steam generator is not greater than the available relief capacity in that steam generator.

A.1 and A.2

With one or more inoperable MSSVs on one or more steam generators, Required Action A.1 requires an appropriate reduction in THERMAL POWER within 4 hours. Therefore, with a reactor power reduction alone there may be insufficient total steam flow capacity provided by the remaining OPERABLE MSSVs to preclude overpressurization in the event of a turbine trip without steam dump. Therefore, a Completion Time of 36 hours is allowed in Required Action A.2 to reduce the Power Range Neutron Flux – High reactor trip setpoints. The Completion Time of 36 hours is based on a reasonable time to correct the MSSV inoperability, the time required to perform the power reduction, operating experience in resetting all channels of a protective function, and on the low probability of the occurrence of a transient that could result in steam generator overpressure during this period.

The maximum THERMAL POWER corresponding to the heat removal capacity of the remaining OPERABLE MSSVs is determined via a conservative heat balance calculation as described in the attachment to Reference 6, with an appropriate allowance for Nuclear Instrumentation System trip channel uncertainties.

Required Action A.2 is modified by a Note, indicating that the Power Range Neutron Flux-High reactor trip setpoint reduction is only required in MODE 1. In MODES 2 and 3 the reactor protection system trips specified in LCO 3.3.1, "Reactor Trip System Instrumentation," provide sufficient protection.

BASES

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## ACTIONS (continued)

The allowed Completion Times are reasonable based on operating experience to accomplish the Required Actions in an orderly manner without challenging unit systems.

B.1 and B.2

If the Required Actions are not completed within the associated Completion Time, or if one or more steam generators have  $\geq 4$  inoperable MSSVs, 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.

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SURVEILLANCE  
REQUIREMENTSSR 3.7.1.1

This SR verifies the OPERABILITY of the MSSVs by the verification of each MSSV lift setpoint in accordance with the Inservice Testing Program. The ASME Code (Ref. 4), requires that safety and relief valve tests be performed in accordance with ANSI/ASME OM-1-2001 through 2003 Addenda (Ref. 5). According to Reference 5, the following tests are required:

- 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 be tested every 24 months. The ASME Code specifies the activities and frequencies necessary to satisfy the requirements. Table 3.7.1-2 allows a  $\pm 3\%$  setpoint tolerance for OPERABILITY; however, the valves are reset to  $\pm 1\%$  during the Surveillance to allow for drift. The lift settings, according to Table 3.7.1-2, correspond to ambient conditions of the valve at nominal operating temperature and pressure.

This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. The MSSVs may be either bench

BASES

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SURVEILLANCE REQUIREMENTS (continued)

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

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REFERENCES

1. UFSAR, Section 10.3.2.
  2. ASME, Boiler and Pressure Vessel Code, Section III, dated 1968, and March 1970 Addenda.
  3. UFSAR, Section 15.2.7.
  4. ASME Code for Operation and Maintenance of Nuclear Power Plants.
  5. ANSI/ASME OM-1-2001 through 2003 Addenda.
  6. NRC Information Notice 94-60, "Potential Overpressurization of the Main Steam System," August 22, 1994.
  7. UFSAR, Section 15.2.2.
  8. AREVA Document 51-5006459-00, "SQN Uncontrolled RCCA Withdrawal Accident Analysis Profile."
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## B 3.7 PLANT SYSTEMS

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

#### BASES

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BACKGROUND	<p>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 generators.</p> <p>One MSIV is located in each main steam line outside, but close to, containment. The MSIVs are downstream from the main steam safety valves (MSSVs) and auxiliary feedwater (AFW) pump turbine steam supply, to prevent MSSV and AFW isolation from the steam generators by MSIV closure. Closing the MSIVs isolates each steam generator from the others, and isolates the turbine, Steam Dump System, and other auxiliary steam supplies from the steam generators.</p> <p>The MSIVs close on a main steam isolation signal generated by either low steam line pressure, high - high containment pressure, or high steam pressure rate. The MSIVs fail closed on loss of control power or control air.</p> <p>Each MSIV has an MSIV bypass valve. Although these bypass valves are normally closed, they receive the same emergency closure signal as do their associated MSIVs. The MSIVs may also be actuated manually.</p> <p>A description of the MSIVs is found in the UFSAR, Section 10.3 (Ref. 1).</p>
APPLICABLE SAFETY ANALYSES	<p>The design basis of the MSIVs is established by the containment analysis for the large steam line break (SLB) inside containment, discussed in the UFSAR, Section 6.2 (Ref. 2). It is also affected by the accident analysis of the SLB events presented in the UFSAR, Section 15.4.2 (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).</p> <p>The limiting case for the containment analysis is the SLB inside containment, with a loss of offsite power following turbine trip, and failure of the MSIV on the affected steam generator to close. At lower powers, the steam generator inventory and temperature are at their maximum, maximizing the analyzed mass and energy release to the containment. Due to reverse flow and failure of the MSIV to close, the additional mass and energy in the steam headers downstream from the other MSIVs contribute to the total release. With the most reactive rod cluster control assembly assumed stuck in the fully withdrawn position, there is an increased possibility that the core will become critical and return to power. The core is ultimately shut down by the boric acid injection delivered by the Emergency Core Cooling System.</p>

## BASES

## APPLICABLE SAFETY ANALYSES (continued)

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

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

- a. A HELB inside containment. In order to maximize the mass and energy release into containment, the analysis assumes that the MSIV for the affected steam generator remains open. For this accident scenario, steam is discharged into containment from all steam generators until the remaining MSIVs close. After MSIV closure, steam is discharged into containment only from the affected steam generator and from the residual steam in the main steam header downstream of the closed MSIVs in the unaffected loops. Closure of the MSIVs isolates the break from the unaffected steam generators.
- b. A break outside of containment and upstream from the MSIVs is not a containment pressurization concern. The uncontrolled blowdown of more than one steam generator must be prevented to limit the potential for uncontrolled RCS cooldown and positive reactivity addition. Closure of the MSIVs isolates the break and limits the blowdown to a single steam generator.
- c. A break downstream of the MSIVs will be isolated by the closure of the MSIVs.
- d. Following a steam generator tube rupture, closure of the MSIVs isolates the ruptured steam generator from the intact steam generators to minimize radiological releases.
- e. The MSIVs are also utilized during other events such as a feedwater line break. This event is less limiting so far as MSIV OPERABILITY is concerned.

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

BASES

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LCO This LCO requires that four MSIVs in the 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.

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

In MODE 4, 5 or 6, the steam generator energy is low; therefore, the MSIVs are not required for isolation of potential high energy secondary system pipe breaks in these MODES.

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ACTIONS A.1

With one MSIV inoperable in MODE 1, action must be taken to restore OPERABLE status within 4 hours. Some repairs to the MSIV can be made with the unit hot. The 4 hour Completion Time is reasonable, considering the low probability of an accident occurring during this time period that would require a closure of the MSIVs.

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 4 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 Times are reasonable, based on operating experience, to reach MODE 2 and to close the MSIVs in an orderly manner and without challenging unit systems.

BASES

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## ACTIONS (continued)

C.1 and C.2

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

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

The 4 hour Completion Time is consistent with that allowed in Condition A.

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

D.1 and D.2

If the MSIVs cannot be restored to OPERABLE status or are not 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 at least in MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from MODE 2 conditions in an orderly manner and without challenging unit systems.

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SURVEILLANCE  
REQUIREMENTSSR 3.7.2.1

This SR verifies that the closure time of each MSIV is within 5 seconds. The MSIV isolation time is within the limit assumed in the accident and containment analyses. This SR also verifies the valve closure time is in accordance with the Inservice Testing Program. This SR is normally performed upon returning the unit to operation following a refueling outage. The MSIVs should not be tested at power, since even a part stroke exercise increases the risk of a valve closure when the unit is generating power. As the MSIVs are not tested at power, they are exempt from the ASME Code (Ref. 5), requirements during operation in MODE 1 or 2.

BASES

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SURVEILLANCE REQUIREMENTS (continued)

The Frequency is in accordance with the Inservice Testing Program.

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

SR 3.7.2.2

This SR verifies that each MSIV can close on an actual or simulated actuation signal. This Surveillance is normally performed upon returning the plant to operation following a refueling outage.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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.

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REFERENCES

1. UFSAR, Section 10.3.
  2. UFSAR, Section 6.2.
  3. UFSAR, Section 15.4.2.
  4. 10 CFR 100.11.
  5. ASME Code for Operation and Maintenance of Nuclear Power Plants.
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## B 3.7 PLANT SYSTEMS

### B 3.7.3 Main Feedwater Isolation Valves (MFIVs), Main Feedwater Regulating Valves (MFRVs) and MFRV Bypass Valves

#### BASES

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**BACKGROUND** The MFIVs isolate main feedwater (MFW) flow to the secondary side of the steam generators following a high energy line break (HELB). The safety related function of the MFRVs is to provide the second isolation of MFW flow to the secondary side of the steam generators following a HELB. Closure of the MFIVs, or MFRVs and MFRV bypass valves terminates flow to the steam generators, terminating the event for feedwater line breaks (FWLBs) occurring upstream of the MFIVs or MFRVs. The consequences of events occurring in the main steam lines or in the MFW lines downstream from the MFIVs will be mitigated by their closure. Closure of the MFIVs, or MFRVs and MFRV 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, or MFRVs and MFRV bypass valves, isolate the nonsafety related portions from the safety related portions of the system. In the event of a secondary side pipe rupture inside containment, the valves limit the quantity of high energy fluid that enters containment through the break, and provide a pressure boundary for the controlled addition of auxiliary feedwater (AFW) to the intact loops.

One MFIV, and one MFRV and its MFRV bypass valve, are located on each MFW line, outside but close to containment. The MFIVs, MFRVs, and MFRV bypass valves are located upstream of the AFW injection point so that AFW may be supplied to the steam generators following MFIV or MFRV closure. The piping volume from these valves to the steam generators must be accounted for in calculating mass and energy releases, and refilled prior to AFW reaching the steam generator following either an SLB or FWLB.

The MFIVs, MFRVs, and MFRV bypass valves, close on receipt of a  $T_{avg}$  - Low coincident with reactor trip (P-4), safety injection, or steam generator water level high-high signal. They may also be actuated manually. In addition to the MFIVs, and the MFRVs and MFRV bypass valves, a check valve outside containment is available. The check valve isolates the feedwater line, penetrating containment, and ensures that the consequences of events do not exceed the capacity of the containment heat removal systems.

BASES

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BACKGROUND (continued)

A description of the MFIVs and MFRVs is found in UFSAR, Section 10.4.7 (Ref. 1).

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APPLICABLE  
SAFETY  
ANALYSES

The design basis of the MFIVs and MFRVs is established by the analyses for the large SLB. It is also influenced by the accident analysis for the large FWLB. Closure of the MFIVs, or MFRVs and MFRV bypass valves, may also be relied on to terminate an SLB for core response analysis and excess feedwater event upon the receipt of a steam generator water level high-high signal or a feedwater isolation signal on high steam generator level.

Failure of an MFIV, MFRV, or the MFRV bypass valves to close following an SLB or FWLB can result in additional mass and energy being delivered to the steam generators, contributing to cooldown. This failure also results in additional mass and energy releases following an SLB or FWLB event.

The MFIVs, MFRVs, and MFRV bypass valves satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

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LCO

This LCO ensures that the MFIVs, MFRVs, and MFRV bypass valves will isolate MFW flow to the steam generators, following an FWLB or main steam line break. These valves will isolate the nonsafety related portions from the safety related portions of the system.

This LCO requires that four MFIVs, four MFRVs and four MFRV bypass valves be OPERABLE. The MFIVs, MFRVs and the MFRV bypass valves are considered OPERABLE when isolation times are within limits and they close on an isolation actuation signal.

Failure to meet the LCO requirements can result in additional mass and energy being released to containment following an SLB or FWLB inside containment. If a feedwater isolation signal on high steam generator level is relied on to terminate an excess feedwater flow event, failure to meet the LCO may result in the introduction of water into the main steam lines.

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APPLICABILITY

The MFIVs, MFRVs and the MFRV bypass valves must be OPERABLE whenever there is significant mass and energy in the Reactor Coolant System and steam generators. This ensures that, in the event of a HELB, a single failure cannot result in the blowdown of more than one steam generator. In MODES 1, 2, and 3, the MFIVs, MFRVs and the MFRV bypass valves are required to be OPERABLE 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 or isolated by a closed manual valve, they are already performing their safety function.

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BASES

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APPLICABILITY (continued)

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

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ACTIONS

The ACTIONS Table is modified by a Note indicating that separate Condition entry is allowed for each valve. This includes separate Condition entry for two valves in the same flow path being inoperable. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable valve. Complying with the Required Actions may allow for continued operation, and subsequent inoperable valves are governed by subsequent Condition entry and application of associated Required Actions.

A.1 and A.2

With one MFIV in one or more flow paths inoperable, action must be taken to restore the affected valves to OPERABLE status, or to close or isolate inoperable affected valves within 72 hours. When these valves are closed or isolated, they are performing their required safety function.

The 72 hour Completion Time takes into account the redundancy afforded by the remaining OPERABLE valves and the low probability of an event occurring during this time period that would require isolation of the MFW flow paths. The 72 hour Completion Time is reasonable, based on operating experience.

Inoperable MFIVs that are closed or isolated must be verified on a periodic basis that they are closed or isolated. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of valve status indications available in the control room, and other administrative controls, to ensure that these valves are closed or isolated.

B.1 and B.2

With one MFRV in one or more flow paths inoperable, action must be taken to restore the affected valves to OPERABLE status, or to close or isolate inoperable affected valves within 72 hours. When these valves are closed or isolated, they are performing their required safety function.

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

## BASES

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### ACTIONS (continued)

flow paths. The 72 hour Completion Time is reasonable, based on operating experience.

Inoperable MFRVs, that are closed or isolated, must be verified on a periodic basis that they are closed or isolated. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of valve status indications available in the control room, and other administrative controls to ensure that the valves are closed or isolated.

#### C.1 and C.2

With one MFRV bypass valve in one or more flow paths inoperable, action must be taken to restore the affected valves to OPERABLE status, or to close or isolate inoperable affected valves within 72 hours. When these valves are closed or isolated, they are performing their required safety function.

The 72 hour Completion Time takes into account the redundancy afforded by the remaining OPERABLE valves and the low probability of an event occurring during this time period that would require isolation of the MFW flow paths. The 72 hour Completion Time is reasonable, based on operating experience.

Inoperable MFRV bypass valves that are closed or isolated must be verified on a periodic basis that they are closed or isolated. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of valve status indications available in the control room, and other administrative controls, to ensure that these valves are closed or isolated.

#### D.1

With two valves in one or more flow paths inoperable, there may be no redundant system to operate automatically and perform the required safety function. Although the containment can be isolated with the failure of two valves in parallel in the same flow path, the double failure 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, at least one valve in each flow path must be restored to OPERABLE status, or the affected flow path isolated within 8 hours. This action returns the system to the condition where at least one valve in each flow path is performing the required

BASES

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ACTIONS (continued)

safety function. The 8 hour Completion Time is reasonable, based on operating experience, to complete the actions required to close the MFIV or MFRV, or otherwise isolate the affected flow path.

E.1 and E.2

If the MFIV(s) and MFRV(s) and the MFRV bypass valve(s) cannot be restored to OPERABLE status, or closed, or isolated 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 full power conditions in an orderly manner and without challenging unit systems.

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SURVEILLANCE  
REQUIREMENTS

SR 3.7.3.1

This SR verifies that the closure time of each MFIV, MFRV, and MFRV bypass valve is within the limit given in Reference 2 and is within that assumed in the accident and containment analyses. This SR also verifies the valve closure time is in accordance with the Inservice Testing Program. This SR is normally performed upon returning the unit to operation following a refueling outage. These valves should not be tested at power since even a part stroke exercise increases the risk of a valve closure with the unit generating power. This is consistent with the ASME Code (Ref. 3), quarterly stroke requirements during operation in MODES 1 and 2.

The Frequency for this SR is in accordance with the Inservice Testing Program.

SR 3.7.3.2

This SR verifies that each MFIV, MFRV, and MFRV bypass valves can close on an actual or simulated actuation signal. This Surveillance is normally performed upon returning the plant to operation following a refueling outage.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

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- REFERENCES
1. UFSAR, Section 10.4.7.
  2. UFSAR, Section 7.3.
  3. ASME Code for Operation and Maintenance of Nuclear Power Plants.
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## B 3.7 PLANT SYSTEMS

### B 3.7.4 Atmospheric Relief Valves (ARVs)

#### BASES

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**BACKGROUND** The ARVs provide a method for cooling the unit to residual heat removal (RHR) entry conditions should the preferred heat sink via the Steam Dump System to the condenser not be available, as discussed in the UFSAR, 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 ARVs may also be required to meet the cooldown when steam pressure drops too low for maintenance of a vacuum in the condenser to permit use of the Steam Dump System.

One ARV line for each of the four steam generators is provided. Each ARV line consists of one ARV.

The ARVs are equipped with pneumatic controllers to permit control of the cooldown rate. The air supplies to the ARVs are from two trains from the plant safety grade Auxiliary Control Air System (ACAS). ACAS train A supplies air to ARVs for steam generators 1 and 3 and train B supplies air to ARVs for steam generators 2 and 4. The ARVs receive the necessary electrical power from the 125 volt vital battery system.

A description of the ARVs is found in Reference 1. The ARVs are OPERABLE with a DC power source and plant safety grade air supply available. In addition, handwheels are provided for manual operation of ARVs for steam generators 1 and 4. Air cylinders connected at control stations outside containment provide an alternate means of operation of the ARVs for steam generators 2 and 3.

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**APPLICABLE SAFETY ANALYSES** The design basis of the ARVs is established by the capability to cool the unit to RHR entry conditions. The ARVs, since their set pressure is slightly lower than the safety valves, prevent excessive lifting of the safety valves. Only two ARVs are required for plant cool down following any credible event.

In the accident analysis presented in Reference 2, the ARVs are assumed to be used by the operator to cool down the unit to RHR entry conditions for accidents accompanied by a loss of offsite power. In Reference 3 (SGTR), the ARVs are assumed to be available following a steam generator tube rupture accompanied by a loss of offsite power. The ARVs allow the operator to establish sufficient subcooling in the RCS so that the primary system will remain subcooled after the RCS pressure is decreased to stop primary to secondary break flow into the ruptured steam generator. Four ARVs are required to be OPERABLE to allow operators to initiate the RCS cooldown, following a steam generator tube rupture, using the ARVs on the intact steam generators. This cooldown

BASES

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APPLICABLE SAFETY ANALYSES (continued)

supports the termination of break flow within the required time specified in the accident analysis to prevent steam generator overfill.

The time required to terminate the primary to secondary break flow for an SGTR is more critical than the time required to cool down to RHR conditions for this event and also for other accidents. Thus, the SGTR is the limiting event for the ARVs.

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

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LCO

Four ARV lines are required to be OPERABLE. One ARV line is required from each of four steam generators to ensure that at least three ARV lines are available to conduct a unit cooldown to establish sufficient subcooling in the RCS following an SGTR, in which one steam generator becomes unavailable.

Failure to meet the LCO can result in the inability to cooldown the RCS to establish sufficient subcooling and prevent steam generator overfill following the steam generator rupture when the condenser is unavailable for use with the Steam Dump System.

An ARV is considered OPERABLE when it is capable of providing controlled relief of the main steam flow and capable of fully opening and closing from the main control room.

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APPLICABILITY

In MODES 1, 2, and 3, and in MODE 4, when a steam generator is being relied upon for heat removal, the ARVs are required to be OPERABLE.

In MODE 5 or 6, an SGTR is not a credible event.

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ACTIONS

A.1

With one or more ARV lines inoperable due to one train of ACAS nonfunctional, action must be taken to restore the ACAS train to functional status within 72 hours. The 72 hour Completion Time is reasonable to repair the nonfunctional ACAS train, based on the availability of the remaining OPERABLE ARV lines, the alternate means to control the inoperable ARVs and the low probability of an event occurring during the time the ACAS train is nonfunctional. Alternate means of operation include valve reach rod handwheels and backup air bottles at the control stations outside containment.

BASES

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ACTIONS (continued)

B.1

With one or more ARV lines inoperable for reasons other than Condition A, action must be taken to restore all ARV lines to OPERABLE status. The 24 hour Completion Time is reasonable to repair inoperable ARV lines, based on the availability of the Steam Dump System and MSSVs, and the low probability of an event occurring during this period that would require the ARV lines.

C.1 and C.2

If the ARV lines cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4, without reliance upon steam generator for heat removal, within 24 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

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SURVEILLANCE  
REQUIREMENTS

SR 3.7.4.1

To perform a controlled cooldown of the RCS, the ARVs must be able to be opened remotely and throttled through their full range. This SR ensures that the ARVs are tested through a full control cycle at least once per fuel cycle. Performance of inservice testing or use of an ARV during a unit cooldown may satisfy this requirement.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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REFERENCES

1. UFSAR, Section 10.3.
  2. UFSAR, Chapter 15.
  3. UFSAR, Section 15.4.3.
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## B 3.7 PLANT SYSTEMS

### B 3.7.5 Auxiliary Feedwater (AFW) System

#### BASES

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**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. Except for the common miniflow line to and supply line from the condensate storage tanks and some shared support facilities such as the condensate storage tanks and parts of the Control Air System, the two reactor units have separate AFW Systems. The normal suction for both units AFW pumps is through a common header connected to two condensate storage tanks (CSTs) (LCO 3.7.6, "Condensate Storage Tank (CST)"). The pumps are grouped into unit specific systems with each systems' pumps aligned to their respective units steam generators secondary side via connections to the main feedwater piping between the main feedwater isolation check valve and the steam generator. The nonessential condensate supply is isolated from the essential portion of the AFW System by check valves. A low AFW pump suction pressure automatically actuates valves from ERCW on a two-out-of-three signal to align the AFW pump suction to ERCW to ensure the AFW pumps have an adequate water supply. 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 relief valves (LCO 3.7.4, "Atmospheric Relief Valves (ARVs)"). If the main condenser is available, steam may be released via the steam dump valves and recirculated to the CST.

The AFW System consists of two motor driven AFW pumps and one steam turbine driven pump configured into three trains. Each motor driven pump provides 100% of AFW flow capacity, and the turbine driven pump provides 200% of the required capacity to the steam generators, as assumed in the accident analysis, except for the Feedwater Line Break (FWLB) and Small Break Loss-of-Coolant accident (SBLOCA). The pumps are equipped with 1½ inch recirculation lines to prevent pump operation against a closed system. Each motor driven AFW pump is powered from an independent Class 1E power supply and feeds two steam generators. The steam turbine driven AFW pump receives steam from two main steam lines upstream of the main steam isolation valves. Each of the steam feed lines will supply 100% of the requirements of the turbine driven AFW pump.

The AFW System is capable of supplying feedwater to the steam generators during normal unit startup, shutdown, and hot standby conditions.

## BASES

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### BACKGROUND (continued)

The turbine driven AFW pump supplies a common header capable of feeding all steam generators with DC powered pneumatic control valves actuated to open by the Engineered Safety Feature Actuation System (ESFAS). One pump at full flow is sufficient to remove decay heat and cool the unit to residual heat removal (RHR) entry conditions. Thus, the requirement for diversity in motive power sources for the AFW System is met.

The AFW System is designed to supply sufficient water to the steam generator(s) to remove decay heat with steam generator pressure at the setpoint of the MSSVs. Subsequently, the AFW System supplies sufficient water to cool the unit to RHR entry conditions, with steam released through the ARVs.

The AFW System actuates automatically on steam generator water level low-low by the ESFAS (LCO 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation"). The system also actuates on loss of offsite power, safety injection, initiation of Anticipated Transient Without Scram (ATWS) Mitigation Actuation Circuitry (AMSAC), trip of one MFW pump with turbine load above 76.6% Unit 1, and trip of all MFW pumps. The AFW System actuations on an AMSAC signal and on a MFW pump trip/power coincident signal are not required as part of this LCO.

The AFW System is discussed in the UFSAR, Section 10.4.7.2 (Ref. 1).

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### APPLICABLE SAFETY ANALYSES

The AFW System mitigates the consequences of any event with loss of normal feedwater.

The design basis of the AFW System is to supply water to the steam generators to remove decay heat and other residual heat by delivering at least the minimum required flow rate to the steam generators at pressures corresponding to the lowest steam generator safety valve set pressure plus 3%.

In addition, the AFW System must supply enough makeup water to replace steam generator secondary inventory lost as the unit cools to MODE 4 conditions. Sufficient AFW flow must also be available to account for flow losses such as pump recirculation and line breaks.

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

- a. Feedwater Line Break (FWLB) and
- b. Loss of MFW.

BASES

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APPLICABLE SAFETY ANALYSES (continued)

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

The AFW System design is such that it can perform its function following an FWLB between the MFW isolation valves and containment, combined with a loss of offsite power following turbine trip, and a single active failure of the steam turbine driven AFW pump. In such a case, one motor driven AFW pump would deliver to the broken MFW header at the pump runout flow until the problem was detected, and flow terminated by the operator. Sufficient flow would be delivered to the intact steam generator by the redundant AFW pump.

The ESFAS automatically actuates the AFW turbine driven pump and associated power operated valves and controls when required to ensure an adequate feedwater supply to the steam generators during loss of power. Air operated valves are provided for each AFW line to control the AFW flow to each steam generator.

The AFW System satisfies the requirements of Criterion 3 of 10 CFR 50.36(c)(2)(ii).

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LCO

This LCO provides assurance that the AFW System will perform its design safety function to mitigate the consequences of accidents that could result in overpressurization of the reactor coolant pressure boundary. Three independent AFW pumps in three diverse trains are required to be OPERABLE to ensure the availability of decay heat removal for all events accompanied by a loss of offsite power and a single failure. This is accomplished by powering two of the pumps from independent emergency buses. The third AFW pump is powered by a different means, a steam driven turbine supplied with steam from a source that is not isolated by closure of the MSIVs.

The AFW System is configured into three trains. The AFW System is considered OPERABLE when the components and flow paths required to provide redundant AFW flow to the steam generators are OPERABLE. This requires that the two motor driven AFW pumps be OPERABLE in two diverse paths, each supplying AFW to separate steam generators. The turbine driven AFW pump is required to be OPERABLE with redundant steam supplies from each of two main steam lines upstream of the MSIVs, and shall be capable of supplying AFW to any of the steam generators. The piping, valves, instrumentation, and controls in the required flow paths are also required to be OPERABLE.

BASES

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## LCO (continued)

Each motor-driven auxiliary feedwater pump (one Train A and one Train B) supplies flow paths to two steam generators. Each flow path contains an automatic air-operated level control valve (LCV). The LCVs have the same train designation as the associated pump and are provided trained air. The turbine driven auxiliary feedwater pump supplies flow paths to all four steam generators. Each of these flow paths contains an automatic opening (non-modulating) air-operated LCV, two of which are designated as Train A, receive A-train air, and provide flow to the same steam generators that are supplied by the B-train motor-driven auxiliary feedwater pump. The remaining two LCVs are designated as Train B, receive B-train air, and provide flow to the same steam generators that are supplied by the A-train motor-driven pump. This design provides the required redundancy to ensure that at least two steam generators receive the necessary flow assuming any single failure. It can be seen from the description provided above that the loss of a single train of air (A or B) will not prevent the auxiliary feedwater system from performing its intended safety function and is no more severe than the loss of a single auxiliary feedwater pump. Therefore, the loss of a single train of auxiliary air only affects the capability of a single motor-driven auxiliary feedwater pump because the turbine-driven pump is still capable of providing flow to two steam generators that are separate from the other motor-driven pump.

Two redundant steam sources are required to be operable to ensure that at least one source is available for the steam-driven auxiliary feedwater (AFW) pump operation following a feedwater or main steam line break. This requirement ensures that the plant remains within its design basis (i.e., AFW to two intact steam generators) given the event of a loss of the No.1 steam generator because of a main steam line or feedwater line break and a single failure of the B-train motor driven AFW pump. The two redundant sources must be aligned such that No.1 steam generator source is open and operable and the No.4 steam generator source is closed and operable.

For instances where one train of emergency raw cooling water (ERCW) is declared inoperable in accordance with technical specifications, the AFW turbine-driven pump is considered operable since it is supplied by both trains of ERCW. Similarly, the AFW turbine-driven pump is considered operable when one train of the AFW loss of power start function is declared inoperable in accordance with Technical Specifications because both 6.9 kilovolt shutdown board logic trains supply this function. This position is consistent with American National Standards Institute/ANSI 58.9 requirements (i.e., postulation of the failure of the opposite train is not required while relying on the TS limiting condition for operation).

## BASES

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### LCO (continued)

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

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### APPLICABILITY

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

In MODE 4 the AFW System may be used for heat removal via the steam generators.

In MODE 5 or 6, the steam generators are not normally used for heat removal, and the AFW System is not required.

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### ACTIONS

A Note prohibits the application of LCO 3.0.4.b to an inoperable AFW train. There is an increased risk associated with entering a MODE or other specified condition in the Applicability with an AFW train inoperable and the provisions of LCO 3.0.4.b, which allow entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, should not be applied in this circumstance.

#### A.1

If the turbine driven AFW train is inoperable due to one inoperable steam supply, or if a turbine driven pump is inoperable for any reason while in MODE 3 immediately following refueling, action must be taken to restore the inoperable equipment to an OPERABLE status within 7 days. The 7 day Completion Time is reasonable, based on the following reasons:

- a. For the inoperability of the turbine driven AFW pump due to one inoperable steam supply, the 7 day Completion Time is reasonable since there is a redundant steam supply line for the turbine driven pump and the turbine driven train is still capable of performing its specified function for most postulated events.
- b. For the inoperability of a turbine driven AFW pump while in MODE 3 immediately subsequent to a refueling, the 7 day Completion Time is reasonable due to the minimal decay heat levels in this situation.

## BASES

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### ACTIONS (continued)

- c. For both the inoperability of the turbine driven pump due to one inoperable steam supply and an inoperable turbine driven AFW pump while in MODE 3 immediately following a refueling outage, the 7 day Completion Time is reasonable due to the availability of redundant OPERABLE motor driven AFW pumps, and due to the low probability of an event requiring the use of the turbine driven AFW pump.

Condition A is modified by a Note which limits the applicability of the Condition for an inoperable turbine driven AFW pump in MODE 3 to when the unit has not entered MODE 2 following refueling. Condition A allows one AFW train to be inoperable for 7 days vice the 72 hour Completion Time in Condition B. This longer Completion Time is based on the reduced decay heat following refueling and prior to the reactor being critical.

#### B.1

With one of the required AFW trains (pump or flow path) inoperable in MODE 1, 2, or 3 for reasons other than Condition A, 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 occurring during this time period.

#### C.1 and C.2

With one of the required motor driven AFW trains (pump or flow path) inoperable and the turbine driven AFW train inoperable due to one inoperable steam supply, action must be taken to restore the affected equipment to OPERABLE status within 48 hours. Assuming no single active failures when in this condition, the accident (a feedline break (FLB) or main steam line break (MSLB)) could result in the loss of the remaining steam supply to the turbine driven AFW pump due to the faulted steam generator (SG).

## BASES

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### ACTIONS (continued)

The 48 hour Completion Time is reasonable based on the fact that the remaining motor driven AFW train is capable of providing 100% of the AFW flow requirements, and the low probability of an event occurring that would challenge the AFW system.

#### D.1 and D.2

When Required Action A.1, B.1, C.1, or C.2 cannot be completed within the required Completion Time, or if two AFW trains are inoperable in MODE 1, 2, or 3 for reasons other than Condition C, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 18 hours.

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

In MODE 4 with two AFW trains inoperable, operation is allowed to continue because only one motor driven pump AFW train is required in accordance with the Note that modifies the LCO. Although not required, the unit may continue to cool down and initiate RHR.

#### E.1

If all three AFW trains are inoperable in MODE 1, 2, or 3, the unit is in a seriously degraded condition with no safety related means for conducting a cooldown, and only limited means for conducting a cooldown with nonsafety related 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.

Required Action E.1 is modified by a Note indicating that all required MODE changes are suspended until one AFW train is restored to OPERABLE status. In this case, LCO 3.0.3 is not applicable because it could force the unit into a less safe condition.

BASES

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ACTIONS (continued)

F.1

In MODE 4, either the reactor coolant pumps or the RHR loops can be used to provide forced circulation. This is addressed in LCO 3.4.6, "RCS Loops - MODE 4." With the required AFW train inoperable, action must be taken to immediately restore the inoperable train to OPERABLE status. The immediate Completion Time is consistent with LCO 3.4.6.

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SURVEILLANCE  
REQUIREMENTS

SR 3.7.5.1

Verifying the correct alignment for manual, power operated, and automatic valves in the AFW System water and steam supply flow paths provides assurance that the proper flow paths will exist for AFW 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 being mispositioned are in the correct position.

The SR is modified by a Note that states one or more AFW trains may be considered OPERABLE during alignment and operation for steam generator level control, if it is capable of being manually (i.e., remotely or locally, as appropriate) realigned to the AFW mode of operation, provided it is not otherwise inoperable. This exception allows the system to be out of its normal standby alignment and temporarily incapable of automatic initiation without declaring the train(s) inoperable. Since AFW may be used during startup, shutdown, hot standby operations, and hot shutdown operations for steam generator level control, and these manual operations are an accepted function of the AFW System, OPERABILITY (i.e., the intended safety function) continues to be maintained.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.7.5.2

Verifying that each AFW pump's developed head at the flow test point is greater than or equal to the required developed head ensures that AFW pump performance has not degraded during the cycle. Flow and differential head are normal tests of centrifugal pump performance required by the ASME Code (Ref 2). Because it is undesirable to introduce cold AFW into the steam generators while they are operating,

## BASES

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### SURVEILLANCE REQUIREMENTS (continued)

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, trend performance, and detect incipient failures by indicating abnormal performance. Performance of inservice testing discussed in the ASME Code (Ref. 2) (only required at 3 month intervals) satisfies this requirement.

This SR is modified by a Note indicating that the SR should be deferred until suitable test conditions are established. This deferral is required because there is insufficient steam pressure to perform the test.

#### SR 3.7.5.3

This SR verifies that AFW can be delivered to the appropriate steam generator in the event of any accident or transient that generates an ESFAS, or in the event the CSTs become depleted, 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 Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

The SR is modified by two Notes. Note 1 states that one or more AFW trains may be considered OPERABLE during alignment and operation for steam generator level control, if it is capable of being manually (i.e., remotely or locally, as appropriate) realigned to the AFW mode of operation, provided it is not otherwise inoperable. This exception allows the system to be out of its normal standby alignment and temporarily incapable of automatic initiation without declaring the train(s) inoperable. Since AFW may be used during startup, shutdown, hot standby operations, and hot shutdown operations for steam generator level control, and these manual operations are an accepted function of the AFW System, OPERABILITY (i.e., the intended safety function) continues to be maintained. Note 2 states that the SR is only required to be met in MODES 1, 2, and 3. It is not required to be met in MODE 4, since the AFW train is only required for the purposes of removing decay heat when the SG is relied upon for heat removal. The operation of the AFW train is by manual means and automatic startup of the AFW train is not required.

BASES

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.5.4

This SR verifies that the AFW pumps will start in the event of any accident or transient that generates an ESFAS by demonstrating that each AFW pump starts automatically on an actual or simulated actuation signal in MODES 1, 2, and 3. In MODE 4, the required pump is already operating and the autostart function is not required.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

This SR is modified by three Notes. Note 1 indicates that the SR may be deferred until suitable test conditions are established. This deferral is required because there is insufficient steam pressure to perform the test. Note 2 states that one or more AFW trains may be considered OPERABLE during alignment and operation for steam generator level control, if it is capable of being manually (i.e., remotely or locally, as appropriate) realigned to the AFW mode of operation, provided it is not otherwise inoperable. This exception allows the system to be out of its normal standby alignment and temporarily incapable of automatic initiation without declaring the train(s) inoperable. Since AFW may be used during startup, shutdown, hot standby operations, and hot shutdown operations for steam generator level control, and these manual operations are an accepted function of the AFW System. OPERABILITY (i.e., the intended safety function) continues to be maintained. Note 3 states that the SR is only required to be met in MODES 1, 2, and 3. It is not required to be met in MODE 4, since the AFW train is only required for the purposes of removing decay heat when the SG is relied upon for heat removal. The operation of the AFW train is by manual means and automatic startup of the AFW train is not required.

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REFERENCES

1. UFSAR, Section 10.4.7.2.
  2. ASME Code for Operation and Maintenance of Nuclear Power Plants.
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## B 3.7 PLANT SYSTEMS

### B 3.7.6 Condensate Storage Tank (CST)

#### BASES

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**BACKGROUND** The CST provides the preferred 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 3.7.5). The steam produced is released to the atmosphere by the main steam safety valves or the atmospheric relief 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 dump valves. The condensed steam is returned to the CST by the hotwell pump. This has the advantage of conserving condensate while minimizing releases to the environment.

The CST consists of a non-seismic qualified carbon steel tank with a capacity of 385,000 gallons. The CST is the preferred and primary source of clean water for the AFW System. The essential raw cooling water (ERCW) system is the backup source of water in addition to being the Safety Grade source of water. The ERCW supply can be manually aligned based on CST level or automatically aligned on a two-out-of-three low-pressure signal in the condensate suction line.

A description of the CST is found in the UFSAR, 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 events in the accident analysis as discussed in the UFSAR, Chapters 6 and 15 (Refs. 2 and 3, respectively). For anticipated operational occurrences and accidents that do not affect the OPERABILITY of the steam generators, the analysis assumption is generally 2 hours at MODE 3, steaming through the MSSVs, followed by a cooldown to residual heat removal (RHR) entry conditions at the design cooldown rate.

The limiting event for the condensate volume is the large feedwater line break coincident with a loss of offsite power. Single failures that also affect this event include the following:

- a. Failure of the diesel generator powering the motor driven AFW pump to the unaffected steam generator (requiring additional steam to drive the remaining AFW pump turbine) and
- b. Failure of the steam driven AFW pump (requiring a longer time for cooldown using only one motor driven AFW pump).

BASES

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APPLICABLE SAFETY ANALYSES (continued)

These are not usually the limiting failures in terms of consequences for these events.

A nonlimiting event is considered a break in either the main feedwater or AFW line near where the two join. This break has the potential for dumping condensate until terminated by operator action. This loss of condensate inventory is partially compensated for by the retention of steam generator inventory.

The CST satisfies Criteria 2 and 3 of 10 CFR 50.36(c)(2)(ii).

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LCO

The CST must contain sufficient cooling water to remove decay heat for 2 hours following a reactor trip from 100.7% RTP, and then to cool down the RCS to RHR entry conditions, assuming a coincident loss of offsite power. In doing this, it must retain sufficient water to ensure adequate net positive suction head for the AFW pumps during cooldown.

The CST level required is equivalent to a usable volume of  $\geq 240,000$  gallons, which is based on holding the unit in MODE 3 for 2 hours, followed by a cooldown to RHR entry conditions within the following 6 hours. This basis is established in Reference 4 and is the minimum volume required for a plant cooldown.

The OPERABILITY of the CST is determined by maintaining the tank level at or above the minimum required level.

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APPLICABILITY

In MODES 1, 2, and 3, and in MODE 4, when steam generator is being relied upon for heat removal, the CST is required to be OPERABLE.

In MODE 5 or 6, the CST is not required because the AFW System is not required.

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ACTIONS

A.1 and A.2

If the CST is not OPERABLE, the OPERABILITY of the backup supply should be verified by administrative means within 4 hours and once every 12 hours thereafter. OPERABILITY of the backup feedwater supply must include verification that the flow paths from the backup water supply to the AFW pumps are OPERABLE, and that the backup supply has the required volume of water available. The CST must be restored to OPERABLE status within 7 days, because the backup supply may be performing this function in addition to its normal functions. The 4 hour Completion Time is reasonable, based on operating experience, to verify the OPERABILITY of the backup water supply. Additionally, verifying the

BASES

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ACTIONS (continued)

backup water supply every 12 hours is adequate to ensure the backup water supply continues to be available. The 7 day Completion Time is reasonable, based on an OPERABLE backup water supply being available, and the low probability of an event occurring during this time period requiring the CST.

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 the steam generator for heat removal, within 18 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

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SURVEILLANCE  
REQUIREMENTS

SR 3.7.6.1

This SR verifies that the CST contains the required volume of cooling water.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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REFERENCES

1. UFSAR, Section 9.2.6.
  2. UFSAR, Chapter 6.
  3. UFSAR, Chapter 15.
  4. UFSAR, Section 10.4.7.
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## B 3.7 PLANT SYSTEMS

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

#### BASES

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<b>BACKGROUND</b>	<p>The CCS 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 CCS also provides this function for various nonessential components, as well as the spent fuel storage pool. The CCS serves as a barrier to the release of radioactive byproducts between potentially radioactive systems and the Essential Raw Cooling Water (ERCW) System, and thus to the environment.</p> <p>The CCS is arranged as two independent, full capacity cooling loops, and has isolatable nonsafety related components. Although each unit's trains are independent, the CCS B trains share components. Up to three of the five CCS pumps may be shared and the two B train component cooling heat exchangers are shared between the two units. Normally, only CCS pump C-S (common-spare) will be aligned to the train B headers of both units along with both 0B heat exchangers, however, either pump 1B-B (Unit 1) or 2B-B (Unit 2) can be realigned to the train B headers if necessary. 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 board, except for the C-S pump which is powered from shared boards. An OPERABLE C-S pump is powered from the Unit 2 "B" board. It can, however, be manually transferred to the Unit 1 "A" board. When the C-S pump is powered from the Unit 1 "A" board, it is considered inoperable because the configuration is not tested. An open surge tank in the system provides an automatic makeup function to ensure that sufficient net positive suction head is available. The pump in each train is automatically started on receipt of a safety injection signal (unit specific safety injection signals except for the C-S pump, which starts from either units safety injection signal).</p> <p>Additional information on the design and operation of the system, along with a list of the components served, is presented in the UFSAR, Section 9.2.1 (Ref. 1). The principal safety related function of the CCS is the removal of decay heat from the reactor via the Residual Heat Removal (RHR) System. This may be during a normal or post accident cooldown and shutdown.</p>
<b>APPLICABLE SAFETY ANALYSES</b>	<p>The design basis of the CCS is for one CCS train to remove the post loss of coolant accident (LOCA) heat load from the containment sump during the recirculation phase, with a maximum CCS temperature of 104.5°F (Ref. 1). The Emergency Core Cooling System (ECCS) LOCA and containment OPERABILITY LOCA each model the maximum and minimum performance of the CCS, respectively. The normal temperature of the CCS is 35 - 95°F, and, during unit cooldown to MODE 5</p>

BASES

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APPLICABLE SAFETY ANALYSES (continued)

( $T_{avg} < 200^{\circ}\text{F}$ ), a maximum temperature of  $120^{\circ}\text{F}$  can be approached. This prevents the containment sump fluid from increasing in temperature during the recirculation phase following a LOCA, and provides a gradual reduction in the temperature of this fluid as it is supplied to the Reactor Coolant System (RCS) by the ECCS pumps.

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

The CCS also functions to cool the unit from RHR entry conditions ( $T_{avg} < 350^{\circ}\text{F}$ ), to MODE 5 ( $T_{avg} < 200^{\circ}\text{F}$ ), during normal and post accident operations. The time required to cool from  $350^{\circ}\text{F}$  to  $200^{\circ}\text{F}$  is a function of the number of CCS and RHR trains operating. One CCS train is sufficient to remove decay heat during subsequent operations with  $T_{avg} < 200^{\circ}\text{F}$ . This assumes a maximum ERCW temperature of  $87^{\circ}\text{F}$  occurring simultaneously with the maximum heat loads on the system.

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

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LCO

The CCS 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 CCS 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 trains of CCS must be OPERABLE. At least one CCS train will operate assuming the worst case single active failure occurs coincident with a loss of offsite power.

A CCS train is considered OPERABLE when:

- a. The pump and associated 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 CCS from other components or systems not required for safety may render those components or systems inoperable but does not affect the OPERABILITY of the CCS.

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APPLICABILITY

In MODES 1, 2, 3, and 4, the CCS is a normally operating system, which must be prepared to perform its post accident safety functions, primarily RCS heat removal, which is achieved by cooling the RHR heat exchanger.

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BASES

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APPLICABILITY (continued)

In MODE 5 or 6, the OPERABILITY requirements of the CCS are determined by the systems it supports.

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ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that the applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops - MODE 4," be entered if an inoperable CCS train results in an inoperable RHR loop. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components.

If one CCS train is inoperable, action must be taken to restore OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE CCS train is adequate to perform the heat removal function. The 72 hour Completion Time is reasonable, based on the redundant capabilities afforded by the OPERABLE train, and the low probability of a DBA occurring during this period.

B.1 and B.2

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

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SURVEILLANCE  
REQUIREMENTS

SR 3.7.7.1

This SR is modified by a Note indicating that the isolation of the CCS flow to individual components may render those components inoperable but does not affect the OPERABILITY of the CCS.

Verifying the correct alignment for manual, power operated, and automatic valves in the CCS flow path provides assurance that the proper flow paths exist for CCS 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 being mispositioned are in the correct position.

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BASES

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SURVEILLANCE REQUIREMENTS (continued)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.7.7.2

This SR verifies proper automatic operation of the CCS pumps on an actual or simulated (i.e, Safety Injection) actuation signal. The CCS is a normally operating system that cannot be fully actuated as part of routine testing during normal operation.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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REFERENCES

1. UFSAR, Section 9.2.1.
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## B 3.7 PLANT SYSTEMS

### B 3.7.8 Essential Raw Cooling Water (ERCW) System

#### BASES

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**BACKGROUND** The ERCW 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, and a normal shutdown, the ERCW system also provides this function for various safety related and nonsafety related components. The safety related function is covered by this LCO.

The ERCW system consists of two separate and independent, 100% capacity, safety related, cooling water trains. The water supply and distribution system is essentially common to both units. Two common trains feed both units. Each train consists of two main supply headers, two strainers, four pumps, two traveling water screens, and associated piping, valving, and instrumentation. To meet the design requirements, with the Ultimate Heat Sink (UHS) temperature at its limit, the system requires two main supply headers, two strainers, and two pumps sharing one traveling screen. The pumps and valves are remote and manually aligned, except in the unlikely event of a loss of coolant accident (LOCA). The pumps aligned to the critical loops and selected by the selector switch are automatically started upon receipt of a safety injection signal, and the essential valves are aligned to their post accident positions. Additionally, each emergency diesel generator has two assigned ERCW pumps. The two assigned ERCW pumps are interlocked so that only the selected pump will start if offsite power is lost.

Additional information about the design and operation of the ERCW system, along with a list of the components served, is presented in the UFSAR, Section 9.2.2 (Ref. 1). The principal safety related function of the ERCW system is the removal of decay heat from the reactor via the CCS.

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**APPLICABLE SAFETY ANALYSES** The design basis of the ERCW system is for one ERCW system train, in conjunction with the CCS and a 100% capacity containment cooling system, to remove core decay heat following a design basis LOCA as discussed in the UFSAR, 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 ECCS pumps. The ERCW system is designed to perform its function with a single failure of any active component, assuming the loss of offsite power.

The ERCW system, in conjunction with the CCS, also cools the unit from residual heat removal (RHR), as discussed in the UFSAR, Section 5.5.7,

BASES

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APPLICABLE SAFETY ANALYSES (continued)

(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 component cooling water and RHR System trains that are operating. One ERCW system train is sufficient to remove decay heat during subsequent operations in MODES 5 and 6. This assumes a maximum ERCW system temperature of 87°F occurring simultaneously with maximum heat loads on the system.

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

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LCO

Two ERCW system trains are required to be OPERABLE to provide the required redundancy to ensure that the system functions to remove post accident heat loads, assuming that the worst case single active failure occurs coincident with the loss of offsite power.

An ERCW system train is considered OPERABLE during MODES 1, 2, 3, and 4 when the required ERCW pumps are operable and the associated piping, valves, heat exchanger, and instrumentation and controls require to perform the safety related function as described in UFSAR Section 9.2.2.

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APPLICABILITY

In MODES 1, 2, 3, and 4, the ERCW system is a normally operating system that is required to support the OPERABILITY of the equipment serviced by the ERCW system and required to be OPERABLE in these MODES.

In MODES 5 and 6, the OPERABILITY requirements of the ERCW system are determined by the systems it supports.

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ACTIONS

A.1 and A.2

Condition A is modified by two Notes that limit the conditions and parameters that allow entry into Condition A. The first Note limits the applicability of Condition A to the time period when the opposite unit is either defueled or in MODE 6 following defueled with refueling water cavity level  $\geq$  23 ft. above the top of the reactor vessel flange. The second Note requires a temperature limitation on UHS Temperature. In

## BASES

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### ACTIONS (continued)

order to credit the temperature limit, the effected ERCW train must be aligned in accordance with UFSAR 9.2.2.2. This will allow the plant configuration to be aligned (i.e., cross-ties exist and isolation of loads to facilitate maintenance activities) to minimize the heat load on the ERCW system to ensure the ERCW system continues to meet its design function.

The 7 day Completion Time is acceptable based on the following:

- The low probability of a DBA occurring during that time;
- The heat load on the ERCW System is substantially lower than assumed for the DBA with the opposite unit defueled or subsequent to defueled; and
- The redundant capabilities afforded by the OPERABLE train.

If one ERCW system train is inoperable for planned maintenance, action must be taken to restore OPERABLE status within 7 days. In this Condition, the remaining OPERABLE ERCW system train is adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure in the OPERABLE ERCW system train could result in loss of ERCW system function.

Required Action A.1 is modified by two Notes. The first Note indicates that the applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources – Operating," should be entered if an inoperable ERCW system 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 ERCW system train results in an inoperable residual heat removal loop. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components.

Required Action A.2 ensures the credited temperature limit for Ultimate Heat Sink is maintained.

### B.1

If one ERCW system train is inoperable for reasons other than Condition A, action must be taken to restore OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE ERCW system train is adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure in the OPERABLE ERCW system train could result in loss of ERCW system function. Required Action B.1 is modified by two Notes.

## BASES

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### ACTIONS (continued)

The first Note indicates that the applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources - Operating," should be entered if an inoperable ERCW system 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 ERCW system train results in an inoperable residual heat removal loop. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components. 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.

#### C.1 and C.2

If the ERCW system train cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours.

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

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### SURVEILLANCE REQUIREMENTS

#### SR 3.7.8.1

This SR is modified by a Note indicating that the isolation of the ERCW system components or systems may render those components inoperable, but does not affect the OPERABILITY of the ERCW system.

Verifying the correct alignment for manual, power operated, and automatic valves in the ERCW system flow path provides assurance that the proper flow paths exist for ERCW system 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 being locked, sealed, or secured. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.8.2

This SR verifies proper automatic operation of the ERCW system valves on an actual or simulated actuation signal. The Safety Injection signal is the automatic actuation signal. The ERCW system is a normally operating system that cannot be fully actuated as part of normal testing. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.7.8.3

This SR verifies proper automatic operation of the ERCW system pumps on an actual or simulated (i.e., Safety Injection) actuation signal. The ERCW system is a normally operating system that cannot be fully actuated as part of normal testing during normal operation.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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REFERENCES

1. UFSAR, Section 9.2.2.
  2. UFSAR, Section 6.2.
  3. UFSAR, Section 5.5.7.
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## B 3.7 PLANT SYSTEMS

### B 3.7.9 Ultimate Heat Sink (UHS)

#### BASES

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<b>BACKGROUND</b>	<p>The UHS provides a heat sink for processing and operating heat from safety related components during a transient or accident, as well as during normal operation. This is done by utilizing the Essential Raw Cooling Water (ERCW) system and the Component Cooling Water System (CCS).</p> <p>The UHS has been defined as that complex of water sources, including necessary retaining structures, and the canals or conduits connecting the sources with, but not including, the cooling water system intake structures as discussed in the UFSAR, Section 9.2.5 (Ref. 1). The two principal functions of the UHS are the dissipation of residual heat after reactor shutdown, and dissipation of residual heat after an accident.</p> <p>Chickamauga Lake (Tennessee River system) is used to meet the requirements for a UHS.</p> <p>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.</p> <p>Additional information on the design and operation of the system, along with a list of components served, can be found in Reference 1.</p>
<b>APPLICABLE SAFETY ANALYSES</b>	<p>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 residual heat removal (RHR) operation. Its maximum post accident heat load occurs approximately 25 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 and RHR are required to remove the core decay heat.</p> <p>The operating limits are based on conservative heat transfer analyses for the worst case LOCA. Reference 1 provides the details of the assumptions used in the analysis, which include worst expected meteorological conditions, conservative uncertainties when calculating decay heat, and worst case single active failure (e.g., single failure of a 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.</p> <p>The UHS satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).</p>

BASES

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LCO The UHS is required to be OPERABLE and is considered OPERABLE if it contains a sufficient volume of water at or below the maximum temperature that would allow the ERCW system 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 ERCW system. To meet this condition, the average ERCW supply header water temperature should not exceed 87°F and the level of the UHS should not fall below 674 ft mean sea level USGS datum during normal unit operation.

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APPLICABILITY In MODES 1, 2, 3, and 4, the UHS is required to support the OPERABILITY of the equipment serviced by the UHS and required to be OPERABLE in these MODES.

In MODE 5 or 6, the OPERABILITY requirements of the UHS are determined by the systems it supports.

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ACTIONS A.1 and A.2

If the UHS is 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.

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SURVEILLANCE REQUIREMENTS SR 3.7.9.1

This SR verifies that adequate long term (30 day) cooling can be maintained. The specified level also ensures that sufficient NPSH is available to operate the ERCW pumps. This SR verifies that the UHS water level is  $\geq$  674 ft mean sea level USGS datum.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.9.2

This SR verifies that the ERCW System is available to cool the CCS to at least its maximum design temperature with the maximum accident or normal design heat loads for 30 days following a Design Basis Accident. This SR verifies that the average ERCW supply header water temperature is  $\leq 87^{\circ}\text{F}$ .

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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- REFERENCES
1. UFSAR, Section 9.2.5.
  2. Regulatory Guide 1.27.
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## B 3.7 PLANT SYSTEMS

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

#### BASES

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<b>BACKGROUND</b>	<p>The CREVS provides a protected environment from which occupants can control the unit following an uncontrolled release of radioactivity, hazardous chemicals, or smoke.</p> <p>The CREVS consists of two independent, redundant trains that recirculate and filter the air in the control room envelope (CRE) and a CRE boundary that limits the inleakage of unfiltered air. Each CREVS train consists of 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, doors, barriers, and instrumentation also form part of the system.</p> <p>The CRE is the area within the confines of the CRE boundary that contains the spaces that control room occupants inhabit to control the unit during normal and accident conditions. The CRE is the area within Elevation 732 of the Control Building which encompasses the Main Control Room, Technical Support Center, Men's and Women's Locker rooms, Men's and Women's Bathrooms, Kitchen, and Relay Room. The CRE is protected during normal operation, natural events, and accident conditions. The CRE boundary is the combination of walls, floor, roof, ducting, doors, penetrations and equipment that physically form the CRE. The OPERABILITY of the CRE boundary must be maintained to ensure that the inleakage of unfiltered air into the CRE will not exceed the inleakage assumed in the licensing basis analysis of design basis accident (DBA) consequences to CRE occupants. The CRE and its boundary are defined in the Control Room Envelope Habitability Program.</p> <p>The CREVS is an emergency system, parts of which may also operate during normal unit operations in the standby mode of operation.</p> <p>Actuation of the CREVS places the system in the emergency mode of operation. Actuation of the system to the emergency radiation state of the emergency mode of operation, closes the unfiltered outside air intake and unfiltered exhaust dampers, and aligns the system for recirculation of the air within the CRE through the redundant trains of HEPA filters and the charcoal adsorbers. The emergency radiation state also initiates pressurization and filtered ventilation of the air supply to the CRE.</p> <p>Outside air is filtered, and added to the air being recirculated from the CRE. Pressurization of the CRE minimizes infiltration of unfiltered air through the CRE boundary from all the surrounding areas adjacent to the CRE boundary.</p>
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BASES

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BACKGROUND (continued)

The air entering the CRE is continuously monitored by radiation monitors. One detector output above the setpoint will cause actuation of the emergency radiation state.

A single CREVS train operating at a flow rate of 4000 cfm ( $\pm 10\%$ ) will pressurize the CRE to 0.125 inches water gauge relative to the outside atmosphere. Additionally, CREVS maintains a slightly positive pressure relative to external areas adjacent to the CRE boundary. The CREVS operation in maintaining the CRE habitable is discussed in the UFSAR, Sections 6.4 and 9.4 (Refs. 1 and 2).

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 CREVS is designed in accordance with Seismic Category I requirements.

The CREVS is designed to maintain a habitable environment in the CRE for 30 days of continuous occupancy after a DBA without exceeding a 5 rem whole body dose or its equivalent to any part of the body.

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APPLICABLE  
SAFETY  
ANALYSES

The CREVS components are arranged in redundant, safety related ventilation trains. The location of components and ducting within the CRE ensures an adequate supply of filtered air to all areas requiring access. The CREVS provides airborne radiological protection for the CRE occupants, as demonstrated by the CRE occupant dose analyses for the most limiting design basis accident, fission product release presented in the UFSAR, Chapter 15 (Ref. 3).

The CREVS provides protection from smoke and hazardous chemicals to the CRE occupants. The analysis of hazardous chemical releases demonstrates that the toxicity limits are not exceeded in the CRE following a hazardous chemical release (Refs. 4 and 5). The evaluation of a smoke challenge demonstrates that it will not result in the inability of the CRE occupants to control the reactor either from the control room or from the remote shutdown panels (Refs. 2 and 4).

The worst case single active failure of a component of the CREVS, assuming a loss of offsite power, does not impair the ability of the system to perform its design function.

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

BASES

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LCO

Two independent and redundant CREVS trains are required to be OPERABLE to ensure that at least one is available if a single active failure disables the other train. Total system failure, such as from a loss of both ventilation trains or from an inoperable CRE boundary, could result in exceeding a dose of 5 rem whole body or its equivalent to any part of the body to the CRE occupants in the event of a large radioactive release.

Each CREVS train is considered OPERABLE when the individual components necessary to limit CRE occupant exposure are OPERABLE. A CREVS train is OPERABLE when the associated:

- a. Fan is OPERABLE;
- b. HEPA filters and charcoal adsorbers are not excessively restricting flow, and are capable of performing their filtration functions;
- c. Ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained; and
- d. Tornado dampers are de-activated and in the open position.

In order for the CREVS trains to be considered OPERABLE, the CRE boundary must be maintained such that the CRE occupant dose from a large radioactive release does not exceed the calculated dose in the licensing basis consequence analyses for DBAs, and that CRE occupants are protected from hazardous chemicals and smoke.

The LCO is modified by a Note allowing the CRE boundary to be opened intermittently under administrative controls. This Note only applies to openings in the CRE boundary that can be rapidly restored to the design condition, such as doors, hatches, floor plugs, and access panels. For entry and exit through doors, the administrative control of the opening is performed by the person(s) entering or exiting the area. For other openings, these controls should be proceduralized and consist of stationing a dedicated individual at the opening who is in continuous communication with the operators in the CRE. This individual will have a method to rapidly close the opening and to restore the CRE boundary to a condition equivalent to the design condition when a need for CRE isolation is indicated.

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APPLICABILITY

In MODES 1, 2, 3, 4, 5, and 6, and during movement of irradiated fuel assemblies, the CREVS must be OPERABLE to ensure that the CRE will remain habitable during and following a DBA.

During movement of irradiated fuel assemblies, the CREVS must be OPERABLE to cope with the release from a fuel handling accident.

BASES

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ACTIONS

A.1

When one CREVS train is inoperable, for reasons other than an inoperable CRE boundary, action must be taken to restore OPERABLE status within 7 days. In this Condition, the remaining OPERABLE CREVS train is adequate to perform the CRE occupant protection function. However, the overall reliability is reduced because a failure in the OPERABLE CREVS train could result in loss of CREVS function. The 7 day Completion Time is based on the low probability of a DBA occurring during this time period, and ability of the remaining train to provide the required capability.

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

If the unfiltered inleakage of potentially contaminated air past the CRE boundary and into the CRE can result in CRE occupant radiological dose greater than the calculated dose of the licensing basis analyses of DBA consequences (allowed to be up to 5 rem whole body or its equivalent to any part of the body), or inadequate protection of CRE occupants from hazardous chemicals or smoke, the CRE boundary is inoperable. Actions must be taken to restore an OPERABLE CRE boundary within 90 days.

During the period that the CRE boundary is considered inoperable, action must be initiated to implement mitigating actions to lessen the effect on CRE occupants from the potential hazards of a radiological or chemical event or a challenge from smoke. Actions must be taken within 24 hours to verify that in the event of a DBA, the mitigating actions will ensure that CRE occupant radiological exposures will not exceed the calculated dose of the licensing basis analyses of DBA consequences, and that CRE occupants are protected from hazardous chemicals and smoke. These mitigating actions (i.e., actions that are taken to offset the consequences of the inoperable CRE boundary) should be preplanned for implementation upon entry into the condition, regardless of whether entry is intentional or unintentional. The 24 hour Completion Time is reasonable based on the low probability of a DBA occurring during this time period, and the use of mitigating actions. The 90 day Completion Time is reasonable based on the determination that the mitigating actions will ensure protection of CRE occupants within analyzed limits while limiting the probability that CRE occupants will have to implement protective measures that may adversely affect their ability to control the reactor and maintain it in a safe shutdown condition in the event of a DBA. In addition, the 90 day Completion Time is a reasonable time to diagnose, plan and possibly repair, and test most problems with the CRE boundary.

BASES

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## ACTIONS (continued)

C.1

When both CREVS trains are inoperable due to the tornado dampers not in the correct position (i.e., open and de-activated) as a result of a tornado warning, action must be taken to restore at least one train of CREVS to OPERABLE status within 8 hours. In this condition, the shutdown of the operating unit would not be reasonable in consideration that the actions that created the inoperable condition were for the protection of the operating unit and would not be expected to last for a significant duration. The 8 hour Completion Time is reasonable based on the low probability of a DBA occurring during this time period, and high probability that the CREVS trains can be returned to OPERABLE status within 8 hours following the tornado warning.

D.1 and D.2

In MODE 1, 2, 3, or 4, if the inoperable CREVS train or the CRE boundary cannot be restored to OPERABLE status within the required Completion Time, the unit must be placed in a MODE that minimizes 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.

E.1 and E.2

In MODE 5 or 6, or during movement of irradiated fuel assemblies, if the inoperable CREVS train cannot be restored to OPERABLE status within the required Completion Time, action must be taken to immediately place the OPERABLE CREVS train in the recirculation mode. This action ensures that the remaining train is OPERABLE, that no failures preventing automatic actuation will occur, and that any active failure would be readily detected.

An alternative to Required Action E.1 is to immediately suspend activities that could result in a release of radioactivity that might require isolation of the CRE. This places the unit in a condition that minimizes the accident risk. This does not preclude the movement of fuel to a safe position.

BASES

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ACTIONS (continued)

F.1

In MODE 5 or 6, or during movement of irradiated fuel assemblies, with two CREVS trains inoperable or with one or more CREVS trains inoperable due to an inoperable CRE boundary, action must be taken immediately to suspend activities that could result in a release of radioactivity that might require isolation of the CRE. This places the unit in a condition that minimizes the accident risk. This does not preclude the movement of fuel to a safe position.

G.1

If both CREVS trains are inoperable in MODE 1, 2, 3, or 4 for reasons other than an inoperable CRE boundary or tornado dampers not in the correct position (i.e., Condition B or C), the CREVS may not be capable of performing the intended function and the unit is in a condition outside the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.

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SURVEILLANCE  
REQUIREMENTS

SR 3.7.10.1

Verifying the correct position of the tornado dampers in the CREVS flow paths provides assurance that the proper flow paths will exist for CREVS operation. This SR does not apply to tornado dampers that are locked, sealed, or otherwise secured in position, since these were verified to be in the correct position prior to locking, sealing, or securing. This Surveillance does not require any testing or damper manipulation. Rather, it involves verification that the tornado dampers are in the correct position (open and de-activated). The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.7.10.2

Standby systems should be checked periodically to ensure that they function properly. As the environment and normal operating conditions on this system are not too severe, testing each train once every month provides an adequate check of this system. Operation for  $\geq 15$  continuous minutes demonstrates OPERABILITY of the system. Periodic operation ensures that blockage, fan or motor failure, or excessive vibration can be detected for corrective action. The CREVS train OPERABILITY will be demonstrated by initiating, from the control room, flow through the HEPA filter and charcoal adsorber train.

BASES

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## SURVEILLANCE REQUIREMENTS (continued)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.7.10.3

This SR verifies that the required CREVS testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The VFTP includes testing the performance of the HEPA filter, charcoal adsorber efficiency, minimum flow rate, and the physical properties of the activated charcoal. Specific test Frequencies and additional information are discussed in detail in the VFTP.

SR 3.7.10.4

This SR verifies that each CREVS train starts automatically, diverts its inlet flow through the HEPA filters and charcoal adsorbers, and operates on an actual or simulated (i.e., safety injection signal or a high radiation signal from the air intake stream) actuation signal.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.7.10.5

This SR verifies the OPERABILITY of the CRE boundary by testing for unfiltered air leakage past the CRE boundary and into the CRE. The details of the testing are specified in the Control Room Envelope Habitability Program.

The CRE is considered habitable when the radiological dose to CRE occupants calculated in the licensing basis analyses of DBA consequences is no more than 5 rem whole body or its equivalent to any part of the body and the CRE occupants are protected from hazardous chemicals and smoke. This SR verifies that the unfiltered air leakage into the CRE is no greater than the flow rate assumed in the licensing basis analyses of DBA consequences. When unfiltered air leakage is greater than the assumed flow rate, Condition B must be entered. Required Action B.3 allows time to restore the CRE boundary to OPERABLE status provided mitigating actions can ensure that the CRE remains within the licensing basis habitability limits for the occupants following an accident. Compensatory measures are discussed in Regulatory Guide 1.196, Section C.2.7.3, (Ref. 6) which endorses, with exceptions, NEI 99-03, Section 8.4 and Appendix F (Ref. 7). These

BASES

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SURVEILLANCE REQUIREMENTS (continued)

compensatory measures may also be used as mitigating actions as required by Required Action B.2. Temporary analytical methods may also be used as compensatory measures to restore OPERABILITY (Ref. 8). Options for restoring the CRE boundary to OPERABLE status include changing the licensing basis DBA consequence analysis, repairing the CRE boundary, or a combination of these actions. Depending upon the nature of the problem and the corrective action, a full scope inleakage test may not be necessary to establish that the CRE boundary has been restored to OPERABLE status.

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REFERENCES

1. UFSAR, Section 6.4.
  2. UFSAR, Section 9.4.
  3. UFSAR, Chapter 15.
  4. UFSAR, Section 2.2.
  5. UFSAR, Section 8.3.1.2.3.
  6. Regulatory Guide 1.196.
  7. NEI 99-03, "Control Room Habitability Assessment," June 2001.
  8. Letter from Eric J. Leeds (NRC) to James W. Davis (NEI) dated January 30, 2004, "NEI Draft White Paper, Use of Generic Letter 91-18 Process and Alternative Source Terms in the Context of Control Room Habitability," (ADAMS Accession No. ML040300694).
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## B 3.7 PLANT SYSTEMS

### B 3.7.11 Control Room Air-Conditioning System (CRACS)

#### BASES

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BACKGROUND	<p>The CRACS provides temperature control for the control room following isolation of the control room.</p>
	<p>The CRACS consists of two independent and redundant trains that provide cooling and heating of recirculated control room air. Each train consists of a chiller package, cooling coils, air handling unit, instrumentation, and controls to provide for control room temperature control. The CRACS is a subsystem providing air temperature control for the control room.</p>
	<p>The CRACS 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 at approximately 75°F. The CRACS operation in maintaining the control room temperature is discussed in the UFSAR, Section 9.4 (Ref. 1).</p>
APPLICABLE SAFETY ANALYSES	<p>The design basis of the CRACS is to maintain the control room temperature for 30 days of continuous occupancy.</p>
	<p>The CRACS components are arranged in redundant, safety related trains. During emergency operation, the CRACS maintains the temperature at approximately 75°F. In addition, the CRACS is designed to maintain the control room temperature at less than the maximum abnormal postulated temperature of 104°F. A single active failure of a component of the CRACS, with a loss of offsite power, does not impair the ability of the system to perform its design function. Redundant detectors and controls are provided for control room temperature control. The CRACS is designed in accordance with Seismic Category I requirements. The CRACS is capable of removing sensible and latent heat loads from the control room, which include consideration of equipment heat loads and personnel occupancy requirements, to ensure equipment OPERABILITY.</p>
	<p>The CRACS satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).</p>
LCO	<p>Two independent and redundant trains of the CRACS are required to be OPERABLE to ensure that at least one is available, assuming a single failure disabling the other train. Total system failure could result in the equipment operating temperature exceeding limits in the event of an accident.</p>

BASES

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LCO (continued)

The CRACS is considered to be OPERABLE when the individual components necessary to maintain the control room temperature are OPERABLE in both trains. These components include the cooling coils, chiller package, air handling unit, and associated temperature control instrumentation. In addition, the CRACS must be OPERABLE to the extent that air circulation can be maintained.

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APPLICABILITY

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

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ACTIONS

A.1

With one CRACS train inoperable, action must be taken to restore OPERABLE status within 30 days. In this Condition, the remaining OPERABLE CRACS train is adequate to maintain the control room temperature within limits. However, the overall reliability is reduced because a single failure in the OPERABLE CRACS train could result in loss of CRACS function. The 30 day Completion Time is based on the low probability of an event requiring control room isolation, the consideration that the remaining train can provide the required protection, and that alternate safety or nonsafety related cooling means are available.

B.1 and B.2

In MODE 1, 2, 3, or 4, if the inoperable CRACS train cannot be restored to OPERABLE status within the required Completion Time, the unit must be placed in a MODE that minimizes the 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.

BASES

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ACTIONS (continued)

C.1 and C.2

In MODE 5 or 6, or during movement of irradiated fuel, if the inoperable CRACS train cannot be restored to OPERABLE status within the required Completion Time, the OPERABLE CRACS 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 active failures will be readily detected.

An alternative to Required Action C.1 is to immediately suspend activities that present a potential for releasing radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes accident risk. This does not preclude the movement of fuel to a safe position.

D.1

In MODE 5 or 6, or during movement of irradiated fuel assemblies, with two CRACS 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 risk. This does not preclude the movement of fuel to a safe position.

E.1

If both CRACS trains are inoperable in MODE 1, 2, 3, or 4, the control room CRACS may not be capable of performing its intended function. Therefore, LCO 3.0.3 must be entered immediately.

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SURVEILLANCE  
REQUIREMENTS

SR 3.7.11.1

This SR verifies that the heat removal capability of the system is sufficient to remove the heat load assumed in the safety analyses in the control room. This SR consists of a combination of testing and calculations.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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REFERENCES

1. UFSAR, Section 9.4.
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## B 3.7 PLANT SYSTEMS

### B 3.7.12 Auxiliary Building Gas Treatment System (ABGTS)

#### BASES

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<b>BACKGROUND</b>	<p>The ABGTS filters airborne radioactive particulates from the area of the fuel pool following a fuel handling accident or loss of coolant accident (LOCA).</p> <p>The ABGTS consists of two independent and redundant trains. Each train consists of a heater, a prefilter, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and a fan. Ductwork, valves or dampers, and instrumentation also form part of the system. The system initiates filtered ventilation of the auxiliary building following receipt of a high radiation signal from the fuel handling area radiation monitors, a high radiation signal from the train-specific Auxiliary Building exhaust vent monitor, a Phase A containment isolation signal from either reactor, or a high temperature signal from the Auxiliary Building air intakes. During plant operations with the containment open to the auxiliary building, the Auxiliary Building Secondary Containment Enclosure (ABSCE) boundary is extended to include the area inside the containment building and the shield building.</p> <p>The ABGTS is a standby system. Upon receipt of the actuating signal, normal air discharge from the auxiliary building is isolated and the stream of ventilation air discharges through the system filter trains.</p> <p>The ABGTS is discussed in the UFSAR, Sections 6.2.3, 15.5.3, and 15.5.6 (Refs. 1, 2 and 3, respectively).</p>
<b>APPLICABLE SAFETY ANALYSES</b>	<p>The ABGTS design basis is established by the consequences of Design Basis Accidents (DBAs), a LOCA during MODES 1, 2, 3 and 4, and a fuel handling accident during operations involving irradiated fuel assemblies. The analysis of the LOCA, given in Reference 2, assumes that radioactive materials leaked from the Emergency Core Cooling System (ECCS) are filtered and adsorbed by the ABGTS. The analysis of the fuel handling accident, given in Reference 3, assumes that the auxiliary building secondary containment enclosure (ABSCE) boundary is capable of being established to ensure the releases from the auxiliary and containment buildings are consistent with the dose consequence analysis, no credit is taken for filtration by the ABGTS.</p> <p>The amount of fission products available for release from the auxiliary building is determined for a fuel handling accident and for a LOCA. These assumptions and the analysis follow the guidance provided in Regulatory Guide 1.4 (Ref. 4) for a LOCA and Regulatory Guide 1.183 (Ref. 5) for the fuel handling accident.</p>

BASES

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APPLICABLE SAFETY ANALYSIS (continued)

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

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LCO

Two independent and redundant trains of the ABGTS are required to be OPERABLE to ensure that at least one train is available, assuming a single failure that disables the other train, coincident with a loss of offsite power. Total system failure could result in the atmospheric release from the auxiliary building exceeding the 10 CFR 100 (Ref. 6) limits in the event of a LOCA.

One train of the ABGTS is required to be OPERABLE to mitigate the consequences of a fuel handling accident involving handling irradiated fuel to limit releases to the environment to within the 10 CFR 50.67 limits.

The ABGTS is considered OPERABLE when the individual components necessary to control exposure in the auxiliary building are OPERABLE in both trains. An ABGTS train is considered OPERABLE when its associated:

- a. Fan is OPERABLE,
- b. HEPA filter and charcoal adsorber are not excessively restricting flow, and are capable of performing their filtration function, and
- c. Heater, ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained.

The LCO is modified by a Note that specifies that only one ABGTS train is required to be OPERABLE during the movement of irradiated fuel assemblies or with fuel stored in the spent fuel pool.

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APPLICABILITY

In MODE 1, 2, 3, or 4, the ABGTS is required to be OPERABLE to provide fission product removal associated with ECCS leaks due to a LOCA and leakage from containment and annulus.

In MODE 5 or 6, the ABGTS is not required to be OPERABLE to provide fission product removal associated with ECCS leaks due to a LOCA since the ECCS is not required to be OPERABLE.

During movement of irradiated fuel, one train of ABGTS is required to be OPERABLE to alleviate the consequences of a fuel handling accident. With fuel stored in the spent fuel pool, one train of ABGTS is required to be OPERABLE to support mitigation of any potential fuel damage resulting from a load drop.

## BASES

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### ACTIONS

LCO 3.0.3 is not applicable while in MODE 5 or 6. However, since crane travel with loads over fuel stored in the spent fuel pool and irradiated fuel assembly movement can occur in MODE 1, 2, 3, or 4, the ACTIONS have been modified by a Note stating that LCO 3.0.3 is not applicable. If storing or moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If storing or moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement or storage is independent of reactor operations. Entering LCO 3.0.3, while in MODE 1, 2, 3, or 4 would require the unit to be shutdown unnecessarily.

#### A.1

With one ABGTS train inoperable in MODE 1, 2, 3, or 4, action must be taken to restore OPERABLE status within 7 days. During this period, the remaining OPERABLE train is adequate to perform the ABGTS function. The 7 day Completion Time is based on the risk from an event occurring requiring the inoperable ABGTS train, and the remaining ABGTS train providing the required protection.

#### B.1

If the ABSCE boundary is inoperable in MODE 1, 2, 3, or 4, the ABGTS trains cannot perform their intended functions. Actions must be taken to restore an OPERABLE ABSCE boundary within 24 hours. During the period that the ABSCE boundary is inoperable, appropriate compensatory measures consistent with the intent, as applicable, of GDC 19, 60, 61, 63, 64 and 10 CFR Part 100 should be utilized to protect plant personnel from potential hazards such as radioactive contamination, toxic chemicals, smoke, temperature and relative humidity, and physical security. Preplanned measures should be available to address these concerns for intentional and unintentional entry into the condition. The 24 hour Completion Time is reasonable based on the low probability of a DBA occurring during this time period, and the use of compensatory measures. The 24 hour Completion Time is a typically reasonable time to diagnose, plan and possibly repair, and test most problems with the ABSCE boundary.

BASES

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ACTIONS (continued)

C.1 and C.2

In MODE 1, 2, 3, or 4, when Required Action A.1 or B.1 cannot be completed within the associated Completion Time, or when both ABGTS trains are inoperable for reasons other than an inoperable ABSCE boundary (i.e., Condition B), the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in MODE 3 within 6 hours and in MODE 5 within 36 hours. The Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

D.1

When the required train of ABGTS is inoperable during movement of irradiated fuel assemblies, action must be taken to place the unit in a condition in which the LCO does not apply. Action must be taken immediately to suspend movement of irradiated fuel assemblies. This does not preclude the movement of fuel to a safe position.

E.1

When the required train of ABGTS is inoperable with fuel stored in the spent fuel pool, action must be taken to prevent the possibility of a load drop over fuel stored in the spent fuel pool. Suspending all crane operation with loads over the spent fuel pool will eliminate the possibility of dropping a load onto fuel assemblies stored in the spent fuel pool.

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SURVEILLANCE  
REQUIREMENTS

SR 3.7.12.1

Standby systems should be checked periodically to ensure that they function properly. As the environmental and normal operating conditions on this system are not severe, testing each train once every month provides an adequate check on this system.

Operation with heaters on for  $\geq 15$  continuous minutes demonstrates OPERABILITY of the system. Periodic operation ensures that heater failure, blockage, fan or motor failure, or excessive vibration can be detected for corrective action. Operation will be demonstrated by initiating, from the control room, flow through the HEPA filter and charcoal adsorber train.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.12.2

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

SR 3.7.12.3

This SR verifies that each ABGTS train starts and operates on an actual or simulated actuation signal. The SR is modified by two Notes that specify when verification of ABGTS actuation for each actuation signal is required to be met. ABGTS actuation on a Containment Phase A isolation signal is required to be met in MODES 1, 2, 3 and 4. ABGTS actuation on fuel storage pool area high radiation signal is required to be met during movement of irradiated fuel assemblies and with fuel stored in the spent fuel pool.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.7.12.4

This SR verifies the integrity of the auxiliary building enclosure (i.e., spent fuel storage area and the ESF pump rooms). The ability of the auxiliary building to maintain negative pressure with respect to potentially uncontaminated adjacent areas is periodically tested to verify proper function of the ABGTS. During the post accident mode of operation, the ABGTS is designed to maintain a slight negative pressure in the auxiliary building, to prevent unfiltered LEAKAGE. The ABGTS is designed to maintain a pressure  $\leq -0.25$  inches water gauge with respect to atmospheric pressure at a flow rate  $\geq 8,100$  and  $\leq 9,900$  cfm to the auxiliary building.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

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- REFERENCES
1. UFSAR, Section 6.2.3.
  2. UFSAR, Section 15.5.3.
  3. UFSAR, Section 15.5.6.
  4. Regulatory Guide 1.4.
  5. Regulatory Guide 1.183.
  6. 10 CFR 100.
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## B 3.7 PLANT SYSTEMS

### B 3.7.13 Spent Fuel Pool Water Level

#### BASES

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**BACKGROUND** The minimum water level in the spent fuel 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 spent fuel pool design is given in the UFSAR, Section 9.1.2 (Ref. 1). A description of the Spent Fuel Pool Cooling and Cleanup System is given in the UFSAR, Section 9.1.3 (Ref. 2). The assumptions of the fuel handling accident are given in the UFSAR, Section 15.5.6 (Ref. 3).

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**APPLICABLE SAFETY ANALYSES** The minimum water level in the spent fuel pool meets the assumptions of the fuel handling accident described in Regulatory Guide 1.183 (Ref. 4). The resultant 2 hour thyroid dose per person at the exclusion area boundary is a small fraction of the 10 CFR 50.67 (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 during a fuel handling accident. With 23 ft of water, 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.

The spent fuel pool water level satisfies Criteria 2 and 3 of 10 CFR 50.36(c)(2)(ii).

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**LCO** The spent fuel pool water level is required to be  $\geq 23$  ft over the top of irradiated fuel assemblies seated in the storage racks. 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 spent fuel pool.

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**APPLICABILITY** This LCO applies whenever irradiated fuel assemblies are in the spent fuel pool, since the potential for a release of fission products exists.

BASES

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ACTIONS

A.1 and A.2

The Required Actions are modified by a Note indicating that LCO 3.0.3 does not apply.

When the initial conditions for prevention of an accident cannot be met, steps should be taken to preclude the accident from occurring. When the spent fuel pool water level is lower than the required level, the movement of fuel assemblies and crane operations with loads in the fuel storage areas is immediately suspended. This action effectively precludes the occurrence of a fuel handling accident. This does not preclude movement of a fuel assembly or crane load to a safe position.

If the spent fuel pool water level is not within the limit while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If the spent fuel pool water level is not within the limit while in MODES 1, 2, 3, and 4, the Required Actions are independent of reactor operations. Therefore, inability to suspend movement of fuel assemblies, suspend crane operations with loads, or restore spent fuel pool water level to within the limit is not sufficient reason to require a reactor shutdown.

The design basis fuel handling accident assumes the drop and damage of an irradiated fuel assembly; however, there are other potential failure mechanisms of the irradiated fuel in the spent fuel pool that could result in the release of fission product gases. As a result, with the spent fuel pool water level less than 23 feet above the top of irradiated fuel assemblies seated in storage racks, the iodine decontamination factor assumption in the design basis fuel handling accident analysis cannot be met.

Required Action A.2 requires the restoration of the spent fuel pool water level to the minimum required level to preserve the assumptions of the fuel handling accident analysis (Ref. 3). The Completion Time of 4 hours is considered sufficient to correct minor problems and restore the water level.

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SURVEILLANCE  
REQUIREMENTS

SR 3.7.13.1

This SR verifies sufficient spent fuel pool water is available in the event of a fuel handling accident. The water level in the spent fuel pool must be checked periodically.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

During refueling operations, the level in the spent fuel pool is in equilibrium with the refueling canal, and the level in the refueling canal is checked daily in accordance with SR 3.9.7.1.

BASES

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- REFERENCES
1. UFSAR, Section 9.1.2.
  2. UFSAR, Section 9.1.3.
  3. UFSAR, Section 15.5.6.
  4. Regulatory Guide 1.183, Rev. 0.
  5. 10 CFR 50.67.
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## B 3.7 PLANT SYSTEMS

### B 3.7.14 Spent Fuel Pool Boron Concentration

#### BASES

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**BACKGROUND** The spent fuel racks have been analyzed in accordance with the Holtec International methodology contained in Holtec Report HI - 992349 (Ref. 1). This methodology ensures that the spent fuel rack multiplication factor,  $k_{\text{eff}}$  is less than or equal to 0.95, as recommended by the NRC guidance contained in NRC Letter to All Power Reactor Licensees from B.K. Grimes, "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," April 14, 1978 and USNRC Internal Memorandum from L. Kopp, "Guidance On The Regulatory Requirements For Criticality Analysis Of Fuel Storage At Light-Water Reactor Power Plants," August 19, 1998 (Refs. 2 and 3). The codes, methods, and techniques contained in the methodology are used to satisfy the  $k_{\text{eff}}$  criterion. The spent fuel storage racks were analyzed using Westinghouse 17x17 Vantage 5H (V5H) fuel assemblies, with enrichments up to 4.95 ( $\pm 0.05$ ) wt% U-235 utilizing credit for checkerboarding, burnup, soluble boron, integral fuel burnable absorbers, gadolinia, and cooling time to ensure that  $k_{\text{eff}}$  is maintained less than or equal to 0.95, including uncertainties, tolerances, and accident conditions. In addition, the Spent Fuel Pool  $k_{\text{eff}}$  is maintained  $< 1.0$ , including uncertainties, tolerances on a 95/95 basis without any soluble boron. Calculations were performed to evaluate the reactivity of fuel types used at SQN. The results show that the Westinghouse 17x17 V5H fuel assembly exhibits the highest reactivity, thereby bounding all fuel types utilized and stored at SQN.

In the high density Spent Fuel Storage Rack design, the spent fuel pool is divided into three separate and distinct regions which, for the purpose of criticality considerations, are considered as separate pools. Region 1 is designed to accommodate new fuel with a maximum enrichment of 4.95 ( $\pm 0.05$ ) wt% U-235, or spent fuel regardless of the discharge fuel burnup in a 1-in-4 checkerboard arrangement of 1 fresh assembly with 3 spent fuel assemblies with specified enrichment, burnup and cooling times. Region 2 is designed to accommodate fuel which has 4.95 ( $\pm 0.05$ ) wt% initial enrichment burned to at least 30.27 megawatt days per kilogram uranium (MWD/KgU) (assembly average), or fuel of other enrichment with a burnup yielding an equivalent reactivity in the fuel racks. Region 3 is designed to accommodate fuel of 4.95 ( $\pm 0.05$ ) wt% initial enrichment or fuel assemblies of any lower reactivity in a 2-out-of-4 checkerboard arrangement with water-filled cells. Fuel assemblies shall be stored in accordance with LCO 3.7.15, "Spent Fuel Pool Storage."

The water in the spent fuel pool normally contains soluble boron, which results in large subcriticality margins under actual operating conditions. However, the NRC guidelines, based upon the accident condition in

BASES

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BACKGROUND (continued)

which all soluble poison is assumed to have been lost, specify that the limiting  $k_{\text{eff}}$  of  $< 1.0$  be evaluated in the absence of soluble boron. Hence, the design of each region is based on the use of unborated water, which maintains each region in a subcritical condition during normal operation with the regions fully loaded. The double contingency principle discussed in ANSI N16.1-1975 and the April 1978 NRC letter (Ref. 4) allows credit for soluble boron under other abnormal or accident conditions, since only a single accident need be considered at one time. For example, the most severe accident scenario is associated with the accidental mishandling of a fresh fuel assembly face adjacent to a fresh fuel assembly of Region 3. This could potentially increase the criticality of Region 3. To mitigate these postulated criticality related accidents, boron is dissolved in the pool water. The soluble boron concentration required to maintain  $k_{\text{eff}} \leq 0.95$  under normal conditions is 300 ppm and 700 ppm under the most severe postulated fuel mis-location accident. Safe operation of the spent fuel storage racks with no movement of assemblies may therefore be achieved by controlling the location of each assembly in accordance with LCO 3.7.15. Prior to movement of an assembly, it is necessary to perform SR 3.7.14.1.

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APPLICABLE  
SAFETY  
ANALYSES

Most accident conditions do not result in an increase in the activity of any of the regions. Examples of these accident conditions are the loss of cooling (reactivity increase with decreasing water density) and the dropping of a fuel assembly on the top of the rack. However, accidents can be postulated that could increase the reactivity. This increase in reactivity is unacceptable with unborated water in the storage pool. Thus, for these accident occurrences, the presence of soluble boron in the spent fuel pool prevents criticality in each region. The most limiting postulated accident with respect to the storage configurations assumed in the spent fuel rack criticality analysis is the misplacement of a nominal  $4.95 (\pm 0.05)$  wt% U-235 fuel assembly into an empty storage cell location in the Region 3 checkerboard storage arrangement.

The amount of soluble boron required to maintain  $k_{\text{eff}} \leq 0.95$  due to either fuel misload accident is 700 ppm (Ref. 1).

A spent fuel boron dilution analysis was performed to ensure that sufficient time is available to detect and mitigate dilution of the spent fuel pool prior to exceeding the  $k_{\text{eff}}$  design basis limit of 0.95 (Ref. 5). The spent fuel pool boron dilution analysis concluded that an inadvertent or unplanned event that would result in a dilution of the spent fuel pool boron concentration from 2000 ppm to 700 ppm is not a credible event. The accident analyses are provided in the UFSAR, Section 4.3.2.7 (Ref. 6).

BASES

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APPLICABLE SAFETY ANALYSES (continued)

The concentration of dissolved boron in the spent fuel pool satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

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LCO The spent fuel pool boron concentration is required to be  $\geq 2000$  ppm. The specified concentration of dissolved boron in the spent fuel pool preserves the assumptions used in the analyses of the potential critical accident scenarios as described in Reference 6. This concentration of dissolved boron is the minimum required concentration for fuel assembly storage and movement within the spent fuel pool.

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

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ACTIONS

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

The Required Actions are modified by a Note indicating that LCO 3.0.3 does not apply.

When the concentration of boron in the spent fuel pool is less than required, immediate action must be taken to preclude the occurrence of an accident or to mitigate the consequences of an accident in progress. This is most efficiently achieved by immediately suspending the movement of fuel assemblies. The concentration of boron is restored simultaneously with suspending movement of fuel assemblies. Alternatively, beginning a verification of the spent fuel pool fuel locations, to ensure proper locations of the fuel, can be performed. However, prior to resuming movement of fuel assemblies, the concentration of boron must be restored. This does not preclude movement of a fuel assembly to a safe position.

If the LCO is not met while moving irradiated fuel assemblies in MODE 5 or 6, LCO 3.0.3 would not be applicable. 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.

BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.7.14.1

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 accidents are fully addressed.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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REFERENCES

1. Stanely E. Turner (Holtec International), "Criticality Safety Analyses of Sequoyah Spent Fuel Racks with Alternative Arrangements," HI-992349.
  2. B.K. Grimes (NRC GL78011), "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," April 14, 1978.
  3. L. Kopp, "Guidance On The Regulatory Requirements For Criticality Analysis Of Fuel Storage At Light-Water Reactor Power Plants," August 19, 1998.
  4. Double contingency principle of ANSI N16.1-1975, as specified in the April 14, 1978 NRC letter (Section 1.2) and implied in the proposed revision to Regulatory Guide 1.13 (Section 1.4, Appendix A).
  5. K K Niyogi (Holtec International), "Boron Dilution Analysis," HI-992302.
  6. UFSAR, Section 4.3.2.7.
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## B 3.7 PLANT SYSTEMS

### B 3.7.15 Spent Fuel Pool Storage

#### BASES

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**BACKGROUND** In the high density Spent Fuel Storage Rack design, the spent fuel storage pool is divided into three separate and distinct regions which, for the purpose of criticality considerations, are considered as separate pools. Region 1 is designed to accommodate new fuel with a maximum enrichment of 4.95 ( $\pm 0.05$ ) wt% U-235, or spent fuel regardless of the discharge fuel burnup in a 1-in-4 checkerboard arrangement of 1 fresh assembly with 3 spent fuel assemblies with specified enrichment, burnup and cooling times. Region 2 is designed to accommodate fuel of 4.95 ( $\pm 0.05$ ) wt% initial enrichment burned to at least 30.27 megawatt days per kilo gram uranium (MWD/KgU) (assembly average), or fuel of other enrichment with a burnup yielding an equivalent reactivity in the fuel racks. Region 3 is designed to accommodate fuel of 4.95 ( $\pm 0.05$ ) wt% initial enrichment or fuel assemblies of any lower reactivity in a 2-out-of-4 checkerboard arrangement with water-filled cells.

The water in the spent fuel storage pool normally contains soluble boron, which results in large subcriticality margins under actual operating conditions. However, the NRC guidelines, based upon the accident condition in which all soluble poison is assumed to have been lost, specify that the limiting  $k_{\text{eff}}$  of 0.95 be evaluated in the absence of soluble boron. Hence, the design of the regions is based on the use of unborated water, which maintains each region in a subcritical condition during normal operation with the regions fully loaded. The double contingency principle discussed in ANSI N-16.1-1975 and the April 1978 NRC letter (Ref. 1) allows credit for soluble boron under other abnormal or accident conditions, since only a single accident need be considered at one time. For example, the most severe accident scenario is associated with the accidental mishandling of a fresh fuel assembly face adjacent to a fresh fuel assembly of Region 3. This could potentially increase the criticality of Region 3. To mitigate these postulated criticality related accidents, boron is dissolved in the pool water. The soluble boron concentration required to maintain  $k_{\text{eff}} \leq 0.95$  under normal conditions is 300 ppm and 700 ppm under the most severe postulated fuel mis-location accident. Safe operation of the spent fuel storage racks may therefore be achieved by controlling the location of each assembly in accordance with the accompanying LCO. Prior to movement of an assembly, it is necessary to perform SR 3.7.14.1.

BASES

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APPLICABLE  
SAFETY  
ANALYSES

Most accident conditions do not result in an increase in the reactivity of any one of the three regions (Ref. 2). Examples of these accident conditions are the loss of cooling and the dropping of a fuel assembly on the top of the rack. However, accidents can be postulated that could increase the reactivity. This increase in reactivity is unacceptable with unborated water in the storage pool. Thus, for these accident occurrences, the presence of soluble boron in the storage pool prevents criticality in all regions. The most limiting postulated accident with respect to the storage configurations assumed in the spent fuel rack criticality analysis is the misplacement of a nominal 4.95 ( $\pm 0.05$ ) wt% U-235 fuel assembly into an empty storage cell location in the Region 3 checkerboard storage arrangement. The amount of soluble boron required to maintain  $k_{\text{eff}} \leq 0.95$  due to this fuel misload accident is 700 ppm.

The configuration of fuel assemblies in the fuel storage pool satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

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LCO

The restrictions on the placement of fuel assemblies within the spent fuel pool, in accordance with Figures 3.7.15-1 through 3.7.15-4 and Tables 3.7.15-1 through 3.7.15-3, in the accompanying LCO, ensures the  $k_{\text{eff}}$  of the spent fuel storage pool will always remain  $\leq 0.95$ , assuming the pool to be flooded with unborated water.

The arrangements in the spent fuel storage pool have the following definitions:

Region 1 is designed to accommodate new fuel with a maximum enrichment of 4.95 ( $\pm 0.05$ ) wt% U-235, (or spent fuel regardless of the fuel burnup), in a 1-in-4 checkerboard arrangement of 1 fresh assembly with 3 spent fuel assemblies with enrichment-burnup and cooling times illustrated in Figure 3.7.15-3 and defined by the equations in Table 3.7.15-1. Cooling time is defined as the period since reactor shutdown at the end of the last operating cycle for the discharged spent fuel assembly. The presence of a removable, non-fissile insert such as a burnable poison rod assembly (BPRA) or either gadolinia or integral fuel burnable absorber (IFBA) in a fresh fuel assembly does not affect the applicability of Figure 3.7.15-3 or Table 3.7.15-1.

Two alternative storage arrays (or sub-arrays) are acceptable in Region 1 if the fresh fuel assemblies contain rods with either gadolinia or IFBA. For these types of assemblies, the minimum burnup of the spent fuel in the 1-of-4 sub-array are defined by the equations in Table 3.7.15-2.

BASES

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LCO (continued)

For Region 1, any of the three sub-arrays illustrated in Figure 3.7.15-2 may be used in any combination provided that:

- a. Each sub-array of 4 fuel assemblies includes, in addition to the fresh fuel assembly, 3 assemblies with enrichment and minimum burnup requirements defined by the equations in Tables 3.7.15-1 and 3.7.15-2, as appropriate.
- b. The arrangement of Region 1 sub-arrays must not allow a configuration with fresh assemblies adjacent to each other.
- c. If Region 1 arrays are used in conjunction with Region 2 or Region 3 arrangements (see below), the arrangements shall not allow fresh fuel assemblies to be adjacent to each other (see also Figure 3.7.15-1).

Region 2 is designed to accommodate fuel of 4.95 ( $\pm$  0.05) wt% U-235 initial enrichment burned to at least 30.27 MWD/KgU (assembly average), or fuel of other enrichments with a burnup yielding an equivalent reactivity in the fuel racks. The minimum required assembly average burnup in MWD/KgU and cooling time is given by the equations in Table 3.7.15-3 in terms of E, where E is the initial enrichment in the axial zone of highest enrichment (wt% U-235). The minimum required burnups are illustrated in Figure 3.7.15-4 in terms of the initial enrichment and cooling time.

The following restrictions apply to the storage of spent fuel in the Region 2 cells:

- a. The spent fuel shall conform to the minimum burnup requirements defined by the equations in Table 3.7.15-3. Linear interpolation between cooling times may be made if desired.
- b. For the interface with Region 1 storage cells, fresh fuel in Region 1 shall not be stored adjacent to spent fuel assemblies in the Region 2 storage cells.

Region 3 is designed to accommodate fuel of 4.95 ( $\pm$  0.05) wt% U-235 initial enrichment (or fuel assemblies of any lower reactivity) in a 2-out-of-4 checkerboard arrangement with water-filled cells. The water-filled cells shall not contain any components bearing any fissile material, but may accommodate miscellaneous items or equipment.

BASES

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LCO (continued)

The following restrictions apply to the storage of spent fuel in the Region 3 cells:

- a. For the interface between Region 1 and Region 3 storage regions, fresh fuel assemblies shall not be stored adjacent to each other.
- b. If miscellaneous items or equipment are stored in the water cells of Region 3, the total volume of the miscellaneous items shall be no more than 75% of the storage cell volume.
- c. No fuel rods, assemblies, or items containing fissile material shall be stored in the water cells of Region 3.

An empty cell is less reactive than any cell containing fuel and therefore may be used as a Region 1, Region 2, or Region 3 cell in any arrangement.

Region 2 array described above may be used in the 15 x 15 storage rack module in the cask loading area of the cask pit.

A nominal concentration of 2000 ppm boron in the pool water. This concentration of soluble boron provides a margin sufficient to allow timely detection of a boron dilution accident and corrective action before the minimum concentration (700 ppm) required to protect against the most severe postulated fuel handling accident or before the minimum concentration (300 ppm) required to maintain the storage configuration design basis ( $k_{eff}$  less than 0.95) is reached.

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APPLICABILITY	This LCO applies whenever any fuel assembly is stored in Regions 1 through 3 of the fuel storage pool.
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ACTIONS	<p><u>A.1</u></p> <p>Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply.</p> <p>When the configuration of fuel assemblies stored in Regions 1 through 3 of the spent fuel storage pool is not in accordance with Figures 3.7.15-1 through 3.7.15-4 and Tables 3.7.15-1 through 3.7.15-3, the immediate action is to initiate action to make the necessary fuel assembly movement(s) to bring the configuration into compliance with Figures 3.7.15-1 through 3.7.15-4 and Tables 3.7.15-1 through 3.7.15-3.</p> <p>If unable to move irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not be applicable. If unable to move irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the action is independent of</p>
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BASES

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ACTIONS (continued)

reactor operation. Therefore, inability to move fuel assemblies is not sufficient reason to require a reactor shutdown.

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SURVEILLANCE  
REQUIREMENTS

SR 3.7.15.1

This SR verifies by administrative means that the initial enrichment and burnup of the fuel assembly is in accordance with Figures 3.7.15-1 through 3.7.15-4 and Tables 3.7.15-1 through 3.7.15.3, in the accompanying LCO.

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REFERENCES

1. Double contingency principle of ANSI N16.1-1975, as specified in the April 14, 1978 NRC letter (Section 1.2) and implied in the proposed revision to Regulatory Guide 1.13 (Section 1.4, Appendix A).
  2. UFSAR, Section 4.3.2.7.
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## B 3.7 PLANT SYSTEMS

### B 3.7.16 Secondary Specific Activity

#### BASES

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##### BACKGROUND

Activity in the secondary coolant results from steam generator tube leakage from the Reactor Coolant System (RCS). Under steady state conditions, the activity is primarily iodines with relatively short half lives and, thus, indicates current conditions. During transients, I-131 spikes have been observed as well as increased releases of some noble gases. Other fission product isotopes, as well as activated corrosion products in lesser amounts, may also be found in the secondary coolant.

A limit on secondary coolant specific activity during power operation minimizes releases to the 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 0.35  $\mu\text{Ci/gm}$  (LCO 3.4.16, "RCS Specific Activity"). The steam line failure is assumed to result in the release of the noble gas and iodine activity contained in the steam generator inventory, the feedwater, and the reactor coolant LEAKAGE. Most of the iodine isotopes have short half lives (i.e., < 20 hours).

With the specified activity limit, the resultant 8 hour thyroid dose to a person at the exclusion area boundary (EAB) would be about 5.4 rem for 8 hours following a trip from full power, with a steam line break and a loss of AC power to plant auxiliaries.

Operating a unit at the allowable limits could result in a 8 hour EAB exposure of a small fraction of the 10 CFR 100 (Ref. 1) limits, or the limits established as the NRC staff approved licensing basis.

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##### APPLICABLE SAFETY ANALYSES

The accident analysis of the main steam line break (MSLB), as discussed in the UFSAR, Chapter 15 (Ref. 2) assumes the initial secondary coolant specific activity to have a radioactive isotope concentration of 0.10  $\mu\text{Ci/gm}$  DOSE EQUIVALENT I-131. This assumption is used in the analysis for determining the radiological consequences of the postulated accident. The accident analysis, based on this and other assumptions, shows that the radiological consequences of an MSLB do not exceed 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 generators are available for core decay heat dissipation by venting steam to the atmosphere through the MSSVs and steam generator atmospheric relief valves (ARVs). The Auxiliary Feedwater System supplies the necessary

BASES

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APPLICABLE SAFETY ANALYSES (continued)

makeup to the steam generators. Venting continues until the reactor coolant temperature and pressure have decreased sufficiently for the Residual Heat Removal System to complete the cooldown.

In the evaluation of the radiological consequences of this accident, the activity released from the steam generator connected to the failed steam line is assumed to be released directly to the environment. The unaffected steam generators are assumed to discharge steam and any entrained activity through the MSSVs and ARVs during the event. Since no credit is taken in the analysis for activity plateout or retention, the resultant radiological consequences represent a conservative estimate of the potential integrated dose due to the postulated steam line failure.

Secondary specific activity limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

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LCO

As indicated in the Applicable Safety Analyses, the specific activity of the secondary coolant is required to be  $\leq 0.10 \mu\text{Ci/gm DOSE EQUIVALENT I-131}$  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.

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APPLICABILITY

In MODES 1, 2, 3, and 4, the limits on secondary specific activity apply due to the potential for secondary steam releases to the atmosphere.

In MODES 5 and 6, the steam generators are not being used for heat removal. Both the RCS and steam generators are depressurized, and primary to secondary LEAKAGE is minimal. Therefore, monitoring of secondary specific activity is not required.

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ACTIONS

A.1 and A.2

DOSE EQUIVALENT I-131 exceeding the allowable value in the secondary coolant, is an indication of a problem in the RCS and contributes to increased post accident doses. If the secondary specific activity cannot be restored to within limits within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

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BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.7.16.1

This SR verifies that the secondary specific activity is within the limits of the accident analysis. A gamma isotopic analysis of the secondary coolant, which determines DOSE EQUIVALENT I-131, confirms the validity of the safety analysis assumptions as 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 Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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REFERENCES

1. 10 CFR 100.11.
  2. UFSAR, Chapter 15.
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## B 3.7 PLANT SYSTEMS

### B 3.7.17 Cask Pit Pool Boron Concentration

#### BASES

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**BACKGROUND** The cask pit pool consists of a deep pool with adjacent shelf area. The cask pit is intended to be used for spent fuel shipment activities. High density spent fuel storage racks have been approved for addition and use in the cask loading area of the cask pit (Ref. 1). The 15 x 15 module could store 225 fuel assemblies and is designed to maintain stored fuel having an initial enrichment of up to 5 wt% U-235, in a safe, coolable, and sub-critical configuration during normal discharge, full core offload storages and postulated accident conditions.

A description of the spent fuel rack analysis is provided in the Bases for LCO 3.7.14, "Spent Fuel Pool Boron Concentration." Fuel assemblies shall be stored in accordance with LCO 3.7.15, "Spent Fuel Pool Storage." As described in the Bases for LCO 3.7.15, Region 2 arrays may be used in the 15 x 15 storage rack module in the cask loading area of the cask pit.

The water in the cask pit pool normally contains soluble boron, which results in large subcriticality margins under actual operating conditions. However, the NRC guidelines, based upon the accident condition in which all soluble poison is assumed to have been lost, specify that the limiting  $k_{\text{eff}}$  of  $< 1.0$  be evaluated in the absence of soluble boron. Hence, the design of each region is based on the use of unborated water, which maintains each region in a subcritical condition during normal operation with the regions fully loaded. The double contingency principle discussed in ANSI N-16.1-1975 and the April 1978 NRC letter (Ref. 2) allows credit for soluble boron under other abnormal or accident conditions, since only a single accident need be considered at one time. For example, the most severe accident scenario is associated with the accidental mishandling of a fresh fuel assembly face adjacent to a fresh fuel assembly of Region 3 in the spent fuel pool. This could potentially increase the criticality of Region 3. To mitigate these postulated criticality related accidents, boron is dissolved in the pool water. The soluble boron concentration required to maintain  $k_{\text{eff}} \leq 0.95$  under normal conditions is 300 ppm and 700 ppm under the most severe postulated fuel mis-location accident. Safe operation of the spent fuel storage racks with no movement of assemblies may therefore be achieved by controlling the location of each assembly in accordance with LCO 3.7.15. Prior to movement of an assembly, it is necessary to perform SR 3.7.17.1.

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**APPLICABLE SAFETY ANALYSES** Most accident conditions do not result in an increase in the activity of any of the regions. Examples of these accident conditions are the loss of cooling (reactivity increase with decreasing water density) and the dropping of a fuel assembly on the top of the rack. However, accidents

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BASES

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APPLICABLE SAFETY ANALYSES (continued)

can be postulated that could increase the reactivity. This increase in reactivity is unacceptable with unborated water in the storage pool. Thus, for these accident occurrences, the presence of soluble boron in the cask pit pool prevents criticality in Region 2. The most limiting postulated accident with respect to the cask pit pool has been determined to occur in the spent fuel pool. The postulated accident with respect to the storage configurations assumed in the spent fuel rack criticality analysis is the misplacement of a nominal 4.95 ( $\pm 0.05$ ) wt% U-235 fuel assembly into a storage cell location in the Region 3 checkerboard storage arrangement for an irradiated fuel assembly.

The amount of soluble boron required to maintain  $k_{\text{eff}} \leq 0.95$  due to either fuel misload accident is 700 ppm (Ref. 1).

A spent fuel boron dilution analysis was performed to ensure that sufficient time is available to detect and mitigate dilution of the spent fuel pool prior to exceeding the  $k_{\text{eff}}$  design basis limit of 0.95 (Ref. 3). The spent fuel pool boron dilution analysis concluded that an inadvertent or unplanned event that would result in a dilution of the spent fuel pool boron concentration from 2000 ppm to 700 ppm is not a credible event. The accident analyses are provided in the UFSAR, Section 4.3.2.7 (Ref. 4).

The concentration of dissolved boron in the cask pit pool satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

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LCO

The cask pit pool boron concentration is required to be  $\geq 2000$  ppm. The specified concentration of dissolved boron in the cask pit pool preserves the assumptions used in the analyses of the potential critical accident scenarios as described in Reference 4. This concentration of dissolved boron is the minimum required concentration for fuel assembly storage and movement within the cask pit pool.

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APPLICABILITY

This LCO applies whenever fuel assemblies are stored in the cask pit pool, until a complete cask pit pool verification has been performed following the last movement of fuel assemblies in the cask pit 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.

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ACTIONS

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

The Required Actions are modified by a Note indicating that LCO 3.0.3 does not apply.

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BASES

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ACTIONS (continued)

When the concentration of boron in the cask pit pool is less than required, immediate action must be taken to preclude the occurrence of an accident or to mitigate the consequences of an accident in progress. This is most efficiently achieved by immediately suspending the movement of fuel assemblies. The concentration of boron is restored simultaneously with suspending movement of fuel assemblies. Alternatively, beginning a verification of the cask pit pool fuel locations, to ensure proper locations of the fuel, can be performed. However, prior to resuming movement of fuel assemblies, the concentration of boron must be restored. This does not preclude movement of a fuel assembly to a safe position.

If the LCO is not met while moving irradiated fuel assemblies in MODE 5 or 6, LCO 3.0.3 would not be applicable. 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.

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SURVEILLANCE  
REQUIREMENTS

SR 3.7.17.1

This SR verifies that the concentration of boron in the cask pit pool is within the required limit. As long as this SR is met, the analyzed accidents are fully addressed.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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REFERENCES

1. Stanely E. Turner (Holtec International), "Criticality Safety Analyses of Sequoyah Spent Fuel Racks with Alternative Arrangements," HI-992349.
  2. Double contingency principle of ANSI N16.1-1975, as specified in the April 14, 1978 NRC letter (Section 1.2) and implied in the proposed revision to Regulatory Guide 1.13 (Section 1.4, Appendix A).
  3. K K Niyogi (Holtec International), "Boron Dilution Analysis," HI-992302.
  4. UFSAR, Section 4.3.2.7.
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## B 3.8 ELECTRICAL POWER SYSTEMS

### B 3.8.1 AC Sources - Operating

#### BASES

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##### BACKGROUND

The Class 1E AC Electrical Power Distribution System AC sources consist of the offsite power sources (preferred power sources), 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 Electrical Power Distribution System is divided into two redundant and independent load groups with two 6.9 kV Shutdown Boards in each load group. Each 6.9 kV Shutdown Board has a connection to a preferred offsite power source and a DG. The 6.9 kV Shutdown Boards in a load group (i.e., 1A-A and 2A-A, or 1B-B and 2B-B) are normally powered by the same offsite power circuit. Each 6.9 kV Shutdown Board can also be powered by a dedicated DG. Two DGs associated with one load group can provide all safety related functions to mitigate a loss of coolant accident (LOCA) in one unit and safely shut down the other unit. The Train A and Train B ESF systems each provide for the minimum safety functions necessary to shut down the plant and maintain it in a safe shutdown condition.

The Unit 1 core cooling systems and containment systems (e.g., Safety Injection (SI), Auxiliary Feedwater (AFW), Residual Heat Removal (RHR), Centrifugal Charging pump, Containment Spray, and Air Return System (ARS) fan) are unitized (not shared with Unit 2) and are powered from 6.9 kV Shutdown Boards 1A-A and 1B-B. However, some safety-related systems (e.g., Essential Raw Cooling Water (ERCW), Component Cooling Water (CCS), Emergency Gas Treatment (EGTS), Auxiliary Building Gas Treatment (ABGTS), Control Room Emergency Ventilation (CREVS), and Control Room Air-Conditioning (CRACS)) are shared with Unit 2. The AC sources for these loads are distributed across all four 6.9 kV Shutdown Boards. Therefore, two qualified circuits between the offsite transmission network and the onsite Class 1E AC Electrical Power Distribution System and a separate and independent DG for each 6.9 kV Shutdown Board ensure the availability of power to each of these systems.

The offsite power distribution system consists of two 161 kV buses supplied by eight 161 kV feeders and two 500 kV buses supplied by five 500 kV feeders. The output of the Unit 1 main generator is fed to the 500 kV buses and the output of the Unit 2 main generator is fed to the 161 kV buses. The output of each unit's main generator is also capable of

## BASES

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### BACKGROUND (continued)

supplying power to an offsite circuit (via the Unit Station Service Transformers (USSTs)) with a Generator Circuit Breaker (GCB) providing isolation between the main generator and the main bank transformers. When the main generator is not operating, the main bank transformers function as step down transformers and supply electrical power from the grid to the USSTs.

Offsite power can also be supplied by the Common Station Service Transformers (CSSTs) via the 6.9 kV Start Buses and 6.9 kV Unit Boards. Offsite power will normally be supplied from the USSTs to the 6.9 kV Shutdown Boards via the 6.9 kV Unit Boards, and will automatically transfer at least one power supply to an alternate power supply (CSST A or CSST C) on a trip of the Power Circuit Breakers (PCBs). CSST C is the alternate power source for 6.9 kV Shutdown Boards 1A-A and 2A-A, and CSST A is the alternate power source for 6.9 kV Shutdown Boards 1B-B and 2B-B. (CSST B is a spare transformer with two sets of secondary windings that can be used to supply a total of two Start Buses for CSST A and/or CSST C, with each Start Bus on a separate CSST B secondary winding.) Therefore, two electrically and physically separated circuits provide AC power through a combination of the USSTs and/or CSSTs to the 6.9 kV Shutdown Boards. Each offsite circuit is capable of providing power to one train of ESF loads. A detailed description of the offsite power network and the circuits to the Class 1E 6.9 kV Shutdown Boards is found in UFSAR, 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 (beginning at the switchyard) to one load group of Class 1E 6.9 kV Shutdown Boards (ending at the supply side of the normal or alternate supply circuit breaker).

The onsite standby power source for each 6.9 kV Shutdown Board is a dedicated DG. DGs 1A-A, 1B-B, 2A-A, and 2B-B are separate and independent and are dedicated to 6.9 kV Shutdown Boards 1A-A, 1B-B, 2A-A, and 2B-B, respectively. Each diesel generator set consists of two diesel engines in tandem driving a common generator with a normal synchronous speed of approximately 900 rpm. A DG starts automatically on a safety injection (SI) signal (i.e., low pressurizer pressure, high containment pressure, or low steam line pressure signals) or on a 6.9 kV Shutdown Board degraded voltage, unbalanced voltage, or loss-of-voltage signal (refer to LCO 3.3.5, "Loss of Power (LOP) Diesel Generator (DG) Start Instrumentation"). After the DG has started, it will automatically tie to its respective 6.9 kV Shutdown Board after offsite power is tripped as a consequence of a 6.9 kV Shutdown Board degraded voltage, unbalanced voltage, or loss-of-voltage signal, independent of or

## BASES

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### BACKGROUND (continued)

coincident with an SI signal. The DGs will also start and operate in the standby mode without tying to the 6.9 kV Shutdown Board on an SI signal alone. Following the trip of offsite power, a loss-of-voltage signal strips nonpermanent loads from the 6.9 kV Shutdown Board. When the DG is tied to the 6.9 kV Shutdown Board, loads are then sequentially connected to its respective 6.9 kV Shutdown Board by individual load sequence timers.

In the event of a loss of preferred power, the 6.9 kV Shutdown Boards 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 LOCA.

Certain required unit loads are returned to service in a predetermined sequence in order to prevent overloading the DG in the process. Within the required time interval (UFSAR Table 8.3.1-2) after the initiating signal is received, automatic and permanently connected loads needed to recover the unit or maintain it in a safe condition are returned to service.

Ratings for the 1A-A, 1B-B, 2A-A and 2B-B DGs satisfy the requirements of Regulatory Guide 1.9 (Ref. 3). The continuous service rating of each DG is 4400 kW with 10% overload permissible for up to 2 hours in any 24 hour period. The ESF loads that are powered from the 6.9 kV Shutdown Boards are listed in Reference 2.

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### APPLICABLE SAFETY ANALYSES

The initial conditions of DBA and transient analyses in the UFSAR, 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 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 10 CFR 50.36(c)(2)(ii).

## BASES

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### LCO

Two qualified circuits between the offsite transmission network and the onsite Class 1E Electrical Power System and separate and independent DGs for each 6.9 kV Shutdown Board 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.

Each qualified offsite circuit must be physically independent, capable of maintaining rated frequency and voltage, and accepting required loads during an accident, while connected to the 6.9 kV Shutdown Boards.

The minimum required switchyard voltages are determined by evaluation of plant accident loading and the associated voltage drops between the transmission network and these loads. These minimum voltage values are provided to TVA's Transmission Operations for use in system studies to support operation of the transmission network in a manner that will maintain the necessary voltages. Transmission Operations is required to notify SQN Operations if it is determined that the transmission network may not be able to support accident loading or shutdown operations as required by 10 CFR 50, Appendix A, GDC-17. Any offsite power circuits supplied by the transmission network that are not able to support accident loading or shutdown operations are inoperable.

The USSTs utilize auto load tap changers to provide the required voltage response for accident loading. The load tap changer associated with a USST is required to be functional and in "automatic" for the USST to supply power to a 6.9 kV Unit Board.

Each required offsite circuit is that combination of power sources described below that are either connected to the Class 1E AC Electrical Power Distribution System, or is available to be connected to the Class 1E AC Electrical Power Distribution System through automatic transfer at the 6.9 kV Unit Boards.

The following offsite power configurations meet the requirements of the LCO:

1. Two offsite circuits consisting of a AND b (no board transfers required; a loss of either circuit will not prevent the minimum safety functions from being performed):
  - a. From the 161 kV transmission network, through CSST A (winding X) to Start Bus 1A to 6.9 kV Shutdown Board 1B-B (through 6.9 kV Unit Board 1C), and CSST A (winding Y) to Start Bus 2A to 6.9 kV Shutdown Board 2B-B (through 6.9 kV Unit Board 2C); AND

## BASES

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### LCO (continued)

- b. From the 161 kV transmission network, through CSST C (winding X) to Start Bus 2B to 6.9 kV Shutdown Board 2A-A (through 6.9 kV Unit Board 2B), and CSST C (winding Y) to Start Bus 1B to 6.9 kV Shutdown Board 1A-A (through 6.9 kV Unit Board 1B).
2. Two offsite circuits consisting of a AND b (relies on automatic transfer from alignment a.1) to b.2)(b), or a.2) to b.1)(a) on a loss of USSTs 1A and 1B, OR relies on automatic transfer from alignment a.3) to b.2)(a), or a.4) to b.1)(b) on a loss of USSTs 2A and 2B):
    - a. Normal power source alignments
      - 1) From the 500 kV switchyard through USST 1A to 6.9 kV Shutdown Board 1A-A (through 6.9 kV Unit Board 1B);
      - 2) From the 500 kV switchyard through USST 1B to 6.9 kV Shutdown Board 1B-B (through 6.9 kV Unit Board 1C);
      - 3) From the 161 kV switchyard through USST 2A to 6.9 kV Shutdown Board 2A-A (through 6.9 kV Unit Board 2B); AND
      - 4) From the 161 kV switchyard through USST 2B to 6.9 kV Shutdown Board 2B-B (through 6.9 kV Unit Board 2C).
    - b. Alternate power source alignments
      - 1) From the 161 kV transmission network, through:
        - (a) CSST A (winding X) to Start Bus 1A to 6.9 kV Shutdown Board 1B-B (through 6.9 kV Unit Board 1C); AND
        - (b) CSST A (winding Y) to Start Bus 2A to 6.9 kV Shutdown Board 2B-B (through 6.9 kV Unit Board 2C); OR
      - 2) From the 161 kV transmission network, through:
        - (a) CSST C (winding X) to Start Bus 2B to 6.9 kV Shutdown Board 2A-A (through 6.9 kV Unit Board 2B); AND
        - (b) CSST C (winding Y) to Start Bus 1B to 6.9 kV Shutdown Board 1A-A (through 6.9 kV Unit Board 1B).

## BASES

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### LCO (continued)

3. Two offsite circuits consisting of a AND b (relies on automatic transfer from alignment a.1) to b.1) and b.2) on a loss of the Unit 2 USSTs; a loss of alignment a.2) or a.3) will not prevent the minimum safety functions from being performed):
  - a. Normal power source alignments
    - 1) From the 161 kV switchyard through USST 2A to 6.9 kV Shutdown Board 2A-A (through 6.9 kV Unit Board 2B), and USST 2B to 6.9 kV Shutdown Board 2B-B (through 6.9 kV Unit Board 2C);
    - 2) From the 161 kV transmission network, through CSST A (winding X) to Start Bus 1A to 6.9 kV Shutdown Board 1B-B (through 6.9 kV Unit Board 1C); AND
    - 3) From the 161 kV transmission network, through CSST C (winding Y) to Start Bus 1B to 6.9 kV Shutdown Board 1A-A (through 6.9 kV Unit Board 1B).
  - b. Alternate power source alignments
    - 1) From the 161 kV transmission network, through CSST A (winding Y) to Start Bus 2A to 6.9 kV Shutdown Board 2B-B (through 6.9 kV Unit Board 2C); AND
    - 2) From the 161 kV transmission network, through CSST C (winding X) to Start Bus 2B to 6.9 kV Shutdown Board 2A-A (through 6.9 kV Unit Board 2B).
4. Two offsite circuits consisting of a AND b (relies on automatic transfer from alignment a.1) to b.1) and b.2) on a loss of the Unit 1 USSTs; a loss of alignment a.2) or a.3) will not prevent the minimum safety functions from being performed):
  - a. Normal power source alignments
    - 1) From the 500 kV switchyard through USST 1A to 6.9 kV Shutdown Board 1A-A (through 6.9 kV Unit Board 1B), and USST 1B to 6.9 kV Shutdown Board 1B-B (through 6.9 kV Unit Board 1C);
    - 2) From the 161 kV transmission network, through CSST A (winding Y) to Start Bus 2A to 6.9 kV Shutdown Board 2B-B (through 6.9 kV Unit Board 2C); AND

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### LCO (continued)

3) From the 161 kV transmission network, through CSST C (winding X) to Start Bus 2B to 6.9 kV Shutdown Board 2A-A (through 6.9 kV Unit Board 2B).

b. Alternate power source alignments

1) From the 161 kV transmission network, through CSST A (winding X) to Start Bus 1A to 6.9 kV Shutdown Board 1B-B (through 6.9 kV Unit Board 1C); AND

2) From the 161 kV transmission network, through CSST C (winding Y) to Start Bus 1B to 6.9 kV Shutdown Board 1A-A (through 6.9 kV Unit Board 1B).

Other offsite power configurations are possible using different combinations of available USSTs and CSSTs, as long as the alignments are consistent with the analyzed configurations, and the alignments comply with the requirements of GDC 17.

Each DG must be capable of starting, accelerating to rated speed and voltage, and connecting to its respective 6.9 kV Shutdown Board on detection of board 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 6.9 kV Shutdown Board. 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.

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.

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### APPLICABILITY

The AC sources 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

## BASES

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### APPLICABILITY (continued)

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

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

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### ACTIONS

The ACTIONS are modified by a Note that prohibits the application of LCO 3.0.4.b to an inoperable DG. There is an increased risk associated with entering a MODE or other specified condition in the Applicability with an inoperable DG and the provisions of LCO 3.0.4.b, which allow entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, should not be applied in this circumstance.

A second Note provides the appropriate restrictions for a de-energized shutdown board. 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 ACTIONS are modified by a Note to indicate that when any Condition(s) is entered with no AC power source to any shutdown board resulting in a de-energized shutdown board, the Conditions and Required Actions for LCO 3.8.9, "Distribution Systems - Operating," must be immediately entered. This allows LCO 3.8.1 Conditions to provide requirements for the loss of any combination of AC Sources, without regard to whether a shutdown board is de-energized and LCO 3.8.9 to provide the appropriate restrictions for a de-energized shutdown board.

#### A.1

To ensure a highly reliable power source remains with one offsite circuit inoperable for reasons other than Condition C, 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 D, for two offsite circuits inoperable, is entered.

#### A.2

Required Action A.2, which only applies if a Unit 1 6.9 kV Shutdown Board cannot be powered from an offsite source, is intended to provide assurance that an event coincident with a single failure of the associated

## BASES

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### ACTIONS (continued)

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. However, due to flow requirements of accident scenarios such as Feedwater Line Break (FWLB) and Small Break Loss of Coolant Accident (SBLOCA), the Turbine Driven Auxiliary Feedwater Pump should also be considered a required redundant feature.

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. 6.9 kV Shutdown Board 1A-A or 1B-B has no offsite power supplying it loads and
- b. A required feature on the other train is inoperable.

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

Discovering no offsite power to a Unit 1 6.9 kV Shutdown Board 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 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.

## BASES

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### ACTIONS (continued)

#### A.3

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

#### B.1

To ensure a highly reliable power source remains with one or more Train A DGs, or one or more Train B DGs inoperable, 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.

#### B.2

Required Action B.2 is intended to provide assurance that a loss of offsite power, during the period that a DG(s) 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. However, due to flow requirements of accident scenarios such as Feedwater Line Break (FWLB) and Small Break Loss of Coolant Accident (SBLOCA), the Turbine Driven Auxiliary Feedwater Pump should also be considered a required redundant feature. Redundant required feature failures consist of inoperable features associated with a train, redundant to the train that has an inoperable DG(s).

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"

## BASES

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### ACTIONS (continued)

for beginning the allowed outage time "clock." In this Required Action, the Completion Time only begins on discovery that both:

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

If at any time during the existence of this Condition (one or more DGs in a train inoperable) a required feature subsequently becomes inoperable, this Completion Time would begin to be tracked.

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

#### B.3.1 and B.3.2

Required Action B.3.1 provides an allowance to avoid unnecessary testing of OPERABLE DG(s). If it can be determined that the cause of the inoperable DG does not exist on the OPERABLE DG, SR 3.8.1.2 does not have to be performed. If the cause of inoperability exists on other DGs, the other DGs would be declared inoperable upon discovery and Condition F of LCO 3.8.1 would be entered if one or more DG(s) in Train A and Train B are inoperable. 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 DGs, performance of SR 3.8.1.2 suffices to provide assurance of continued OPERABILITY of that DG.

## BASES

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### ACTIONS (continued)

In the event the inoperable DG(s) is restored to OPERABLE status prior to completing either B.3.1 or B.3.2, the 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

In Condition B, the remaining OPERABLE DG(s) and offsite circuits are adequate to supply electrical power to the onsite Class 1E AC Electrical Power Distribution System. The 7 day 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.

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

Condition C is entered for an offsite circuit inoperable solely due to an inoperable power source to 6.9 kV Shutdown Board 2A-A or 2B-B. Required Action C.1 verifies the OPERABILITY of the remaining offsite circuit within an hour of the inoperability and every 8 hours thereafter. Since the Required Action only specifies "perform," a failure of the SR 3.8.1.1 acceptance criteria does not result in a Required Action not met.

The Completion Time for Required Action C.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. 6.9 kV Shutdown Board 2A-A or 2B-B has no offsite power; and
- b. A required feature on the other train is inoperable.

## BASES

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### ACTIONS (continued)

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

A Completion Time of 24 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 support these functions. The Completion Time takes into account the component OPERABILITY of the remaining redundant feature(s), a reasonable time for repairs, and the low probability of a DBA occurring during this period.

Operation may continue in Condition C for a period of 7 days. With one offsite circuit inoperable, the reliability of the functions is degraded. The potential for the loss of offsite power to the redundant feature(s) is increased, with the attendant potential for a challenge to their safety functions.

The required offsite circuit must be returned to OPERABLE status within 7 days. The 7 days Completion Time takes into account the capacity and capability of the remaining AC sources providing electrical power to the required feature(s), a reasonable time for repairs and the low probability of a DBA occurring during this period of time.

#### D.1 and D.2

Required Action D.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.

## BASES

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### ACTIONS (continued)

The Completion Time for Required Action D.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.

If at any time during the existence of Condition D (two offsite circuits inoperable) 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 D 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.

## BASES

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### ACTIONS (continued)

With both 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 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 or Condition C, as applicable.

#### E.1 and E.2

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

In Condition E, individual redundancy is lost in both the offsite electrical power system and the onsite AC electrical power system. Since power system redundancy is provided by two diverse sources of power, however, the reliability of the power systems in this Condition may appear higher than that in Condition D (loss of both 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.

## BASES

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### ACTIONS (continued)

#### F.1

With one or more Train A DG(s) and one or more Train B DG(s) inoperable, there are insufficient standby AC sources available to power an entire load group. Thus, with an assumed loss of offsite electrical power, insufficient standby AC sources are available to power the minimum required ESF functions. Since the offsite electrical power system is the only source of AC power for this level of degradation, the risk associated with continued operation for a very short time could be less than that associated with an immediate controlled shutdown (the immediate shutdown could cause grid instability, which could result in a total loss of AC power). Since any inadvertent generator trip could also result in a total loss of offsite AC power, however, the time allowed for continued operation is severely restricted. The intent here is to avoid the risk associated with an immediate controlled shutdown and to minimize the risk associated with this level of degradation.

In this Condition, operation may continue for a period that should not exceed 2 hours, consistent with the guidance provided in Reference 6.

#### G.1 and G.2

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

#### H.1 and I.1

Conditions H and I correspond to a level of degradation in which redundancy in the AC electrical power supplies cannot be assured. At this severely degraded level, any further losses in the AC electrical power system will cause a loss of function. Therefore, no additional time is justified for continued operation. The unit is required by LCO 3.0.3 to commence a controlled shutdown.

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### SURVEILLANCE REQUIREMENTS

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

Where the SRs discussed herein specify voltage and frequency tolerances, the following is applicable. The minimum steady state output voltage of 6210 V is 90% of the nominal 6900 V output voltage. This value, which is specified in ANSI C84.1 (Ref. 10), allows for voltage drop to the terminals of 6600 V motors whose minimum operating voltage is specified as 90% or 5940 V. It also allows for voltage drops to motors and other equipment down through the 120 V level where minimum operating voltage is also usually specified as 90% of name plate rating. The specified maximum steady state output voltage of 7260 V is equal to the maximum operating voltage specified for 6600 V motors. It ensures that for a lightly loaded distribution system, the voltage at the terminals of 6600 V motors is no more than the maximum rated operating voltages. The steady state minimum and maximum frequency values are 59.8 Hz and 60.2 Hz, which are consistent with the recommendations in Regulatory Guide 1.9 (Ref. 3). These values ensure that the safety related plant equipment powered from the DGs is capable of performing its safety functions.

#### SR 3.8.1.1

This SR ensures proper circuit continuity for the offsite AC electrical power supply to the onsite distribution network and availability of offsite AC electrical power. The breaker alignment verifies that each breaker is in its correct position to ensure that distribution buses and loads are connected to their preferred power source, and that appropriate independence of offsite circuits is maintained.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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### SURVEILLANCE REQUIREMENTS (continued)

#### 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 1 for SR 3.8.1.2 and Note for SR 3.8.1.7) 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.

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

In order to reduce stress and wear on diesel engines, the manufacturer recommends 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. These start procedures are the intent of Note 2.

SR 3.8.1.7 requires that 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 UFSAR, Chapter 15 (Ref. 5).

The 10 second start requirement is not applicable to SR 3.8.1.2 (see Note 2) when a modified start procedure as described above is used. During this testing, the diesel is not in an accident mode and the frequency is controlled by the operator instead of the governor's accident speed reference. If a modified start is not used, the 10 second start requirement of SR 3.8.1.7 applies.

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

In addition to the SR requirements, the time for the DG to reach steady state operation, unless the modified DG start method is employed, is periodically monitored and the trend evaluated to identify degradation of governor and voltage regulator performance.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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### SURVEILLANCE REQUIREMENTS (continued)

#### SR 3.8.1.3

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

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

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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 board loads, do not invalidate this test. Similarly, momentary power factor transients above the limit do 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 engine-mounted "day" 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 Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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### 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 engine-mounted “day” tank 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 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 Frequency is controlled under the Surveillance Frequency Control Program.

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 the storage system to the engine-mounted “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 Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

#### SR 3.8.1.7

See SR 3.8.1.2.

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.8

Transfer of the power supply to each 6.9 kV Unit Board from the normal supply to the alternate supply demonstrates the OPERABILITY of the alternate supply to power the shutdown loads. This SR is modified by two Notes.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

This SR is modified by two Notes. The reason for Note 1 is that, during operation with the reactor critical, performance of this SR for the 1A, 1B, 1C, and 1D Unit Boards could cause perturbations to the electrical distribution systems that could challenge continued steady state operation and, as a result, unit safety systems. This restriction from normally performing the Surveillance in MODE 1 or 2 is further amplified to allow the Surveillance to be performed for the purpose of reestablishing OPERABILITY (e.g., post work testing following corrective maintenance, corrective modification, deficient or incomplete surveillance testing, and other unanticipated OPERABILITY concerns) provided an assessment determines plant safety is maintained or enhanced. This assessment shall, as a minimum, consider the potential outcomes and transients associated with a failed Surveillance, a successful Surveillance, and a perturbation of the offsite or onsite system when they are tied together or operated independently for the Surveillance; as well as the operator procedures available to cope with these outcomes. These shall be measured against the avoided risk of a plant shutdown and startup to determine that plant safety is maintained or enhanced when the Surveillance is performed in MODE 1 or 2. Risk insights or deterministic methods may be used for this assessment. Credit may be taken for unplanned events that satisfy this SR.

Note 2 specifies that transfer capability is only required to be met for 6.9 kV Unit Boards that require normal and alternate power supplies. When both load groups are being supplied power by the USSTs, only the 6.9 kV Unit Boards associated with one load group are required to have normal and alternate power supplies. Therefore, only one CSST is required to be OPERABLE and available as an alternate power supply. Manual transfers between the normal supply and the alternate supply are also required to meet the SR. However, delayed access to an offsite circuit is not credited in the accident analysis.

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.9

Each DG is provided with an engine overspeed trip to prevent damage to the engine. Recovery from the transient caused by the loss of a large load could cause diesel engine overspeed, which, if excessive, might result in a trip of the engine. This Surveillance demonstrates the DG load response characteristics and capability to reject the largest single load (600 kW) without exceeding predetermined voltage and frequency and while maintaining a specified margin to the overspeed trip. 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 board, or
- b. Tripping its associated single largest post-accident load with the DG solely supplying the board.

Consistent with Regulatory Guide 1.9 (Ref. 3), 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 and voltage 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 maximum transient 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 Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

This SR is modified by a Note. The Note ensures that the DG is tested under load conditions that are as close to design basis conditions as possible. When synchronized with offsite power, testing should be performed at a power factor of  $\leq 0.89$ . This power factor is representative of the actual inductive loading a DG would see under design basis accident conditions. Under certain conditions, however, the Note allows the Surveillance to be conducted at a power factor other than  $\leq 0.89$ . These conditions occur when grid voltage is high, and the additional field excitation needed to get the power factor to  $\leq 0.89$  results in voltages on the emergency boards that are too high.

## BASES

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### SURVEILLANCE REQUIREMENTS (continued)

Under these conditions, the power factor should be maintained as close as practicable to 0.89 while still maintaining acceptable voltage limits on the emergency boards. In other circumstances, the grid voltage may be such that the DG excitation levels needed to obtain a power factor of 0.89 may not cause unacceptable voltages on the emergency boards, but the excitation levels are in excess of those recommended for the DG. In such cases, the power factor shall be maintained as close as practicable to 0.89 without exceeding the DG excitation limits.

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

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

This SR has been modified by a Note. The Note ensures that the DG is tested under load conditions that are as close to design basis conditions as possible. When synchronized with offsite power, testing should be performed at a power factor of  $\leq 0.89$ . This power factor is representative of the actual inductive loading a DG would see under design basis accident conditions. Under certain conditions, however, the Note allows the Surveillance to be conducted at a power factor other than  $\leq 0.89$ . These conditions occur when grid voltage is high, and the additional field excitation needed to get the power factor to  $\leq 0.89$  results in voltages on the emergency boards that are too high. Under these conditions, the power factor should be maintained as close as practicable to 0.89 while still maintaining acceptable voltage limits on the emergency boards. In other circumstances, the grid voltage may be such that the DG excitation levels needed to obtain a power factor of 0.89 may not cause unacceptable voltages on the emergency boards, but the excitation levels are in excess of those recommended for the DG. In such cases, the power factor shall be maintained as close as practicable to 0.89 without exceeding the DG excitation limits.

BASES

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.11

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

The DG autostart 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 is achieved.

The requirement to verify the connection and power supply 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, Emergency Core Cooling Systems (ECCS) injection valves are not desired to be stroked open, or high pressure injection systems are not capable of being operated at full flow, or residual heat removal (RHR) 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 systems 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 Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

This SR is modified by two Notes. The reason for Note 1 is to minimize wear and tear on the DGs during testing. For the purpose of this testing, the DGs must be started from standby conditions, that is, with the engine coolant and oil continuously circulated and temperature maintained consistent with manufacturer recommendations. The reason for Note 2 is that performing the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems. This restriction from normally performing the Surveillance for DGs 1A-A and 1B-B in MODE 1 or 2 is further amplified to allow portions of the Surveillance to be performed for the purpose of

## BASES

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### SURVEILLANCE REQUIREMENTS (continued)

reestablishing OPERABILITY (e.g., post work testing following corrective maintenance, corrective modification, deficient or incomplete surveillance testing, and other unanticipated OPERABILITY concerns) provided an assessment determines plant safety is maintained or enhanced. This assessment shall, as a minimum, consider the potential outcomes and transients associated with a failed partial Surveillance, a successful partial Surveillance, and a perturbation of the offsite or onsite system when they are tied together or operated independently for the partial Surveillance; as well as the operator procedures available to cope with these outcomes. These shall be measured against the avoided risk of a plant shutdown and startup to determine that plant safety is maintained or enhanced when portions of the Surveillance are performed in MODE 1 or 2. Risk insights or deterministic methods may be used for the assessment. Credit may be taken for unplanned events that satisfy this SR.

#### SR 3.8.1.12

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

The requirement to verify the connection of permanent and 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, or high pressure injection systems are not capable of being operated at full flow, or RHR 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 Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

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## SURVEILLANCE REQUIREMENTS (continued)

This SR is modified by two Notes. The reason for Note 1 is to minimize wear and tear on the DGs during testing. For the purpose of this testing, the DGs must be started from standby conditions, that is, with the engine coolant and oil continuously circulated and temperature maintained consistent with manufacturer recommendations. The reason for Note 2 is that 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. This restriction from normally performing the Surveillance for DGs 1A-A and 1B-B in MODE 1 or 2 is further amplified to allow portions of the Surveillance to be performed for the purpose of reestablishing OPERABILITY (e.g., post work testing following corrective maintenance, corrective modification, deficient or incomplete surveillance testing, and other unanticipated OPERABILITY concerns) provided an assessment determines plant safety is maintained or enhanced. This assessment shall, as a minimum, consider the potential outcomes and transients associated with a failed partial Surveillance, a successful partial Surveillance, and a perturbation of the offsite or onsite system when they are tied together or operated independently for the partial Surveillance; as well as the operator procedures available to cope with these outcomes. These shall be measured against the avoided risk of a plant shutdown and startup to determine that plant safety is maintained or enhanced when portions of the Surveillance are performed in MODE 1 or 2. Risk insights or deterministic methods may be used for the assessment. 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, an ESF actuation test signal, or both. Noncritical automatic trips are all automatic trips except:

- a. Engine overspeed; and
- b. Generator differential current.

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.

## BASES

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### SURVEILLANCE REQUIREMENTS (continued)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

The SR is modified by a Note. The reason for the Note is that performing the Surveillance would remove a required DG from service. This restriction from normally performing the Surveillance for DGs 1A-A and 1B-B in MODE 1 or 2 is further amplified to allow the Surveillance to be performed for the purpose of reestablishing OPERABILITY (e.g., post work testing following corrective maintenance, corrective modification, deficient or incomplete surveillance testing, and other unanticipated OPERABILITY concerns) provided an assessment determines plant safety is maintained or enhanced. This assessment shall, as a minimum, consider the potential outcomes and transients associated with a failed Surveillance, a successful Surveillance, and a perturbation of the offsite or onsite system when they are tied together or operated independently for the Surveillance; as well as the operator procedures available to cope with these outcomes. These shall be measured against the avoided risk of a plant shutdown and startup to determine that plant safety is maintained or enhanced when the Surveillance is performed in MODE 1 or 2. Risk insights or deterministic methods may be used for this assessment. Credit may be taken for unplanned events that satisfy this SR.

#### SR 3.8.1.14

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

The load band is provided to avoid routine overloading of the DG. Routine overloading may result in more frequent teardown inspections in accordance with vendor recommendations in order to maintain DG OPERABILITY.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

## BASES

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### SURVEILLANCE REQUIREMENTS (continued)

This Surveillance is modified by two Notes. Note 1 states that momentary transients due to changing board loads do not invalidate this test. Similarly, momentary power factor transients above the power factor limit will not invalidate the test. Note 2 ensures that the DG is tested under load conditions that are as close to design basis conditions as possible. When synchronized with offsite power, testing should be performed at a power factor of  $\leq 0.89$ . This power factor is representative of the actual inductive loading a DG would see under design basis accident conditions. Under certain conditions, however, Note 2 allows the Surveillance to be conducted at a power factor other than  $\leq 0.89$ . These conditions occur when grid voltage is high, and the additional field excitation needed to get the power factor to  $\leq 0.89$  results in voltages on the emergency boards that are too high. Under these conditions, the power factor should be maintained as close as practicable to 0.89 while still maintaining acceptable voltage limits on the emergency boards. In other circumstances, the grid voltage may be such that the DG excitation levels needed to obtain a power factor of 0.89 may not cause unacceptable voltages on the emergency boards, but the excitation levels are in excess of those recommended for the DG. In such cases, the power factor shall be maintained close as practicable to 0.89 without exceeding the DG excitation limits.

#### 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 Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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 board 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.

## BASES

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### 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 the DG can be returned to ready to load status when offsite power is restored. It also ensures that the autostart 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 an autoclose signal on board undervoltage, and the load sequence timers are reset.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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. This restriction from normally performing the Surveillance for DGs 1A-A and 1B-B in MODE 1 or 2 is further amplified to allow the Surveillance to be performed for the purpose of reestablishing OPERABILITY (e.g., post work testing following corrective maintenance, corrective modification, deficient or incomplete surveillance testing, and other unanticipated OPERABILITY concerns) provided an assessment determines plant safety is maintained or enhanced. This assessment shall, as a minimum, consider the potential outcomes and transients associated with a failed Surveillance, a successful Surveillance, and a perturbation of the offsite or onsite system when they are tied together or operated independently for the Surveillance; as well as the operator procedures available to cope with these outcomes. These shall be measured against the avoided risk of a plant shutdown and startup to determine that plant safety is maintained or enhanced when the Surveillance is performed in MODE 1 or 2. Risk insights or deterministic methods may be used for this assessment. Credit may be taken for unplanned events that satisfy this SR.

BASES

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.17

Under accident and loss of offsite power conditions loads are sequentially connected to the board by the load sequence timers. 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 5% load sequence time interval tolerance ensures that sufficient time exists for the DG to restore frequency and voltage prior to applying the next load and that safety analysis assumptions regarding ESF equipment time delays are not violated. Reference 2 provides a summary of the automatic loading of Shutdown Boards.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.8.1.18

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 Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

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SURVEILLANCE REQUIREMENTS (continued)

This SR is modified by two Notes. The reason for Note 1 is to minimize wear and tear on the DGs during testing. For the purpose of this testing, the DGs must be started from standby conditions, that is, with the engine coolant and oil continuously circulated and temperature maintained consistent with manufacturer recommendations for DGs. The reason for Note 2 is that the performance of the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems. This restriction from normally performing the Surveillance for DGs 1A-A and 1B-B in MODE 1 or 2 is further amplified to allow portions of the Surveillance to be performed for the purpose of reestablishing OPERABILITY (e.g., post work testing following corrective maintenance, corrective modification, deficient or incomplete surveillance testing, and other unanticipated OPERABILITY concerns) provided an assessment determines plant safety is maintained or enhanced. This assessment shall, as a minimum, consider the potential outcomes and transients associated with a failed partial Surveillance, a successful partial Surveillance, and a perturbation of the offsite or on-site system when they are tied together or operated independently for the partial Surveillance; as well as the operator procedures available to cope with these outcomes. These shall be measured against the avoided risk of a plant shutdown and startup to determine that plant safety is maintained or enhanced when portions of the Surveillance are performed in MODE 1 or 2. Risk insights or deterministic methods may be used for the assessment. Credit may be taken for unplanned events that satisfy this SR.

SR 3.8.1.19

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 Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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.

BASES

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- REFERENCES
1. 10 CFR 50, Appendix A, GDC 17.
  2. UFSAR, Chapter 8.
  3. Regulatory Guide 1.9, Rev. 0.
  4. UFSAR, Chapter 6.
  5. UFSAR, Chapter 15.
  6. Regulatory Guide 1.93, Rev. 0, December 1974.
  7. Generic Letter 84-15, "Proposed Staff Actions to Improve and Maintain Diesel Generator Reliability," July 2, 1984.
  8. 10 CFR 50, Appendix A, GDC 18.
  9. Regulatory Guide 1.108, Rev. 1, August 1977.
  10. ANSI C84.1, Voltage Ratings for Electric Power Systems and Equipment (60 Hz).
  11. UFSAR Chapter 10.
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## B 3.8 ELECTRICAL POWER SYSTEMS

### B 3.8.2 AC Sources - Shutdown

#### BASES

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BACKGROUND	A description of the AC sources is provided in the Bases for LCO 3.8.1, "AC Sources - Operating."
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APPLICABLE SAFETY ANALYSES	<p>The OPERABILITY of the minimum AC sources during MODES 5 and 6 and during movement of irradiated fuel assemblies ensures that:</p> <ol style="list-style-type: none"><li>The unit can be maintained in the shutdown or refueling condition for extended periods,</li><li>Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status, and</li><li>Adequate AC electrical power is provided to mitigate events postulated during shutdown, such as a fuel handling accident.</li></ol>
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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. This is because many Design Basis Accidents (DBAs) that are analyzed in MODES 1, 2, 3, and 4 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 result in minimal consequences. These limitations during shutdown conditions are reflected in the LCO for required systems.

During MODES 1, 2, 3, and 4, various deviations from the analysis assumptions and design requirements are allowed within the Required Actions. This allowance is in 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 planned and administratively controlled. Relaxations from MODE 1, 2, 3, and 4 LCO requirements are acceptable during shutdown modes based on:

BASES

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APPLICABLE SAFETY ANALYSES (continued)

- 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 10 CFR 50.36(c)(2)(ii).

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LCO

One offsite circuit capable of supplying the onsite Class 1E power distribution subsystem(s) of LCO 3.8.10, "Distribution Systems - Shutdown," ensures that all required loads are powered from offsite power. Two OPERABLE DGs, associated with a distribution system train required to be OPERABLE by LCO 3.8.10, ensures a diverse power source is available to provide electrical power support, assuming a loss of the offsite circuit. Together, OPERABILITY of the required offsite circuit and DGs 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 6.9 kV shutdown boards. Qualified offsite circuits are those that are described in the Bases for LCO 3.8.1 and are part of the licensing basis for the unit.

Each required offsite circuit is that combination of power sources described in the Bases of LCO 3.8.1.

## BASES

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### LCO (continued)

The DGs must be capable of starting, accelerating to rated speed and voltage, and connecting to their respective 6.9 kV shutdown board on detection of board undervoltage. This sequence must be accomplished within 10 seconds. The DG must be capable of accepting required loads within the assumed loading sequence intervals, and continue to operate until offsite power can be restored to the 6.9 kV shutdown boards. 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.

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### APPLICABILITY

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

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

The AC power requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.1.

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### ACTIONS

LCO 3.0.3 is not applicable while in MODE 5 or 6. However, since irradiated fuel assembly movement can occur in MODE 1, 2, 3, or 4, the ACTIONS have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operations. Entering LCO 3.0.3, while in MODE 1, 2, 3, or 4 would require the unit to be shutdown unnecessarily.

#### A.1

An offsite circuit would be considered inoperable if it were not available to one required ESF train. Although two trains are required by LCO 3.8.10, the one train with offsite power available may be capable of supporting sufficient required features to allow continuation of irradiated fuel

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## BASES

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### ACTIONS (continued)

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.

#### 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 one or more required DGs inoperable, the minimum required diversity of AC power sources is not available. It is, therefore, required to suspend movement of irradiated fuel assemblies, CORE ALTERATIONS, and operations involving positive reactivity additions that could result in loss of required SDM (MODE 5) or boron concentration (MODE 6). Suspending positive reactivity additions that could result in failure to meet the minimum SDM or boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than that what would be required in the RCS for minimum SDM or refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation. Introduction of temperature changes including temperature increases when operating with a positive MTC must also be evaluated to ensure they do not result in a loss of required SDM.

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 would not be 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 6.9 kV shutdown board, the ACTIONS for

BASES

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ACTIONS (continued)

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 would provide the appropriate restrictions for the situation involving a de-energized train.

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SURVEILLANCE  
REQUIREMENTS

SR 3.8.2.1

SR 3.8.2.1 requires the SRs from LCO 3.8.1 that are necessary for ensuring the OPERABILITY of the AC sources in other than MODES 1, 2, 3, and 4. SR 3.8.1.8 is not required to be met since only one offsite circuit is required to be OPERABLE. SR 3.8.1.12 and SR 3.8.1.18 are not required to be met because the ESF actuation signal is not required to be OPERABLE. SR 3.8.1.19 is excepted because starting independence is not required with the DG(s) that is not required to be OPERABLE.

This SR is modified by a Note. The reason for the Note is to preclude requiring the OPERABLE DG(s) from being paralleled with the offsite power network or otherwise rendered inoperable during performance of SRs, and to preclude deenergizing a required 6.9 kV shutdown board 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 DGs. It is the intent that these SRs must still be capable of being met, but actual performance is not required during periods when the DGs and offsite circuit is required to be OPERABLE. Refer to the corresponding Bases for LCO 3.8.1 for a discussion of each SR.

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REFERENCES

None.

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## B 3.8 ELECTRICAL POWER SYSTEMS

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

#### BASES

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**BACKGROUND** Each diesel generator (DG) is provided with a 7-day storage tank, comprised of four inter-connected tanks, having a fuel oil capacity sufficient to operate that diesel for a period of 7 days while the DG is supplying maximum post loss of coolant accident load demand discussed in the UFSAR, Section 9.5.4.3 (Ref. 1) and Regulatory Guide 1.137 (Ref. 2). The maximum load demand assumed is the diesel generator operating at 110% rated load for the first two hours then operating at 100% rated load for the remaining 166 hours. 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 each diesel generator 7-day storage tank to two engine-mounted "day" tanks by either of two transfer pumps associated with the 7-day 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. Each 7-day tank is embedded in the concrete substructure below its respective DG with the two 550-gallon engine-mounted "day" tanks located in the respective DG room.

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

The DG lubrication system is designed to provide sufficient lubrication to permit proper operation of its associated DG under all loading conditions. The system is required to circulate the lube oil to the diesel engine working surfaces and to remove excess heat generated by friction during operation. Each engine oil sump contains an inventory capable of supporting a minimum of 7 days of operation without requiring replenishment. This supply is sufficient to allow the operator to replenish lube oil from outside sources.

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

BASES

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BACKGROUND (continued)

The Starting Air System for each diesel engine includes two tanks. The tanks are aligned in series to the starter motors. The upstream tank (Tank A) is maintained at 250-300 psig by the air compressor while the downstream tank (Tank B) is maintained at approximately 195 psig via a pressure control valve (PCV) between the two tanks.

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APPLICABLE  
SAFETY  
ANALYSES

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

Since diesel fuel oil, lube oil, and the air start subsystem support the operation of the standby AC power sources, they satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

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LCO

Stored diesel fuel oil is required to have sufficient supply for 7 days of full load operation. It is also required to meet specific standards for quality. Additionally, sufficient 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 engine-mounted "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 tank(s).

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APPLICABILITY

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

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BASES

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ACTIONS

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

A.1

In this Condition, the 7 day fuel oil supply for a DG is not available. However, the Condition is restricted to fuel oil level reductions that maintain at least a 6 day supply. The fuel oil level equivalent to a 6 day supply is 53719 gallons. 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

In this Condition, the 7 day lube oil inventory i.e., sufficient lubricating oil to support 7 days of continuous DG operation at full load conditions is not available. However, the Condition is restricted to lube oil volume reductions that maintain at least a 6 day supply. The lube oil inventory equivalent to a 6 day supply is 120 gallons (per diesel engine). This restriction allows sufficient time to obtain the requisite replacement volume. A period of 48 hours is considered sufficient to complete 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 3.8.3.3 for the stored fuel. Normally, trending of particulate levels allows sufficient time to correct high particulate levels prior to reaching the limit of acceptability. Poor sample procedures (bottom

## BASES

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### ACTIONS (continued)

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

#### E.1

With starting air Tank A pressure < 200 psig, sufficient capacity for five successive DG start attempts does not exist. However, as long as Tank B pressure is > 150 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 DG's 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.

BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.8.3.1

This SR provides verification that there is an adequate inventory of fuel oil in the storage tanks to support each DG's operation for 7 days at full load. The fuel oil level equivalent to a 7 day supply is 62,000 gallons when calculated in accordance with References 2 and 3. The required fuel storage volume is determined using the most limiting energy content of the stored fuel. Using the known correlation of diesel fuel oil absolute specific gravity or API gravity to energy content, the required diesel generator output, the corresponding fuel consumption rate, the onsite fuel storage volume required for 7 days of operation can be determined. SR 3.8.3.3 requires a new fuel to be tested to verify that the absolute specific gravity or API gravity is within the range assumed in the diesel fuel oil consumption calculations. 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 Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.8.3.2

This Surveillance ensures that sufficient lube oil inventory is available to support at least 7 days of full load operation for each DG. The lube oil inventory equivalent to a 7 day supply is 142 gallons (per diesel engine) and is based on the DG manufacturer consumption values for the run time of the DG.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

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SURVEILLANCE REQUIREMENTS (continued)

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 7-day storage tanks without concern for contaminating the entire volume of fuel oil in the 7-day storage tanks. These tests are to be conducted prior to adding the new fuel to the 7-day 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 D4057-1988 (Ref. 6),
- b. Verify in accordance with the tests specified in ASTM D975-1990 (Ref. 6) that the sample has an absolute specific gravity at 60/60°F of  $\geq 0.83$  and  $\leq 0.89$  or an API gravity at 60°F of  $\geq 27^\circ$  and  $\leq 39^\circ$  when tested in accordance with ASTM D1298-1985 (Ref. 6), a kinematic viscosity at 40°C of  $\geq 1.9$  centistokes and  $\leq 4.1$  centistokes, and a flash point of  $\geq 125^\circ\text{F}$ , and
- c. Verify that the new fuel oil has a clear and bright appearance with proper color when tested in accordance with ASTM D4176-2004 or a water and sediment content within limits when tested in accordance with ASTM D1796-1997 (Ref. 6).

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 7-day 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 D975-1990 (Ref. 7) are met for new fuel oil when tested in accordance with ASTM D975-1990 (Ref. 6), except that the analysis for sulfur may be performed in accordance with ASTM D1552-1990, ASTM D2622-1987, or ASTM D4294-2002 (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

## BASES

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### SURVEILLANCE REQUIREMENTS (continued)

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 D6217-11 (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. Each fuel oil storage tank (7-day 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.

#### 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. The pressure specified in this SR is intended to reflect the lowest value at which the five starts can be accomplished.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

#### 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 (7-day tanks) 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 Frequency is controlled under the Surveillance Frequency Control Program.

BASES

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REFERENCES

1. UFSAR, Section 9.5.4.3.
  2. Regulatory Guide 1.137, 1979.
  3. ANSI N195, 1976.
  4. UFSAR, Chapter 6.
  5. UFSAR, Chapter 15.
  6. ASTM Standards: D4057-1988; D975-1990; D1298-1985;  
D4176-2004; D1796-1997; D1552-1990; D2622-1987; D4294-2002;  
D6217-11.
  7. ASTM Standards, D975-1990, Table 1.
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## B 3.8 ELECTRICAL POWER SYSTEMS

### B 3.8.4 DC Sources - Operating

#### BASES

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**BACKGROUND** The DC electrical power system consists of the 125 V Vital DC electrical power subsystem and the diesel generator (DG) DC electrical power subsystem. The 125 V Vital DC electrical power subsystem 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 instrument power board power (via inverters). Control power and generator field flashing for each DG is provided by the DG DC electrical power subsystem. 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 V Vital DC electrical power subsystem consists of two independent and redundant safety related Class 1E DC electrical power trains (Train A is associated with channels I and III and Train B is associated with channels II and IV). Each train consists of two 125 VDC batteries, the associated battery charger(s) for each battery, and all the associated control equipment and interconnecting cabling.

The 125 V Vital DC electrical power subsystem has manual access to a fifth vital battery system. The fifth 125 VDC Vital Battery System is intended to serve as a replacement for any one of the four 125 VDC vital batteries during testing, maintenance, and outages with no loss of system reliability under any mode of operation. Additionally there is one spare battery charger for channels I and II and one spare charger for channels III and IV, 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 trains are maintained.

During normal operation, the 125 V Vital DC 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 Vital DC electrical power trains provide the control power for its associated Class 1E AC power load group, 6.9 kV Shutdown Boards, and 480 V load centers. The DC electrical power trains also provide DC electrical power to the inverters, which in turn power the AC vital instrument power boards.

## BASES

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### BACKGROUND (continued)

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

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

Each Vital battery has adequate storage capacity to meet the duty cycle(s) discussed in the UFSAR, Chapter 8 (Ref 4). The battery is designed with additional capacity above that required by the design duty cycle to allow for temperature variations and other factors.

The batteries for 125 V Vital DC electrical power trains are sized to produce required capacity at 82% of nameplate rating, corresponding to warranted capacity at end of life cycles and the 100% design demand. The minimum design voltage limit is 105 V.

The Vital battery cells are of flooded lead acid construction with a nominal specific gravity of 1.215. This specific gravity corresponds to an open circuit battery voltage of 123.78 V for a 60 cell battery (i.e., cell voltage of 2.063 volts per cell (Vpc)). The open circuit voltage is the voltage maintained when there is no charging or discharging. Optimal long term performance however, is obtained by maintaining a minimum float voltage of 129 V DC. This provides adequate over-potential, which limits the formation of lead sulfate and self discharge.

Each Vital DC electrical power train battery charger 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 excess capacity to restore the battery from the design minimum charge to its fully charged state within 12 hours (with accident loads being supplied) following a 30 minute AC power outage and in approximately 36 hours (with normal loads being supplied) following a 4 hour AC power outage.

Each vital battery charger is normally in the float-charge mode. Float-charge is the condition in which the charger is supplying the connected loads and the battery cells are receiving adequate current to optimally charge the battery. This assures the internal losses of a battery are overcome and the battery is maintained in a fully charged state.

BASES

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BACKGROUND (continued)

When desired, each Vital battery charger can be placed in the equalize mode. The equalize mode is at a higher voltage than the float mode and charging current is correspondingly higher. The battery charger is operated in the equalize mode after a battery discharge or for routine maintenance. Following a battery discharge, the battery recharge characteristic accepts current at the current limit of the battery charger (if the discharge was significant, e.g., following a battery service test) until the battery terminal voltage approaches the charger voltage setpoint. Charging current then reduces exponentially during the remainder of the recharge cycle. Lead-calcium batteries have recharge efficiencies of greater than 95%, so once at least 105% of the ampere-hours discharged have been returned, the battery capacity would be restored to the same condition as it was prior to the discharge. This can be monitored by direct observation of the exponentially decaying charging current or by evaluating the amp-hours discharged from the battery and amp-hours returned to the battery.

Control power for the DGs is provided by the four DG battery systems, one per DG. Each system is comprised of one 125 VDC battery, the associated charger for each battery, and all associated control equipment and interconnecting cabling.

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APPLICABLE  
SAFETY  
ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the UFSAR, Chapter 6 (Ref. 5) and Chapter 15 (Ref. 6), assume that Engineered Safety Feature (ESF) systems are OPERABLE. The DC electrical power system provides normal and emergency DC electrical power for the DGs, emergency auxiliaries, and control and switching during all MODES of operation.

The OPERABILITY of the DC sources is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the 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 10 CFR 50.36(c)(2)(ii).

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LCO

The Vital DC electrical power trains, each train consisting of two batteries, battery charger for each battery and the corresponding control equipment and interconnecting cabling supplying power to the associated board 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

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## BASES

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### LCO (continued)

postulated DBA. Loss of any DC electrical power train does not prevent the minimum safety function from being performed (Ref. 4).

The DG DC electrical power subsystems, each subsystem consisting of one battery, one battery charger and the corresponding control equipment and interconnecting cabling supplying power to the associated DG control circuit are required to be OPERABLE to ensure the availability of the required power to support the OPERABILITY of the diesel generator.

An OPERABLE Vital DC electrical power train requires all required batteries and respective chargers to be operating and connected to the associated DC board(s).

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### APPLICABILITY

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

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

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

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### ACTIONS

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

Condition A represents one train with one or two vital battery chargers inoperable (e.g., the voltage limit of SR 3.8.4.1 is not maintained). The ACTIONS provide a tiered response that focuses on returning the battery to the fully charged state and restoring a fully qualified charger to OPERABLE status in a reasonable time period. Required Action A.1 requires that the battery terminal voltage be restored to greater than or equal to the minimum established float voltage within 2 hours. This time provides for returning the inoperable charger to OPERABLE status or providing an alternate means of restoring battery terminal voltage to greater than or equal to the minimum established float voltage. Restoring the battery terminal voltage to greater than or equal to the minimum established float voltage provides good assurance that, within 12 hours, the battery will be restored to its fully charged condition (Required Action A.2) from any discharge that might have occurred due to the charger inoperability.

A discharged battery having terminal voltage of at least the minimum established float voltage indicates that the battery is on the exponential charging current portion (the second part) of its recharge cycle. The time

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## BASES

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### ACTIONS (continued)

to return a battery to its fully charged state under this condition is simply a function of the amount of the previous discharge and the recharge characteristic of the battery. Thus there is good assurance of fully recharging the battery within 12 hours, avoiding a premature shutdown with its own attendant risk.

If established battery terminal float voltage cannot be restored to greater than or equal to the minimum established float voltage within 2 hours, and the charger is not operating in the current-limiting mode, a faulty charger is indicated. A faulty charger that is incapable of maintaining established battery terminal float voltage does not provide assurance that it can revert to and operate properly in the current limit mode that is necessary during the recovery period following a battery discharge event that the DC system is designed for.

If the charger is operating in the current limit mode after 2 hours that is an indication that the battery is partially discharged and its capacity margins will be reduced. The time to return the battery to its fully charged condition in this case is a function of the battery charger capacity, the amount of loads on the associated DC system, the amount of the previous discharge, and the recharge characteristic of the battery. The charge time can be extensive, and there is not adequate assurance that it can be recharged within 12 hours (Required Action A.2).

Required Action A.2 requires that the battery float current be verified as less than or equal to 2 amps. This indicates that, if the battery had been discharged as the result of the inoperable battery charger, it is now fully capable of supplying the maximum expected load requirement. The 2 amp value is based on returning the battery to 98% charge and assumes a 5% design margin for the battery. If at the expiration of the initial 12 hour period the battery float current is not less than or equal to 2 amps this indicates there may be additional battery problems and the battery must be declared inoperable.

Required Action A.3 limits the restoration time for the inoperable vital battery charger to 7 days. This action is applicable if an alternate means of restoring battery terminal voltage to greater than or equal to the minimum established float voltage has been used (e.g., fifth battery charger). The 7 day Completion Time reflects a reasonable time to effect restoration of the qualified battery charger to OPERABLE status.

## BASES

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### ACTIONS (continued)

#### B.1

Condition B represents one vital DC 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 vital DC distribution subsystem.

If one of the required vital DC electrical power trains is inoperable for reasons other than Condition A (e.g., inoperable battery charger), the remaining vital DC electrical power train has the capacity to support a safe shutdown and to mitigate an accident condition. Since a subsequent worst case single failure could, however, result in the loss of the minimum necessary vital DC electrical trains to mitigate a worst case accident, continued power operation should not exceed 2 hours. The 2 hour Completion Time is based on Regulatory Guide 1.93 (Ref. 7) and reflects a reasonable time to assess unit status as a function of the inoperable vital DC electrical power train and, if the vital DC electrical power train is not restored to OPERABLE status, to prepare to effect an orderly and safe unit shutdown.

#### C.1 and C.2

If the inoperable vital DC electrical power train cannot be restored to OPERABLE status within the required Completion Time, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging plant systems. The Completion Time to bring the unit to MODE 5 is consistent with the time required in Regulatory Guide 1.93 (Ref. 7).

#### D.1

If the DG DC electrical power subsystem(s) is inoperable, the associated DG(s) may be incapable of performing their intended function and must be immediately declared inoperable. This declaration also requires entry into applicable Conditions and Required Actions for inoperable DG(s), LCO 3.8.1, "AC Sources – Operating."

## BASES

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### SURVEILLANCE REQUIREMENTS

#### SR 3.8.4.1

Verifying battery terminal voltage while on float charge for the batteries helps to ensure the effectiveness of the battery chargers, which support 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 and maintain the battery in a fully charged state while supplying the continuous steady state loads of the associated DC train or subsystem. On float charge, battery cells will receive adequate current to optimally charge the battery. The voltage requirements are based on the nominal design voltage of the battery and are consistent with the minimum float voltage established by the battery manufacturer (129 V for the Vital batteries and 124 V for the DG batteries). This voltage maintains the battery plates in a condition that supports maintaining the grid life.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

#### SR 3.8.4.2

This SR verifies the design capacity of the vital battery chargers. According to Regulatory Guide 1.32 (Ref. 8), the battery charger supply is recommended to be based on the largest combined demands of the various steady state loads and the charging capacity to restore the battery from the design minimum charge state to the fully charged state, irrespective of the status of the unit during these demand occurrences. The minimum required amperes and duration ensures that these requirements can be satisfied.

This SR provides two options. One option requires that each battery charger be capable of supplying 150 amps at the minimum established float voltage (129 V DC) for 4 hours. The ampere requirements are based on the output rating of the chargers. The voltage requirements are based on the charger voltage level after a response to a loss of AC power.

The other option requires that each vital battery charger be capable of recharging the battery after a service test coincident with supplying the largest coincident demands of the various continuous steady state loads (irrespective of the status of the plant during which these demands occur). This level of loading may not normally be available following the battery service test and will need to be supplemented with additional loads. The duration for this test may be longer than the charger sizing criteria since the battery recharge is affected by float voltage,

## BASES

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### SURVEILLANCE REQUIREMENTS (continued)

temperature, and the exponential decay in charging current. The battery is recharged when the measured charging current is  $\leq 2$  amps.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

#### SR 3.8.4.3

A battery service test is a special test of the battery capability, as found, to satisfy the design requirements (battery duty cycle (2 hours for Vital batteries)) 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 is controlled under the Surveillance Frequency Control Program.

This SR is modified by three Notes. Notes 1 and 2 allow the performance of a modified performance discharge test in lieu of a service test.

The reason for Note 3 is that performing the Surveillance on in-service vital batteries would perturb the electrical distribution system and challenge safety systems. Therefore, prior to performing a battery service test, the in-service vital battery to be tested is removed from service and the spare vital battery is placed in-service. Credit may be taken for unplanned events that satisfy this SR.

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|------------|---|
| REFERENCES | 1. 10 CFR 50, Appendix A, GDC 17.         |
|            | 2. Regulatory Guide 1.6, March 10, 1971.  |
|            | 3. IEEE-308-1971.                         |
|            | 4. UFSAR, Chapter 8.                      |
|            | 5. UFSAR, Chapter 6.                      |
|            | 6. UFSAR, Chapter 15.                     |
|            | 7. Regulatory Guide 1.93, December 1974.  |
|            | 8. Regulatory Guide 1.32, February 1972.  |
|            | 9. Regulatory Guide 1.129, December 1974. |
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## B 3.8 ELECTRICAL POWER SYSTEMS

### B 3.8.5 DC Sources - Shutdown

#### BASES

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BACKGROUND	A description of the DC sources is provided in the Bases for LCO 3.8.4, "DC Sources - Operating."
APPLICABLE SAFETY ANALYSES	<p>The initial conditions of Design Basis Accident (DBA) and transient analyses in the UFSAR, Chapter 6 (Ref. 1) and Chapter 15 (Ref. 2), assume that Engineered Safety Feature systems are OPERABLE. The DC electrical power system provides normal and emergency DC electrical power for the diesel generators, emergency auxiliaries, and control and switching during all MODES of operation.</p> <p>The OPERABILITY of the DC subsystems is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY.</p> <p>The OPERABILITY of the minimum DC electrical power sources during MODES 5 and 6 and during movement of irradiated fuel assemblies ensures that:</p> <ol style="list-style-type: none"><li>The unit can be maintained in the shutdown or refueling condition for extended periods,</li><li>Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status, and</li><li>Adequate DC electrical power is provided to mitigate events postulated during shutdown, such as a fuel handling accident.</li></ol> <p>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 DBAs that are analyzed in MODES 1, 2, 3, and 4 have no specific analyses 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.</p>

BASES

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APPLICABLE SAFETY ANALYSES (continued)

The shutdown Technical Specification requirements are designed to ensure that the unit has the capability to mitigate the consequences of certain postulated accidents. Worst case DBAs which are analyzed for operating MODES are generally viewed not to be a significant concern during shutdown MODES due to the lower energies involved. The Technical Specifications therefore require a lesser complement of electrical equipment to be available during shutdown than is required during operating MODES. More recent work completed on the potential risks associated with shutdown, however, have found significant risk associated with certain shutdown evolutions. As a result, in addition to the requirements established in the Technical Specifications, the industry has adopted NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management," as an Industry initiative to manage shutdown tasks and associated electrical support to maintain risk at an acceptable low level. This may require the availability of additional equipment beyond that required by the shutdown Technical Specifications.

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

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LCO

One vital DC electrical power train consists of two channels. Train A consists of channels I and III and Train B consists of channels II and IV. The required train consisting of two batteries, one battery charger per battery, and the corresponding control equipment and interconnecting cabling within the train, is required to be OPERABLE. 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).

Control power for the DGs is provided by four DG DC electrical power subsystems, one per DG. Each DG DC electrical power subsystem is comprised of one 125 VDC battery, an associated charger, and associated control equipment and interconnecting cabling. One DG DC electrical power subsystem is required to be OPERABLE for each required DG.

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APPLICABILITY

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

- a. Required features to provide adequate coolant inventory makeup are available for the irradiated fuel assemblies in the core,
  - b. Required features needed to mitigate a fuel handling accident are available,
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## BASES

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### APPLICABILITY (continued)

- c. Required features necessary to mitigate the effects of events that can lead to core damage during shutdown are available, and
- d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

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

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### ACTIONS

LCO 3.0.3 is not applicable while in MODE 5 or 6. However, since irradiated fuel assembly movement can occur in MODE 1, 2, 3, or 4, the ACTIONS have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operations. Entering LCO 3.0.3, while in MODE 1, 2, 3, or 4 would require the unit to be shutdown unnecessarily.

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

With the required train of DC electrical power sources inoperable, the minimum required DC electrical power source is not available. Therefore, the allowance for sufficiently conservative actions is made (i.e., to suspend movement of irradiated fuel assemblies, and operations involving positive reactivity additions) that could result in loss of required SDM (MODE 5) or boron concentration (MODE 6). Suspending positive reactivity additions that could result in failure to meet the minimum SDM or boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than that what would be required in the RCS for minimum SDM or refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation. Introduction of temperature changes including temperature increases when operating with a positive MTC must also be evaluated to ensure they do not result in a loss of required SDM.

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 train and to continue this action until restoration is accomplished in order to provide the necessary DC electrical power to the unit safety systems.

BASES

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ACTIONS (continued)

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.

B.1

If one or more DG DC electrical power subsystems are inoperable, the associated DGs may be incapable of performing their intended function and must be immediately declared inoperable. This declaration also requires entry into the applicable Conditions and Required Actions for inoperable DGs, LCO 3.8.2, "AC Sources – Shutdown."

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SURVEILLANCE  
REQUIREMENTS

SR 3.8.5.1

SR 3.8.5.1 requires performance of all Surveillances required by SR 3.8.4.1 through SR 3.8.4.3. Therefore, see the corresponding Bases for LCO 3.8.4 for a discussion of each SR.

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REFERENCES

1. UFSAR, Chapter 6.
  2. UFSAR, Chapter 15.
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## B 3.8 ELECTRICAL POWER SYSTEMS

### B 3.8.6 Battery Parameters

#### BASES

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**BACKGROUND** This LCO delineates the limits on battery float current as well as electrolyte temperature, level, and float voltage for the Vital and diesel generator (DG) 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." In addition to the limitations of this Specification, the Battery Monitoring and Maintenance Program also implements a program specified in Specification 5.5.15 for monitoring various battery parameters.

The Vital battery cells are of flooded lead acid construction with a nominal specific gravity of 1.215. This specific gravity corresponds to an open circuit battery voltage of approximately 123.78 V for 60 cell battery (i.e., cell voltage of 2.063 volts per cell (Vpc)). The open circuit voltage is the voltage maintained when there is no charging or discharging. Optimal long term performance however, is obtained by maintaining a float voltage 2.17 Vpc. This provides adequate over-potential which limits the formation of lead sulfate and self discharge.

The DG battery cells are of flooded lead acid construction with a nominal specific gravity of 1.215. Each DG battery consists of 58 cells; however, a battery is considered OPERABLE with 57 cells if one is strapped out. Optimal long term performance is obtained by maintaining a float voltage of 2.20 to 2.25 Vpc. This provides adequate over-potential which limits the formation of lead sulfate and self-discharge.

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**APPLICABLE SAFETY ANALYSES** The initial conditions of Design Basis Accident (DBA) and transient analyses in the UFSAR, Chapter 6 (Ref. 3) and Chapter 15 (Ref. 4), 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 parameters satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

BASES

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LCO Battery 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. Battery parameter limits are conservatively established, allowing continued DC electrical system function even with limits not met. Additional preventative maintenance, testing, and monitoring performed in accordance with the Battery Monitoring and Maintenance Program is conducted as specified in Specification 5.5.15.

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

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

With one or more cells in one or more batteries < 2.07 V, the battery cell is degraded. Within 2 hours verification of the required battery charger OPERABILITY is made by monitoring the battery terminal voltage (SR 3.8.4.1) and of the overall battery state of charge by monitoring the battery float charge current (SR 3.8.6.1). This assures that there is still sufficient battery capacity to perform the intended function. Therefore, the affected battery is not required to be considered inoperable solely as a result of one or more cells in one or more batteries < 2.07 V, and continued operation is permitted for a limited period up to 24 hours.

Since the Required Actions only specify "perform," a failure of SR 3.8.4.1 or SR 3.8.6.1 acceptance criteria does not result in this Required Action not met. However, if one of the SRs is failed the appropriate Condition(s), depending on the cause of the failures, is entered. If SR 3.8.6.1 is failed then there is no assurance that there is still sufficient battery capacity to perform the intended function and the battery must be declared inoperable immediately.

B.1 and B.2

One or more vital batteries with float current > 2 amps or one or more DG batteries with float current > 1 amp indicates that a partial discharge of the battery capacity has occurred. This may be due to a temporary loss of a battery charger or possibly due to one or more battery cells in a low voltage condition reflecting some loss of capacity. Within 2 hours verification of the required battery charger OPERABILITY is made by monitoring the battery terminal voltage. If the terminal voltage is found to be less than the minimum established float voltage there are two possibilities, the battery charger is inoperable or is operating in the

## BASES

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### ACTIONS (continued)

current limit mode. Condition A addresses charger inoperability. If the charger is operating in the current limit mode after 2 hours that is an indication that the battery has been substantially discharged and likely cannot perform its required design functions. The time to return the battery to its fully charged condition in this case is a function of the battery charger capacity, the amount of loads on the associated DC system, the amount of the previous discharge, and the recharge characteristic of the battery. The charge time can be extensive, and there is not adequate assurance that it can be recharged within 12 hours (Required Action B.2.1 or B.2.2). The battery must therefore be declared inoperable.

If the float voltage is found to be satisfactory but there are one or more battery cells with float voltage less than 2.07 V, the associated "OR" statement in Condition F is applicable and the battery must be declared inoperable immediately. If float voltage is satisfactory and there are no cells less than 2.07 V there is good assurance that, within 12 hours, the battery will be restored to its fully charged condition (Required Action B.2.1 or B.2.2) from any discharge that might have occurred due to a temporary loss of the battery charger.

A discharged battery with float voltage (the charger setpoint) across its terminals indicates that the battery is on the exponential charging current portion (the second part) of its recharge cycle. The time to return a battery to its fully charged state under this condition is simply a function of the amount of the previous discharge and the recharge characteristic of the battery. Thus there is good assurance of fully recharging the battery within 12 hours, avoiding a premature shutdown with its own attendant risk.

If the condition is due to one or more cells in a low voltage condition but still greater than 2.07 V and float voltage is found to be satisfactory, this is not indication of a substantially discharged battery and 12 hours is a reasonable time prior to declaring the battery inoperable.

Since Required Action B.1 only specifies "perform," a failure of SR 3.8.4.1 acceptance criteria does not result in the Required Action not met. However, if SR 3.8.4.1 is failed, the appropriate Condition(s), depending on the cause of the failure, is entered.

## BASES

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### ACTIONS (continued)

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

With one or more batteries with one or more cells electrolyte level above the top of the plates, but below the minimum established design limits, the battery still retains sufficient capacity to perform the intended function. Therefore, the affected battery is not required to be considered inoperable solely as a result of electrolyte level not met. Within 31 days the minimum established design limits for electrolyte level must be re-established.

With electrolyte level below the top of the plates there is a potential for dryout and plate degradation. Required Actions C.1 and C.2 address this potential (as well as provisions in Specification 5.5.15, Battery Monitoring and Maintenance Program). They are modified by a Note that indicates they are only applicable if electrolyte level is below the top of the plates. Within 8 hours level is required to be restored to above the top of the plates. The Required Action C.2 requirement to verify that there is no leakage by visual inspection and the Specification 5.5.15.b item to initiate action to equalize and test in accordance with manufacturer's recommendation are taken from IEEE Standard 450. They are performed following the restoration of the electrolyte level to above the top of the plates. Based on the results of the manufacturer's recommended testing the batteries may have to be declared inoperable and the affected cells replaced.

#### D.1

With one or more batteries with pilot cell temperature less than the minimum established design limits, 12 hours is allowed to restore the temperature to within limits. A low electrolyte temperature limits the current and power available. Since the battery is sized with margin, while battery capacity is degraded, sufficient capacity exists to perform the intended function and the affected battery is not required to be considered inoperable solely as a result of the pilot cell temperature not met.

#### E.1

With one or more batteries in redundant trains with battery parameters not within limits there is not sufficient assurance that battery capacity has not been affected to the degree that the batteries can still perform their required function, given that redundant batteries are involved. With redundant batteries involved this potential could result in a total loss of function on multiple systems that rely upon the batteries. The longer Completion Times specified for battery parameters on non-redundant

BASES

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ACTIONS (continued)

batteries not within limits are therefore not appropriate, and the parameters must be restored to within limits on at least one train within 2 hours.

E.1

With one or more batteries with any battery parameter outside the allowances of the Required Actions for Condition A, B, C, D, or E, sufficient capacity to supply the maximum expected load requirement is not assured and the corresponding battery must be declared inoperable. Additionally, discovering one or more vital batteries with one or more battery cells float voltage less than 2.07 V and float current greater than 2 amps indicates that the battery capacity may not be sufficient to perform the intended functions. Similarly, discovering one or more DG batteries with one or more battery cells float voltage less than 2.07 V and float current greater than 1 amp indicates that the battery capacity may not be sufficient to perform the intended functions. The associated vital or DG battery must therefore be declared inoperable. In addition, if SR 3.8.6.6 or SR 3.8.6.7 are not met, the associated vital or DG battery is declared inoperable.

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SURVEILLANCE  
REQUIREMENTS

SR 3.8.6.1

Verifying battery float current while on float charge is used to determine the state of charge of the battery. Float charge is the condition in which the charger is supplying the continuous charge required to overcome the internal losses of a battery and maintain the battery in a charged state. The float current requirements are based on the float current indicative of a charged battery. Use of float current to determine the state of charge of the battery is consistent with IEEE-450 (Ref. 1). The minimum required procedural time to measure battery float current will be 30 seconds or as recommended by the float current measurement instrument manufacturer. The minimum float current measurement time is required to provide a more accurate battery float current reading.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

This SR is modified by a Note that states the float current requirement is not required to be met when battery terminal voltage is less than the minimum established float voltage of SR 3.8.4.1. When this float voltage is not maintained the Required Actions of LCO 3.8.4 ACTION A are being taken, which provide the necessary and appropriate verifications of the battery condition. Furthermore, the float current limit of 2 amps is

## BASES

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### SURVEILLANCE REQUIREMENTS (continued)

established based on the nominal float voltage value and is not directly applicable when this voltage is not maintained.

#### SR 3.8.6.2 and SR 3.8.6.5

Optimal long term battery performance is obtained by maintaining a float voltage greater than or equal to the minimum established design limits provided by the battery manufacturer, which corresponds to 129 V for vital batteries and 124 V for DG batteries at the battery terminals. This provides adequate over-potential, which limits the formation of lead sulfate and self discharge, which could eventually render the battery inoperable. Float voltages in this range or less, but greater than 2.07 Vpc, are addressed in Specification 5.5.15. SRs 3.8.6.2 and 3.8.6.5 require verification that the cell float voltages are equal to or greater than the short term absolute minimum voltage of 2.07 V.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

#### SR 3.8.6.3

The limit specified for electrolyte level ensures that the plates suffer no physical damage and maintains adequate electron transfer capability.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

#### SR 3.8.6.4

This Surveillance verifies that the pilot cell temperature is greater than or equal to the minimum established design limit (i.e., 60°F). Pilot cell electrolyte temperature is maintained above this temperature to assure the battery can provide the required current and voltage to meet the design requirements. Temperatures lower than assumed in battery sizing calculations act to inhibit or reduce battery capacity.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

#### SR 3.8.6.5

See SR 3.8.6.2 Bases

BASES

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.6.6 and SR 3.8.6.7

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.

Either the battery performance discharge test or the modified performance discharge test is acceptable for satisfying SR 3.8.6.6; however, only the modified performance discharge test may be used to satisfy the battery service test requirements of SR 3.8.4.3.

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.

It may consist of just two rates; for instance the one minute rate 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 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 must 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.

The acceptance criteria for this Surveillance are consistent with IEEE-450 (Ref. 1) and IEEE-485 (Ref. 5). These references recommend that the battery be replaced if its capacity is below 80% of the manufacturer's rating. A capacity of 80% shows that the battery rate of deterioration is increasing, even if there is ample capacity to meet the load requirements. Furthermore, the battery is sized to meet the assumed duty cycle loads when the battery design capacity reaches this 80% limit. The minimum battery capacity for the vital batteries has been raised from 80% to 82% to allow for possible discharge during the 5-minute delay associated with the Diesel Generator Start and Load Shed Timer.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

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SURVEILLANCE REQUIREMENTS (continued)

If the battery shows degradation, or if the battery has reached 85% of its expected life and capacity is < 100% of the manufacturer's rating, the Surveillance Frequency is reduced to 12 months. However, if the battery shows no degradation but has reached 85% of its expected life, the Surveillance Frequency is only reduced to 24 months for batteries that retain capacity  $\geq$  100% of the manufacturer's ratings. Degradation is indicated, according to IEEE-450 (Ref. 1), when the battery capacity drops by more than 10% relative to its capacity on the previous performance test or when it is  $\geq$  10% below the manufacturer's rating. These Frequencies are consistent with the recommendations in IEEE-450 (Ref. 1).

SR 3.8.6.6 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. This restriction from normally performing the Surveillance in MODE 1, 2, 3, or 4 is further amplified to allow portions of the Surveillance to be performed for the purpose of reestablishing OPERABILITY (e.g., post work testing following corrective maintenance, corrective modification, deficient or incomplete surveillance testing, and other unanticipated OPERABILITY concerns) provided an assessment determines plant safety is maintained or enhanced. This assessment shall, as a minimum, consider the potential outcomes and transients associated with a failed partial Surveillance, a successful partial Surveillance, and a perturbation of the offsite or onsite system when they are tied together or operated independently for the partial Surveillance; as well as the operator procedures available to cope with these outcomes. These shall be measured against the avoided risk of a plant shutdown and startup to determine that plant safety is maintained or enhanced when portions of the Surveillance are performed in MODE 1 or 2. Risk insights or deterministic methods may be used for the assessment. Credit may be taken for unplanned events that satisfy this SR.

SR 3.8.6.7 is modified by a Note stating that credit may be taken for unplanned events that satisfy this SR.

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REFERENCES

1. IEEE-450-2002.
  2. UFSAR, Chapter 8.
  3. UFSAR, Chapter 6.
  4. UFSAR, Chapter 15.
  5. IEEE-485-1983, June 1983.
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## B 3.8 ELECTRICAL POWER SYSTEMS

### B 3.8.7 Inverters - Operating

#### BASES

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**BACKGROUND** The inverters are the preferred source of power for the AC vital instrument power boards because of the stability and reliability they achieve. There are two unit inverters and one spare inverter per channel, each capable of supplying its associated AC vital instrument power boards, making a total of twelve inverters. Inverters 1-I and 2-I are connected to DC Channel I, inverters 1-II and 2-II are connected to DC Channel II, inverters 1-III and 2-III are connected to DC Channel III, and inverters 1-IV and 2-IV are connected to DC Channel IV. The spare inverter for a specified channel may be substituted for one of the two inverters of the same channel. The function of the inverter is to provide AC electrical power to the vital instrument power boards. The inverters can be powered from an internal AC source/rectifier or from the station battery. The station batteries provide 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 UFSAR, Chapter 8 (Ref. 1).

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**APPLICABLE SAFETY ANALYSES** The initial conditions of Design Basis Accident (DBA) and transient analyses in the UFSAR, 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 instrument power boards 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 10 CFR 50.36(c)(2)(ii).

## BASES

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### LCO

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

Maintaining the required inverters OPERABLE ensures that the redundancy incorporated into the design of the RPS and ESFAS instrumentation and controls is maintained. The eight inverters (two per channel) ensure an uninterruptible supply of AC electrical power to the AC vital instrument power boards even if the 480 volt safety boards are de-energized.

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

This LCO is modified by a Note that allows two inverters to be disconnected from a common battery for  $\leq 24$  hours, if the vital instrument power board(s) is powered from an inverter using internal AC source during the period and the remaining required inverters are OPERABLE. This allows an equalizing charge to be placed on one battery. If the inverters were not disconnected, the resulting voltage condition might damage the inverters. 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 instrument power board 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 remaining required inverters must be aligned to their associated batteries, regardless of the number of inverters or unit design.

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### APPLICABILITY

The inverters are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure that:

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

BASES

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APPLICABILITY (continued)

Inverter requirements for MODES 5 and 6 are covered in the Bases for LCO 3.8.8, "Inverters - Shutdown."

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ACTIONS

A.1

With a required inverter inoperable, its associated AC vital instrument power board becomes inoperable until it is manually re-energized from its inverter using internal AC source.

For this reason a Note has been included in Condition A requiring the entry into the Conditions and Required Actions of LCO 3.8.9, "Distribution Systems - Operating." This ensures that the vital instrument power board is re-energized within 8 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 instrument power board 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 instrument power board is the preferred source for powering instrumentation trip setpoint devices.

B.1 and B.2

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

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SURVEILLANCE  
REQUIREMENTS

SR 3.8.7.1

This Surveillance verifies that the inverters are functioning properly with all required circuit breakers closed and AC vital instrument power boards 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 instrument power boards.

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BASES

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SURVEILLANCE REQUIREMENTS (continued)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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REFERENCES

1. UFSAR, Chapter 8.
  2. UFSAR, Chapter 6.
  3. UFSAR, Chapter 15.
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B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.8 Inverters - Shutdown

BASES

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BACKGROUND	A description of the inverters is provided in the Bases for LCO 3.8.7, "Inverters - Operating."
APPLICABLE SAFETY ANALYSES	<p>The initial conditions of Design Basis Accident (DBA) and transient analyses in the UFSAR, 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.</p> <p>The OPERABILITY of the inverters is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY.</p> <p>The OPERABILITY of the minimum inverters to each AC vital board during MODES 5 and 6 ensures that:</p> <ul style="list-style-type: none"><li>a. The unit can be maintained in the shutdown or refueling condition for extended periods,</li><li>b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status, and</li><li>c. Adequate power is available to mitigate events postulated during shutdown, such as a fuel handling accident.</li></ul> <p>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 DBAs that are analyzed in MODES 1, 2, 3, and 4 have no specific analyses 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.</p>

BASES

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APPLICABLE SAFETY ANALYSES (continued)

The shutdown Technical Specification requirements are designed to ensure that the unit has the capability to mitigate the consequences of certain postulated accidents. Worst case DBAs which are analyzed for operating MODES are generally viewed not to be a significant concern during shutdown MODES due to the lower energies involved. The Technical Specifications therefore require a lesser complement of electrical equipment to be available during shutdown than is required during operating MODES. More recent work completed on the potential risks associated with shutdown, however, have found significant risk associated with certain shutdown evolutions. As a result, in addition to the requirements established in the Technical Specifications, the industry has adopted NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management," as an Industry initiative to manage shutdown tasks and associated electrical support to maintain risk at an acceptable low level. This may require the availability of additional equipment beyond that required by the shutdown Technical Specifications.

The inverters were previously identified as part of the distribution system and, as such, satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

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LCO

The inverters ensure the availability of electrical power for the instrumentation for systems required to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence or a postulated DBA. The battery powered inverters provide an uninterruptible supply of AC electrical power to the AC vital boards even if the 480 volt shutdown boards are de-energized. OPERABILITY of the inverters requires that each required vital board is powered by an inverter. This ensures the availability of sufficient inverter power sources to operate the unit in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents).

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APPLICABILITY

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

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

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### APPLICABILITY (continued)

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

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### ACTIONS

LCO 3.0.3 is not applicable while in MODE 5 or 6. However, since irradiated fuel assembly movement can occur in MODE 1, 2, 3, or 4, the ACTIONS have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operations. Entering LCO 3.0.3, while in MODE 1, 2, 3, or 4 would require the unit to be shutdown unnecessarily.

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

With one or more required inverters inoperable, the minimum required vital AC electrical power source is not available. Therefore, the allowance for sufficiently conservative actions is made (i.e., to suspend movement of irradiated fuel assemblies, and operations involving positive reactivity additions) that could result in loss of required SDM (MODE 5) or boron concentration (MODE 6). Suspending positive reactivity additions that could result in failure to meet the minimum SDM or boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than that what would be required in the RCS for minimum SDM or refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation. Introduction of temperature changes including temperature increases when operating with a positive MTC must also be evaluated to ensure they do not result in a loss of required SDM.

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.

BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.8.8.1

This Surveillance verifies that the inverters are functioning properly with all required circuit breakers closed and AC vital boards energized from the inverter. The verification of proper voltage and frequency output ensures that the required power is readily available for the instrumentation connected to the AC vital boards.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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REFERENCES

1. UFSAR, Chapter 6.
  2. UFSAR, Chapter 15.
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## B 3.8 ELECTRICAL POWER SYSTEMS

### B 3.8.9 Distribution Systems - Operating

#### BASES

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##### BACKGROUND

The two units share several structures and systems including the preferred and emergency (standby) electric power systems (UFSAR, Chapter 8.0). The vital DC Power System is shared to the extent that a few loads (e.g., the vital inverters) in one nuclear unit are energized by the DC power channels assigned primarily to power loads of the other unit. In no case does the sharing inhibit the safe shutdown of one unit while the other unit is experiencing an accident. The Standby Power System serving each unit is divided into two redundant load groups (power trains). These power trains (Train A and Train B for each unit) supply power to safety-related equipment. Generally, the Engineered Safety Feature (ESF) loads assigned to a unit are supplied by the unit designated trains. For example, Safety Injection (SI) pump 1A-A (associated with Unit 1) is supplied by Shutdown Board 1A-A (also associated with Unit 1) while SI pump 2A-A (associated with Unit 2) is supplied by Shutdown Board 2A-A (also associated with Unit 2).

Separate and similar systems and equipment are provided for each unit when required. In certain instances, both units share systems or some components of a system. Shared systems are the exception to the unit/power system association. Because both units share the power system, one unit's power system(s) supports certain components required by the other unit (e.g., emergency gas treatment system).

The onsite Class 1E AC, vital DC, DG DC, and AC vital instrument electrical power distribution systems are divided into two redundant and independent trains. Each electrical power distribution train consists of:

- a. an AC electrical power distribution subsystem,
- b. an AC vital instrument power distribution subsystem,
- c. a vital DC electrical power distribution subsystem, and
- d. a diesel generator (DG) DC electrical power distribution subsystem.

Each AC electrical power subsystem consists of two 6.9 kV Shutdown Boards and four 480 V Shutdown Boards. Each train of 6.9 kV Shutdown Boards has two separate and independent offsite sources of power as well as a dedicated onsite diesel generator (DG) source for each 6.9 kV Shutdown Board. Each 6.9 kV Shutdown Board is normally connected to a preferred offsite source. If the offsite sources are unavailable, the onsite emergency DGs supply power to the affected 6.9 kV Shutdown Boards.

## BASES

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### BACKGROUND (continued)

Control power for the 6.9 kV Shutdown Board 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."

Each 120 VAC vital instrument power distribution subsystem consists of two Unit 1 120 V AC vital instrument power boards and two Unit 2 120 V AC vital instrument power boards and are powered from the inverters.

Each 125 V vital DC electrical power distribution subsystem consists of two 125 V boards. The 125 V vital DC electrical power distribution subsystem has manual access to a fifth vital battery. The fifth vital battery is intended to serve as a replacement for any one of the four 125 V DC vital batteries with no loss of system reliability under any mode of operation.

Each DG DC electrical power distribution subsystem consists of two 125 V DC distribution panels that supply power to the respective DG's auxiliary loads. During normal operation, power is supplied to the distribution panel by a 480 VAC board through a battery charger. During emergency operation of the DG (loss of offsite power source), the distribution panel is supplied power from a dedicated battery. This panel supplies power for DG control, protection, and the engine DC lube oil circulating pump.

The list of electrical power boards and distribution panels required for the AC, vital DC, DG DC, and AC vital instrument electrical power distribution subsystems is presented in Table B 3.8.9-1.

Associated with each board listed in Table B 3.8.9-1 are a number of safety significant electrical loads. When one or more of the boards specified in Table B 3.8.9-1 becomes inoperable, entry into the appropriate ACTIONS of LCO 3.8.9 is required. Some boards, distribution panels, and motor control centers (MCCs), which help comprise the AC and DC electrical power distribution subsystems, are not listed in Table B 3.8.9-1. The loss of electrical loads associated with these boards, panels, or MCCs may not result in a complete loss of a safety function necessary to shut down the reactor and maintain it in a safe condition. Therefore, should one or more of these boards, panels, or MCCs become inoperable due to a failure not affecting the OPERABILITY of a board listed in Table B 3.8.9-1 (e.g., a breaker supplying a single distribution panel fails open), the individual loads associated with the board, panel, or MCC are declared inoperable, and the appropriate Conditions and Required Actions of the LCOs governing the individual loads are entered.

## BASES

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### APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the UFSAR, Chapter 6 (Ref. 1), and in the UFSAR, Chapter 15 (Ref. 2), assume ESF systems are OPERABLE. The AC, vital DC, DG DC, and AC vital instrument 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, vital DC, DG DC, and AC vital instrument 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 electrical 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 electrical power distribution systems satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

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### LCO

The required electrical power distribution subsystems listed in Table B 3.8.9-1 ensure the availability of AC, vital DC, DG DC, and AC vital instrument 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. Two electrical power distribution trains are required to be OPERABLE. Each train includes:

- a. an AC electrical power distribution subsystem (i.e., one Unit 1 6.9 kV shutdown board, one Unit 2 6.9 kV shutdown board, and associated 480 V shutdown boards),
  - b. an AC vital instrument power distribution subsystem (i.e., two Unit 1 120 V AC instrument power boards and two Unit 2 120 V AC instrument power boards),
  - c. a vital DC electrical power distribution subsystem (i.e., two 125 V DC boards), and
  - d. a DG DC electrical power distribution subsystem (i.e., two 125 V DG distribution panels).
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## BASES

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LCO (continued) Maintaining two electrical power distribution trains 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 trains will not prevent safe shutdown of the reactor.

OPERABLE AC electrical power distribution subsystems require the associated boards to be energized to their proper voltages. OPERABLE vital DC and DG DC electrical power distribution subsystems require the associated boards and distribution panels, as applicable, to be energized to their proper voltage from either the associated battery or charger. OPERABLE AC vital instrument power distribution subsystems require the associated boards to be energized to their proper voltage from the associated inverter via inverted DC voltage, or inverter using internal 120 volt regulated AC source.

In addition, tie breakers between redundant safety related AC, vital DC, DG DC, and AC vital instrument electrical power distribution subsystems, if they exist, must be open. This prevents any electrical malfunction in any power distribution subsystem from propagating to the redundant subsystem, that could cause the failure of a redundant subsystem and a loss of essential safety function(s). If any tie breakers are closed, the affected redundant electrical power distribution subsystems are considered inoperable. This applies to the onsite, safety related redundant electrical power distribution subsystems. It does not, however, preclude redundant Class 1E 6.9 kV Shutdown Boards from being powered from the same offsite circuit.

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

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

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

BASES

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ACTIONS

A.1

With one or more AC electrical power distribution subsystems (except AC vital instrument boards), inoperable due to one or more inoperable Unit 1 AC shutdown boards, and a loss of function has not occurred, the remaining portions of the AC 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 portions of the power distribution subsystems could result in the minimum required ESF functions not being supported. Therefore, the Unit 1 AC electrical distribution subsystems must be restored to OPERABLE status within 8 hours.

Condition A worst case scenario is one train of Unit 1 boards 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 Unit 1 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.

Required Action A.1 is modified by a Note that requires the applicable Conditions and Required Actions of LCO 3.8.4, "DC Sources - Operating," to be entered for vital DC electrical power trains made inoperable by inoperable AC electrical power distribution subsystems. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components. Inoperability of one or more AC electrical power distribution subsystems can result in loss of charging power to batteries and eventual loss of DC power. This Note ensures that the appropriate attention is given to restoring charging power to batteries, if necessary, after loss of distribution subsystems.

## BASES

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### ACTIONS (continued)

#### B.1

With one or more AC vital instrument power distribution subsystems inoperable, and a loss of function has not yet occurred, the remaining OPERABLE portions of the AC vital instrument power distribution subsystems 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 AC vital instrument power distribution subsystem(s) must be restored to OPERABLE status within 8 hours by powering the affected subsystems from the associated inverter via inverted DC or inverter using internal 120 volt regulated AC source.

Condition B represents one or more AC vital instrument power distribution subsystems 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 instrument power boards and restoring power to the affected vital instrument power boards.

This 8 hour limit is more conservative than Completion Times allowed for the vast majority of components that are without adequate vital AC power. Taking exception to LCO 3.0.2 for components without adequate vital AC power, that would have the Required Action Completion Times shorter than 8 hours if declared inoperable, is acceptable because of:

- 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 8 hour Completion Time takes into account the importance to safety of restoring the AC vital instrument power distribution subsystems to OPERABLE status, the redundant capability afforded by the remaining

## BASES

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### ACTIONS (continued)

OPERABLE AC vital instrument power boards, and the low probability of a DBA occurring during this period.

#### C.1

With one or more vital DC electrical power distribution subsystems inoperable, and a loss of function has not yet occurred, the remaining portions of the vital 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 portions of the vital DC electrical power distribution subsystems could result in the minimum required ESF functions not being supported. Therefore, the vital DC subsystems must be restored to OPERABLE status within 2 hours by powering the subsystem from the associated battery or charger.

Condition C represents one or more vital DC electrical power distribution subsystems 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 panels and restoring power to the affected panels.

This 2 hour limit is more conservative than Completion Times allowed for the vast majority of components that 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 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 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 vital DC electrical power distribution subsystems is consistent with Regulatory Guide 1.93 (Ref. 3).

## BASES

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### ACTIONS (continued)

#### D.1

With one or more AC electrical power distribution subsystems (except AC vital instrument boards) inoperable due to one or more inoperable Unit 2 AC shutdown boards and a loss of function has not occurred, the remaining AC 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 power distribution subsystems could result in the minimum required ESF functions not being supported. Therefore, the associated required feature(s) must be declared inoperable immediately and the corresponding Condition(s) entered. The Required Action(s) of these Condition(s) will determine the impact of the inoperable Unit 2 AC shutdown board(s).

Condition D is modified by two notes that limit the conditions and parameters that allow entry into Condition D. The first note states that Condition D is only applicable during planned maintenance. This will allow the plant configuration to be aligned to minimize features being inoperable when the opposite unit shutdown board is made inoperable. The second note limits the applicability of Condition D to the time period when the opposite unit is either defueled or in MODE 6 following defueled with refueling water cavity level  $\geq 23$  ft. above the top of the reactor vessel flange. This note limits the time period allowing Condition D to be entered, minimizing when the allowance can be utilized. The allowance for Condition D is acceptable based on the following:

- a. The opposite unit's AC shutdown boards are not as critical to the operating unit (fewer operating unit loads) as the operating unit's AC shutdown boards.
- b. Performing maintenance on these components will increase the reliability of the Class 1E AC Electrical Power Distribution System.
- c. The Required Actions associated with the features declared inoperable provide compensatory measures during the performance of the planned maintenance.
- d. The limited opportunities that allow the planned maintenance to occur.

During the planned maintenance of the Unit 2 AC shutdown boards, if a condition is discovered on these boards requiring corrective maintenance, this maintenance may be performed under Condition D.

## BASES

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### ACTIONS (continued)

#### E.1

With one or more AC electrical power distribution subsystems (except AC vital instrument boards) inoperable due to one or more inoperable Unit 2 AC shutdown boards for reasons other than Condition D and a loss of function has not occurred, the remaining AC 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 power distribution subsystems could result in the minimum required ESF functions not being supported. Therefore, the inoperable Unit 2 AC electrical power distribution subsystem(s) must be returned to OPERABLE status within 24 hours. The 24 hour time limit before requiring a unit shutdown in this Condition is acceptable because the opposite unit's AC shutdown boards are not as critical to the operating unit (fewer operating unit loads) as the operating unit's AC shutdown boards.

#### F.1

With one or more DG DC electrical power distribution panels inoperable there is no longer assurance the supported DG(s) is able to start and perform its necessary safety function. The affected DG(s) must therefore be declared inoperable immediately and the corresponding Condition(s) entered.

#### G.1 and G.2

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

#### H.1

Condition H corresponds to a level of degradation in the electrical power distribution system that results in a loss of safety function. When more than one inoperable electrical power distribution subsystem results in the loss of a safety function, the plant is in a condition outside the accident analysis. Therefore, no additional time is justified for continued operation. LCO 3.0.3 must be entered immediately to commence a controlled shutdown.

BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.8.9.1

This Surveillance verifies that the required AC, vital DC, DG DC, and AC vital instrument electrical power distribution subsystems are functioning properly, with the correct circuit breaker alignment. The correct breaker alignment ensures the appropriate separation and independence of the electrical power distribution trains is maintained, and the appropriate voltage is available to each required board. The verification of proper voltage availability on the boards ensures that the required voltage is readily available for motive as well as control functions for critical system loads connected to these boards.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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REFERENCES

1. UFSAR, Chapter 6.
  2. UFSAR, Chapter 15.
  3. Regulatory Guide 1.93, December 1974.
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Table B 3.8.9-1 (page 1 of 1)  
AC and DC Electrical Power Distribution Systems

TYPE	VOLTAGE (nominal)	SR 3.8.9.1 Voltage Range	TRAIN A		TRAIN B	
			SUBSYSTEMS		SUBSYSTEMS	
AC electrical power	6900 V	$\geq 6560 \text{ V}$ and $\leq 7260 \text{ V}$	<u>Unit 1</u> SD BD 1A-A	<u>Unit 2</u> SD BD 2A-A	<u>Unit 1</u> SD BD 1B-B	<u>Unit 2</u> SD BD 2B-B
	480 V	$\geq 440 \text{ V}$ and $\leq 508 \text{ V}$	SD BD 1A1-A 1A2-A	SD BD 2A1-A 2A2-A	SD BD 1B1-B 1B2-B	SD BD 2B1-B 2B2-B
AC vital instrument electrical power	120 V	$\geq 120.6 \text{ V}$ and $\leq 126.6 \text{ V}$	<u>Unit 1</u> Board 1-I Board 1-III	<u>Unit 2</u> Board 2-I Board 2-III	<u>Unit 1</u> Board 1-II Board 1-IV	<u>Unit 2</u> Board 2-II Board 2-IV
Vital DC electrical power	125 V	$\geq 129 \text{ V}$ and $\leq 140 \text{ V}$	Board I	Board III	Board II	Board IV
DG DC electrical power	125 V	$\geq 124 \text{ V}$ and $\leq 135 \text{ V}$	DG 1A-A Dist. Pnl.	DG 2A-A Dist. Pnl.	DG 1B-B Dist. Pnl.	DG 2B-B Dist. Pnl.

## B 3.8 ELECTRICAL POWER SYSTEMS

### B 3.8.10 Distribution Systems - Shutdown

#### BASES

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BACKGROUND	A description of the AC, vital DC, diesel generator (DG) DC, and AC vital instrument electrical power distribution systems is provided in the Bases for LCO 3.8.9, "Distribution Systems - Operating."
APPLICABLE SAFETY ANALYSES	<p>The initial conditions of Design Basis Accident and transient analyses in the UFSAR, Chapter 6 (Ref. 1) and Chapter 15 (Ref. 2), assume Engineered Safety Feature (ESF) systems are OPERABLE. The AC, vital DC, DG DC, and AC vital instrument 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.</p> <p>The OPERABILITY of the AC, vital DC, DG DC, and AC vital instrument electrical power distribution system is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY.</p> <p>The OPERABILITY of the minimum AC, vital DC, DG DC, and AC vital instrument electrical power distribution subsystems during MODES 5 and 6, and during movement of irradiated fuel assemblies ensures that:</p> <ol style="list-style-type: none"><li>The unit can be maintained in the shutdown or refueling condition for extended periods,</li><li>Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status, and</li><li>Adequate power is provided to mitigate events postulated during shutdown, such as a fuel handling accident.</li></ol> <p>The AC and DC electrical power distribution systems satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).</p>

BASES

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

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

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

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

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

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ACTIONS LCO 3.0.3 is not applicable while in MODE 5 or 6. However, since irradiated fuel assembly movement can occur in MODE 1, 2, 3, or 4, the ACTIONS have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operations. Entering LCO 3.0.3, while in MODE 1, 2, 3, or 4 would require the unit to be shutdown unnecessarily.

## BASES

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### ACTIONS (continued)

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

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 irradiated 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 subsystem 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 movement of irradiated fuel assemblies, and operations involving positive reactivity additions that could result in loss of required SDM (MODE 5) or boron concentration (MODE 6). Suspending positive reactivity additions that could result in failure to meet the minimum SDM or boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than that what would be required in the RCS for minimum SDM or refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation. Introduction of temperature changes including temperature increases when operating with a positive MTC must also be evaluated to ensure they do not result in a loss of required SDM.

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

Notwithstanding performance of the above conservative Required Actions, a required residual heat removal (RHR) subsystem may be inoperable. In this case, Required Actions A.2.1 through A.2.3 do not adequately address the concerns relating to coolant circulation and heat removal. Pursuant to LCO 3.0.6, the RHR ACTIONS would not be entered. Therefore, Required Action A.2.4 is provided to direct declaring RHR inoperable and not in operation, which results in taking the appropriate RHR actions.

BASES

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ACTIONS (continued)

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.

B.1

If one or more required DG DC electrical power distribution panels are inoperable, the associated DGs may be incapable of performing their intended function and must be immediately declared inoperable. This declaration also requires entry into the applicable Conditions and Required Actions for inoperable DGs, LCO 3.8.2, "AC Sources – Shutdown."

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SURVEILLANCE  
REQUIREMENTS

SR 3.8.10.1

This Surveillance verifies that the AC, vital DC, DG DC, and AC vital instrument electrical power distribution subsystems are functioning properly, with all the boards energized. The verification of proper voltage availability on the boards ensures that the required power is readily available for motive as well as control functions for critical system loads connected to these boards.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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REFERENCES

1. UFSAR, Chapter 6.
  2. UFSAR, Chapter 15.
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## B 3.9 REFUELING OPERATIONS

### B 3.9.1 Boron Concentration

#### BASES

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

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

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 storage tank through the open reactor vessel by gravity feeding or by the use of the Residual Heat Removal (RHR) System pumps.

The pumping action of the RHR System in the RCS and the natural circulation due to thermal driving heads in the reactor vessel and refueling cavity mix the added concentrated boric acid with the water in the refueling canal. The RHR System is in operation during refueling (see LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level," and LCO 3.9.6, "Residual Heat Removal (RHR) 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.

BASES

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APPLICABLE  
SAFETY  
ANALYSES

An uncontrolled boron dilution accident is not credible during refueling. This accident is prevented by administrative controls which isolate the RCS from significant sources of unborated water. 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 plant refueling procedures that verify the correct fuel loading plan (including full core mapping) ensure that the  $k_{\text{eff}}$  of the core will remain  $\leq 0.95$  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 Bases B 3.1.1, "SHUTDOWN MARGIN (SDM)."

The RCS boron concentration satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

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LCO

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

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APPLICABILITY

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

The Applicability is modified by a Note. The Note states that the limits on boron concentration are only applicable to the refueling canal and the refueling cavity when those volumes are connected to the RCS. When the refueling canal and the refueling cavity are isolated from the RCS, no potential path for boron dilution exists.

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ACTIONS

A.1

Continuation of 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

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## BASES

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### ACTIONS (continued)

volume in the RCS, the refueling canal, or the refueling cavity is less than its limit, all operations involving positive reactivity additions must be suspended immediately.

Suspension of positive reactivity additions shall not preclude moving a component to a safe position. Operations that individually add limited positive reactivity (e.g., temperature fluctuations from inventory addition or temperature control fluctuations), but when combined with all other operations affecting core reactivity (e.g., intentional boration) result in overall net negative reactivity addition, are not precluded by this action.

#### A.2

In addition to immediately suspending positive reactivity additions, boration to restore the concentration must be initiated immediately.

In determining the required combination of boration flow rate and concentration, no unique Design Basis Event must be satisfied. The only requirement is to restore the boron concentration to its required value as soon as possible. In order to raise the boron concentration as soon as possible, the operator should begin boration with the best source available for unit conditions.

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

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### SURVEILLANCE REQUIREMENTS

#### SR 3.9.1.1

This SR ensures that the coolant boron concentration in the RCS, and connected portions of the refueling canal and the refueling cavity, is within the COLR limits. The boron concentration of the coolant in each required volume is determined periodically by chemical analysis. Prior to re-connecting portions of the refueling canal or the refueling cavity to the RCS, this SR must be met per SR 3.0.4. If any dilution activity has occurred while the cavity or canal were disconnected from the RCS, this SR ensures the correct boron concentration prior to communication with the RCS.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

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- REFERENCES
1. 10 CFR 50, Appendix A, GDC 26.
  2. UFSAR, Chapter 15.
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## B 3.9 REFUELING OPERATIONS

### B 3.9.2 Unborated Water Source Isolation Valves

#### BASES

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**BACKGROUND** During MODE 6 operations, all isolation valves in a specified combination for reactor makeup water sources containing unborated water that are connected to the Reactor Coolant System (RCS) must be closed to prevent unplanned boron dilution of the reactor coolant. The isolation valves must be secured in the closed position.

The Chemical and Volume Control System is capable of supplying borated and unborated water to the RCS through various flow paths. Since a positive reactivity addition made by reducing the boron concentration is inappropriate during MODE 6, isolation of all unborated water sources prevents an unplanned boron dilution.

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**APPLICABLE SAFETY ANALYSES** The possibility of an inadvertent boron dilution event (Ref. 1) occurring during MODE 6 refueling operations is precluded by adherence to this LCO, which requires that potential dilution sources be isolated. Closing the required valves during refueling operations prevents the flow of unborated water to the filled portion of the RCS. The valves are used to isolate unborated water sources. These valves have the potential to indirectly allow dilution of the RCS boron concentration in MODE 6. By isolating unborated water sources, a safety analysis for an uncontrolled boron dilution accident in accordance with the Standard Review Plan (Ref. 2) is not required for MODE 6.

The RCS boron concentration satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

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**LCO** This LCO requires that flow paths to the RCS from unborated water sources be isolated to prevent unplanned boron dilution during MODE 6 and thus avoid a reduction in SDM. These flow paths are isolated by securing, in the closed position, each valve in one of the valve combinations listed in Table B 3.9.2-1.

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**APPLICABILITY** In MODE 6, this LCO is applicable to prevent an inadvertent boron dilution event by ensuring isolation of all sources of unborated water to the RCS.

For all other MODES, the boron dilution accident was analyzed and was found to be capable of being mitigated.

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BASES

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ACTIONS

The ACTIONS Table has been modified by a Note that allows separate Condition entry for each unborated water source isolation valve in the required valve combination.

A.1

Preventing inadvertent dilution of the reactor coolant boron concentration is dependent on maintaining the unborated water isolation valves secured closed. Securing the valves in the closed position ensures that the valves cannot be inadvertently opened. The Completion Time of "immediately" requires an operator to initiate actions to close an open valve and secure the isolation valve in the closed position immediately. The intent of this Required Action is that once actions are initiated, they must be continued until the valves are secured in the closed position.

A.2

Due to the potential of having diluted the boron concentration of the reactor coolant, SR 3.9.1.1 (verification of boron concentration) must be performed whenever Condition A is entered to demonstrate 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.

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SURVEILLANCE  
REQUIREMENTS

SR 3.9.2.1

At least one combination of valves, listed in Table B 3.9.2-1, is to be secured closed to isolate possible dilution paths. The likelihood of a significant reduction in the boron concentration during MODE 6 operations is remote due to the large mass of borated water in the refueling cavity and the fact that all unborated water sources are isolated, precluding a dilution. The boron concentration is checked, during MODE 6, under SR 3.9.1.1. This Surveillance demonstrates that the valves are closed by administrative means.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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REFERENCES

1. UFSAR, Section 15.2.4.
  2. NUREG-0800, Section 15.4.6.
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Table B 3.9.2-1  
 Unborated Water Source Isolation Valves

Isolation Valve Combination	Valve Numbers
Combination A	1-81-536 1-62-922 1-62-916 1-62-933
Combination B	1-81-536 1-62-922 1-62-916 1-62-940 1-62-696 1-62-929 1-62-932 1-FCV-62-128
Combination C	1-81-536 1-62-907 1-62-914 1-62-921 1-62-933
Combination D	1-81-536 1-62-907 1-62-914 1-62-921 1-62-940 1-62-929 1-62-932 1-62-696 1-FCV-62-128

## B 3.9 REFUELING OPERATIONS

### B 3.9.3 Nuclear Instrumentation

#### BASES

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**BACKGROUND** The source range neutron flux monitors are used during refueling operations to monitor the core reactivity condition. The installed source range neutron flux monitors are part of the Nuclear Instrumentation System (NIS). These detectors are located external to the reactor vessel and detect neutrons leaking from the core.

The installed source range neutron flux monitors are Dual Chamber Unguarded Fission Chamber detectors. The detectors monitor the neutron flux in counts per second. The instrument range covers six decades of neutron flux (1E+6 cps) with a 7% instrument accuracy. The detectors also provide continuous visual indication in the control room and an audible count rate in the containment and the control room. The NIS is designed in accordance with the criteria presented in Reference 1.

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**APPLICABLE SAFETY ANALYSES** Two OPERABLE source range neutron flux monitors are required to provide a signal to alert the operator to unexpected changes in core reactivity such as with a boron dilution accident (Ref. 2) or an improperly loaded fuel assembly (Ref. 3). The need for a requirement for the source range neutron flux monitors to mitigate an uncontrolled boron dilution accident is eliminated by isolating all unborated water sources as required by LCO 3.9.2, "Unborated Water Source Isolation Valves."

Fuel assembly loading errors are prevented by administrative procedures implemented during core loading (Ref. 3). These administrative procedures include detailed neutron count rate monitoring to determine that the just loaded fuel assembly does not excessively increase the count rate and that the extrapolated inverse count rate ratio is not decreasing for unexplained reasons.

The source range neutron flux monitors are not assumed to function during a MODE 6 design basis accident or transient. However, because the source range neutron flux monitors provide the primary on-scale monitoring of neutron flux levels during refueling, they are retained in the technical specifications.

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**LCO** This LCO requires that two source range neutron flux monitors be OPERABLE to ensure that redundant monitoring capability is available to detect changes in core reactivity. To be OPERABLE, each monitor must provide visual indication in the control room.

## BASES

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**APPLICABILITY** In MODE 6, the source range neutron flux monitors must be OPERABLE to determine changes in core reactivity. There are no other direct means available to check core reactivity levels. In MODES 2, 3, 4, and 5, these same installed source range detectors and circuitry are also required to be OPERABLE by LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation," and LCO 3.3.9, "Boron Dilution Monitoring Instrumentation (BDMI)."

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**ACTIONS** A.1 and A.2

With only one source range neutron flux monitor OPERABLE, redundancy has been lost. Since these instruments are the only direct means of monitoring core reactivity conditions, CORE ALTERATIONS and introduction of coolant into the RCS with boron concentration less than required to meet the minimum boron concentration of LCO 3.9.1 must be suspended immediately. Suspending CORE ALTERATIONS is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than what would be required in the RCS for minimum refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation. Performance of Required Action A.1 shall not preclude completion of movement of a component to a safe position.

### B.1

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

### B.2

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

The Completion Time of once per 12 hours is sufficient to obtain and analyze a reactor coolant sample for boron concentration and ensures that unplanned changes in boron concentration would be identified. The

BASES

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ACTIONS (continued)

12 hour Frequency is reasonable, considering the low probability of a change in core reactivity during this time period.

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SURVEILLANCE  
REQUIREMENTS

SR 3.9.3.1

SR 3.9.3.1 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 Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.9.3.2

SR 3.9.3.2 is the performance of a CHANNEL CALIBRATION. 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 Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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REFERENCES

1. 10 CFR 50, Appendix A, GDC 13, GDC 26, GDC 28, and GDC 29.
  2. UFSAR, Section 15.2.4.
  3. UFSAR, Section 15.3.3.
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## B 3.9 REFUELING OPERATIONS

### B 3.9.4 Containment Penetrations

#### BASES

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

The containment serves to contain fission product radioactivity that may be released from the reactor core following an accident, such that offsite radiation exposures are maintained within the requirements of 10 CFR 50.67. Additionally, the containment provides radiation shielding from the fission products that may be present in the containment atmosphere following accident conditions.

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

The containment air locks, which are also part of the containment pressure boundary, provide a means for personnel access during MODES 1, 2, 3, and 4 unit operation in accordance with LCO 3.6.2, "Containment Air Locks." Each air lock has a door at both ends. The doors are normally interlocked to prevent simultaneous opening when containment OPERABILITY is required. During periods of unit shutdown when containment closure is not required, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary. During 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 capable of being closed.

## BASES

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### BACKGROUND (continued)

The requirements for containment penetration closure ensure that a release of fission product radioactivity within containment will be restricted to within regulatory limits.

The Reactor Building Purge Ventilation (RBPV) System includes three subsystems. The normal subsystem includes four 24 inch purge penetrations and two 24 inch exhaust penetrations. The second subsystem, a pressure relief system, includes an 8 inch exhaust penetration. The third subsystem includes a 12 inch instrument room supply penetration and a 12 inch exhaust penetration. During MODES 1, 2, 3, and 4, no more than one pair of containment purge lines (one set of supply valves and one set of exhaust valves) may be opened (Ref. 4). None of the subsystems are subject to a Specification in MODE 5.

In MODE 6, large air exchangers are necessary to conduct refueling operations. The normal 24 inch purge system is used for this purpose, and all valves are closed by Containment Ventilation Isolation in accordance with LCO 3.3.6, "Containment Ventilation Isolation Instrumentation."

The other containment penetrations that provide direct access from containment atmosphere to outside atmosphere must be isolated on at least one side. Isolation may be achieved by an OPERABLE automatic isolation valve (either open or closed), 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 irradiated fuel movements (Ref. 1).

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### APPLICABLE SAFETY ANALYSES

During 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 resulting from dropping a single irradiated fuel assembly (Ref. 2). The requirements of LCO 3.9.7, "Refueling Cavity Water Level," in conjunction with a minimum decay time of 100 hours prior to irradiated fuel movement with containment closure capability, ensures that the release of fission product radioactivity, subsequent to a fuel handling accident, results in doses that are within the values specified in 10 CFR 50.67 or the NRC staff approved licensing basis (e.g., Regulatory Guide 1.183, (Ref. 3) limits).

Containment penetrations satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

## BASES

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### LCO

This LCO limits the consequences of a fuel handling accident involving handling irradiated fuel 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 or to the auxiliary building secondary containment enclosure, to be closed except for the OPERABLE containment purge and exhaust penetrations and the containment personnel air locks. For the OPERABLE containment purge and exhaust penetrations, this LCO ensures that these penetrations are isolable by an automatic Containment Ventilation isolation valve. The OPERABILITY requirements for this LCO ensure that the containment ventilation isolation valve closure times specified in the UFSAR can be achieved and, therefore, meet the assumptions used in the safety analysis to ensure that releases through the valves are terminated, such that radiological doses are within the acceptance limit.

During movement of recently irradiated fuel assemblies within containment, the equipment hatch is required to be held in place by at least four bolts.

The LCO is modified by a Note allowing penetration flow paths with direct access from the containment atmosphere that transverse and terminate in the Auxiliary Building Secondary Containment Enclosure to be unisolated under administrative controls. Administrative controls ensure that 1) appropriate personnel are aware of the open status of the penetration flow path during movement of irradiated fuel assemblies within containment, and 2) specified individuals are designated and readily available to isolate the flow path in the event of a fuel handling accident.

The containment personnel air lock doors may be open during movement of irradiated fuel in the containment provided that one door is capable of being closed in the event of a fuel handling accident. Should a fuel handling accident occur inside containment, at least one personnel air lock door will be closed following an evacuation of containment.

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### APPLICABILITY

The containment penetration requirements are applicable when there is a potential for the limiting fuel handling accident (FHA). The applicability requirements are based on the FHA analysis which assumes a fuel assembly is dropped and damaged during refueling. In MODES 1, 2, 3, and 4, containment penetration requirements are addressed by LCO 3.6.1. In MODES 5 and 6, when movement of irradiated fuel assemblies within containment is not being conducted, the potential for a fuel handling accident does not exist. Additionally, due to radioactive decay, a fuel handling accident involving handling irradiated fuel that is not "recently" irradiated (i.e., fuel that has occupied part of a critical reactor core within the previous 100 hours) will result in doses that are within the values specified in 10 CFR 50.67 even without containment closure capability. The applicability of 3.9.4.a. for the Containment Building

BASES

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APPLICABILITY (continued)

Equipment Hatch is "During the movement of recently irradiated fuel in containment" which maintains the containment closure requirements when the fuel has not sufficiently decayed to remain within these limits. The applicability of 3.9.4.b. and c. for the Containment Air Lock Doors and containment penetrations that provide direct access from containment atmosphere to outside atmosphere is "During movement of irradiated fuel in containment."

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ACTIONS

A.1

If the containment equipment hatch, is not in the required status, the unit must be placed in a condition where the isolation function is not needed. This is accomplished by immediately suspending movement of recently irradiated fuel assemblies within containment. Performance of these actions shall not preclude completion of movement of a component to a safe position.

B.1

If the containment building air lock doors or any other containment penetration that provides direct access from the containment atmosphere to the outside atmosphere is not in the required status, including the Containment Ventilation Isolation valve(s) not capable of automatic actuation when the purge and exhaust valves are open, the unit must be placed in a condition where the isolation function is not needed. This is accomplished by immediately suspending movement of irradiated fuel assemblies within containment. Performance of these actions shall not preclude completion of movement of a component to a safe position.

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SURVEILLANCE  
REQUIREMENTS

SR 3.9.4.1

This Surveillance demonstrates that each containment penetration is in its required status. The requirement that penetrations are capable of being closed by an OPERABLE automatic containment ventilation isolation valve, can be verified by ensuring that each required containment ventilation isolation valve operator has motive power.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.9.4.2

This Surveillance demonstrates that each containment ventilation isolation valve, that is not locked, sealed, or otherwise secured in

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## BASES

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### SURVEILLANCE REQUIREMENTS (continued)

position, actuates to its isolation position on manual initiation or on an actual or simulated actuation signal.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

The SR is modified by a Note stating that this Surveillance is not required to be met for valves in isolated penetrations. The LCO provides the option to close penetrations in lieu of requiring automatic actuation capability.

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### REFERENCES

1. GPU Nuclear Safety Evaluation SE-0002000-001, Rev. 0, May 20, 1988.
  2. Document ID: LTR-CRA-02-219, Westinghouse Electric Company, "Radiological Consequences of Fuel Handling Accidents for the Sequoyah Nuclear Plant Units 1 and 2."
  3. Regulatory Guide 1.183, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors, July 2000.
  4. UFSAR, Section 9.4.7.
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## B 3.9 REFUELING OPERATIONS

### B 3.9.5 Residual Heat Removal (RHR) and Coolant Circulation - High Water Level

#### BASES

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**BACKGROUND** The purpose of the RHR System in MODE 6 is to remove decay heat and sensible heat from the Reactor Coolant System (RCS), as required by GDC 34 (Ref. 1). Operation of the RHR system provides mixing of borated coolant and prevents boron stratification. Heat is removed from the RCS by circulating reactor coolant through the RHR heat exchanger(s), where the heat is transferred to the Component Cooling Water System. The coolant is then returned to the RCS via the RCS cold leg(s). Operation of the RHR System for normal cooldown or decay heat removal is manually accomplished from the control room. The heat removal rate is adjusted by controlling the flow of reactor coolant through the RHR heat exchanger(s) and the bypass. Mixing of the reactor coolant is maintained by this continuous circulation of reactor coolant through the RHR System.

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**APPLICABLE SAFETY ANALYSES** If the reactor coolant temperature is not maintained below 200°F, boiling of the reactor coolant could result. This could lead to a loss of coolant in the reactor vessel. Additionally, boiling of the reactor coolant could lead to a reduction in boron concentration in the coolant due to boron plating out on components near the areas of the boiling activity. 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 loop of the RHR System is required to be operational in MODE 6, with the water level  $\geq$  23 ft above the top of the reactor vessel flange, to prevent this challenge. The LCO does permit the RHR pump to be removed from operation for short durations, under the condition that the boron concentration is not diluted. This conditional stopping of the RHR pump does not result in a challenge to the fission product barrier.

The RHR System satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

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

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

BASES

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LCO (continued)

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

The LCO is modified by a Note that allows the required operating RHR loop to be removed from operation for up to 1 hour per 8 hour period, provided no operations are permitted that would dilute the RCS boron concentration by introduction of coolant into the RCS with boron concentration less than required to meet the minimum boron concentration of LCO 3.9.1, "Boron Concentration." Boron concentration reduction with coolant at boron concentrations less than required to assure the RCS boron concentration is maintained 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 RHR isolation valve testing. During this 1 hour period, decay heat is removed by natural convection to the large mass of water in the refueling cavity.

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APPLICABILITY

One RHR loop must be OPERABLE and in operation in MODE 6, with the water level  $\geq$  23 ft above the top of the reactor vessel flange, to provide decay heat removal. The 23 ft water level was selected because it corresponds to the 23 ft requirement established for fuel movement in LCO 3.9.7, "Refueling Cavity Water Level." Requirements for the RHR 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). RHR loop requirements in MODE 6 with the water level  $<$  23 ft are located in LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level."

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ACTIONS

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

A.1

If RHR loop requirements are not met, there will be no forced circulation to provide mixing to establish uniform boron concentrations. Suspending positive reactivity additions that could result in failure to meet the minimum boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than that what would be required in the RCS for minimum refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation.

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BASES

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ACTIONS (continued)

A.2

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

A.3

If RHR loop requirements are not met, actions shall be initiated and continued in order to satisfy RHR loop requirements. With the unit in MODE 6 and the refueling water level  $\geq$  23 ft above the top of the reactor vessel flange, corrective actions shall be initiated immediately.

A.4, A.5, A.6.1, and A.6.2

If no RHR is in operation, the following actions must be taken within 4 hours:

- a. The equipment hatch must be closed and secured with four bolts;
- b. One door in each air lock must be closed; and
- c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere must be either closed by a manual or automatic isolation valve, blind flange, or equivalent, or verified to be capable of being closed by an OPERABLE automatic Containment Ventilation isolation valve.

With RHR loop requirements not met, the potential exists for the coolant to boil and release radioactive gas to the containment atmosphere. Performing the actions described above ensures that all containment penetrations are either closed or can be closed so that the dose limits are not exceeded.

The Completion Time of 4 hours allows fixing of most RHR problems and is reasonable, based on the low probability of the coolant boiling in that time.

BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.9.5.1

This Surveillance demonstrates that the RHR 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 Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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REFERENCES

1. UFSAR, Section 5.5.7.
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## B 3.9 REFUELING OPERATIONS

### B 3.9.6 Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level

#### BASES

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**BACKGROUND** The purpose of the RHR System in MODE 6 is to remove decay heat and sensible heat from the Reactor Coolant System (RCS), as required by GDC 34 (Ref. 1). Operation of the RHR system provides mixing of borated coolant, and prevents boron stratification. Heat is removed from the RCS by circulating reactor coolant through the RHR heat exchangers where the heat is transferred to the Component Cooling Water System. The coolant is then returned to the RCS via the RCS cold leg(s). Operation of the RHR System for normal cooldown decay heat removal is manually accomplished from the control room. The heat removal rate is adjusted by controlling the flow of reactor coolant through the RHR heat exchanger(s) and the bypass lines. Mixing of the reactor coolant is maintained by this continuous circulation of reactor coolant through the RHR System.

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**APPLICABLE SAFETY ANALYSES** If the reactor coolant temperature is not maintained below 200°F, boiling of the reactor coolant could result. This could lead to a loss of coolant in the reactor vessel. Additionally, boiling of the reactor coolant could lead to a reduction in boron concentration in the coolant due to the boron plating out on components near the areas of the boiling activity. The loss of reactor coolant and the reduction of boron concentration in the reactor coolant will eventually challenge the integrity of the fuel cladding, which is a fission product barrier. Two loops of the RHR System are required to be OPERABLE, and one loop in operation, in order to prevent this challenge.

The RHR System satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

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**LCO** In MODE 6, with the water level < 23 ft above the top of the reactor vessel flange, both RHR loops must be OPERABLE. Additionally, one loop of RHR must be in operation in order to provide:

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

This LCO is modified by two Notes. Note 1 permits the RHR pumps to be removed from operation for  $\leq 15$  minutes when switching from one train to another. The circumstances for stopping both RHR pumps are to be limited to situations when the outage time is short and the core outlet

BASES

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LCO (continued)

temperature is maintained > 10 degrees F below saturation temperature (e.g., if saturation temperature = 190°F, core outlet temperature must be < 180°F). The Note prohibits boron dilution or draining operations when RHR forced flow is stopped.

Note 2 allows one RHR loop to be inoperable for a period of 2 hours provided the other loop is OPERABLE and in operation. Prior to declaring the loop inoperable, consideration should be given to the existing plant configuration. This consideration should include that the core time to boil is short, there is no draining operation to further reduce RCS water level and that the capability exists to inject borated water into the reactor vessel. This permits surveillance tests to be performed on the inoperable loop during a time when these tests are safe and possible.

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

Both RHR pumps may be aligned to the Refueling Water Storage Tank to support filling or draining the refueling cavity or for performance of required testing.

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APPLICABILITY

Two RHR loops are required to be OPERABLE, and one RHR loop must be in operation in MODE 6, with the water level < 23 ft above the top of the reactor vessel flange, to provide decay heat removal. Requirements for the RHR System in other MODES are covered by LCOs in Section 3.4, Reactor Coolant System (RCS), and Section 3.5, Emergency Core Cooling Systems (ECCS). RHR loop requirements in MODE 6 with the water level  $\geq$  23 ft are located in LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level."

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ACTIONS

A.1 and A.2

If less than the required number of RHR loops are OPERABLE, action shall be immediately initiated and continued until the RHR loop is restored to OPERABLE status or until  $\geq$  23 ft of water level is established above the reactor vessel flange. When the water level is  $\geq$  23 ft above the reactor vessel flange, the Applicability changes to that of LCO 3.9.5, and only one RHR loop is required to be OPERABLE and in operation. An immediate Completion Time is necessary for an operator to initiate corrective actions.

BASES

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ACTIONS (continued)

B.1

If no RHR loop is in operation, there will be no forced circulation to provide mixing to establish uniform boron concentrations. Suspending positive reactivity additions that could result in failure to meet the minimum boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than that what would be required in the RCS for minimum refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation.

B.2

If no RHR loop is in operation, actions shall be initiated immediately, and continued, to restore one RHR loop to operation. Since the unit is in Conditions A and B concurrently, the restoration of two OPERABLE RHR loops and one operating RHR loop should be accomplished expeditiously.

B.3, B.4, B.5.1, and B.5.2

If no RHR is in operation, the following actions must be taken within 4 hours:

- a. The equipment hatch must be closed and secured with four bolts;
- b. One door in each air lock must be closed; and
- c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere must be either closed by a manual or automatic isolation valve, blind flange, or equivalent, or verified to be capable of being closed by an OPERABLE automatic Containment Ventilation isolation valve.

With RHR loop requirements not met, the potential exists for the coolant to boil and release radioactive gas to the containment atmosphere. Performing the actions stated above ensures that all containment penetrations are either closed or can be closed so that the dose limits are not exceeded.

The Completion Time of 4 hours allows fixing of most RHR problems and is reasonable, based on the low probability of the coolant boiling in that time.

BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.9.6.1

This Surveillance demonstrates that one RHR 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. In addition, during operation of the RHR loop with the water level in the vicinity of the reactor vessel nozzles, the RHR pump suction requirements must be met.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.9.6.2

Verification that the required pump is OPERABLE ensures that a RHR 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 Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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REFERENCES

1. UFSAR, Section 5.5.7.
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## B 3.9 REFUELING OPERATIONS

### B 3.9.7 Refueling Cavity Water Level

#### BASES

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**BACKGROUND** The movement of irradiated fuel assemblies 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, refueling canal, fuel transfer canal, refueling cavity, and spent fuel pool. Sufficient water is necessary to retain iodine fission product activity in the water in the event of a fuel handling accident (Refs. 1 and 2). Sufficient iodine activity would be retained to limit offsite doses from the accident to 10 CFR 50.67 limits, further restricted by the guidance of Reference 1.

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**APPLICABLE SAFETY ANALYSES** During movement of irradiated fuel assemblies, the water level in the refueling canal and the refueling cavity is an initial condition design parameter in the analysis of a fuel handling accident in containment, as postulated by Regulatory Guide 1.183 (Ref. 1). A minimum water level of 23 ft (Appendix B of Ref. 1) allows a decontamination factor of 200 (Appendix B of Ref. 1) to be used in the accident analysis for iodine. This relates to the assumption that 99.5% 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 8% I-131, 10% Kr-85, and 5% of other iodines and noble gases of the total fuel rod inventory (Ref. 1).

The fuel handling accident analysis inside containment is described in Reference 2. With a minimum water level of 23 ft and a minimum decay time of 100 hours prior to fuel handling, the analysis and test programs demonstrate that the iodine release due to a postulated fuel handling accident is adequately captured by the water and offsite doses are maintained within allowable limits (Refs. 1 and 3).

Refueling cavity water level satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

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**LCO** A minimum refueling cavity water level of 23 ft above the reactor vessel flange is required to ensure that the radiological consequences of a postulated fuel handling accident inside containment are within acceptable limits, as provided by the guidance of Reference 1.

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**APPLICABILITY** LCO 3.9.7 is applicable when moving irradiated fuel assemblies within containment. The LCO minimizes the possibility of a fuel handling accident in containment that is beyond the assumptions of the safety analysis. If irradiated fuel assemblies are 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 3.7.13, "Spent Fuel Pool Water Level."

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BASES

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ACTIONS

A.1

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

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

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SURVEILLANCE  
REQUIREMENTS

SR 3.9.7.1

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

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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REFERENCES

1. Regulatory Guide 1.183, July 2000.
  2. UFSAR, Section 15.5.6.
  3. 10 CFR 50.67.
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## B 3.9 REFUELING OPERATIONS

### B 3.9.8 Decay Time

#### BASES

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BACKGROUND	The primary purpose of the decay time requirement is to ensure that the fission product inventories assumed in the fuel handling accident analysis are met. As soon as the reactor is subcritical, the quantity of fission products in the core decreases as the fission products undergo natural radioactive decay. As long as the reactor remains subcritical, this decrease will continue and the radiation levels will also decrease.
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APPLICABLE SAFETY ANALYSES	The fuel handling accident is the postulated event of concern in MODE 6 during fuel handling operations (Ref. 1). It establishes the minimum decay time. It is assumed that all of the fuel rods in the equivalent of one fuel assembly are damaged to the extent that all the gap activity in the rods is released. The damaged fuel assembly is assumed to be the assembly with the highest fission product inventory. The fission product inventories are those assumed to be present 100 hours after the reactor becomes subcritical.
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The decay time satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

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LCO	The LCO requires that the reactor be subcritical for at least 100 hours prior to commencing CORE ALTERATIONS. The requirement to be subcritical for greater than or equal to 100 hours ensures that the fission product radioactivity has undergone natural radioactive decay and that the consequences of a fuel handling accident will be within the bounds of the safety analysis.
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APPLICABILITY	This LCO applies during CORE ALTERATIONS, since the potential for a release of fission products exists.
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ACTIONS	<p><u>A.1</u></p> <p>With the reactor subcritical for less than 100 hours, there shall be no operations involving CORE ALTERATIONS. This will preclude a fuel handling accident with fuel containing more fission product radioactivity than assumed in the safety analysis.</p> <p>The immediate Completion Time is consistent with the required times for actions to be performed without delay and in a controlled manner.</p>
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SURVEILLANCE REQUIREMENTS	<p><u>SR 3.9.8.1</u></p> <p>Prior to CORE ALTERATIONS, the reactor must be determined to be subcritical for greater than or equal to 100 hours by verifying the date and time that the reactor achieved subcritical conditions.</p>
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BASES

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REFERENCES      1. UFSAR, Section 15.5.6.

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**ATTACHMENT 2**  
**SEQUOYAH NUCLEAR PLANT, UNIT 2,**  
**TECHNICAL SPECIFICATION BASES**  
**CHANGED PAGES**

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# SEQUOYAH NUCLEAR PLANT

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## B 2.0 SAFETY LIMITS (SLs)

### B 2.1.1 Reactor Core

#### BASES

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**BACKGROUND** GDC 10 (Ref. 1) requires that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). This is accomplished by having a departure from nucleate boiling (DNB) design basis, which corresponds to a 95% probability at a 95% confidence level (the 95/95 DNB criterion) that DNB will not occur and by requiring that fuel centerline temperature stays below the melting temperature.

The restrictions of this SL prevent overheating of the fuel cladding (due to departure from nucleate boiling) and overheating of the fuel pellet (centerline fuel melt (CFM)), either of which could result in cladding perforation, which would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady state peak linear heat rate (LHR) below the level at which fuel centerline melting occurs. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime, where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of DNB and the corresponding significant reduction in heat transfer coefficient from the outer surface of the cladding to the reactor coolant water. Inside the steam film, high cladding temperatures are reached, and a cladding water (zirconium water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB. The DNB correlations have been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

To meet the DNB Design Basis, a statistical core design (SCD) process has been used to develop an appropriate statistical DNBR design limit. Uncertainties in plant operating parameters, nuclear and thermal parameters, and fuel fabrication parameters are considered statistically such that there is at least a 95 percent probability at a 95 percent

## BASES

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### BACKGROUND (continued)

confidence level that the minimum DNBR for the limiting rod is greater than or equal to the DNBR limit. This DNBR uncertainty derived from the SCD analysis, combined with the applicable DNB critical heat flux correlation limit, establishes the statistical DNBR design limit which must be met in plant safety analysis using values of input parameters without adjustment for uncertainty.

Operation above the maximum local linear heat generation rate for fuel melting could result in excessive fuel pellet temperature and cause melting of the fuel at its centerline. Fuel centerline melting occurs when the local LHR, or power peaking, in a region of the fuel is high enough to cause the fuel centerline temperature to reach the melting point of the fuel. Expansion of the pellet upon centerline melting may cause the pellet to stress the cladding to the point of failure, allowing an uncontrolled release of activity to the reactor coolant. The melting point of uranium dioxide varies slightly with burnup. As uranium is depleted and fission products produced, the net effect is a decrease in the melting point. Fuel centerline temperature is not a directly measurable parameter during operation. The maximum local fuel pin centerline temperature is maintained by limiting the local linear heat generation rate in the fuel. The local linear heat generation rate in the fuel is limited so that the maximum fuel centerline temperature will not exceed the acceptance criteria in the safety analysis.

The curves provided in Figure 2.1.1-1 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature for which the minimum DNBR is no less than the safety analysis DNBR limit, or the average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid.

These lines are bounding for all fuel types. The curves provided in Figure 2.1.1-1 are based upon enthalpy rise hot channel factors that result in acceptable DNBR performance of each fuel type. Acceptable DNBR performance is assured by operation within the DNB-based Limiting Safety Limit System Settings (Reactor Trip System trip limits). The plant trip setpoints are verified to be less than the limits defined by the safety limit lines provided in Figure 2.1.1-1 converted from power to delta-temperature and adjusted for uncertainty.

The limiting heat flux conditions for DNB are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion assuming the axial power imbalance or Delta-I ( $\Delta I$ ) is within the limits of the  $f_1(\Delta I)$  function of the Overtemperature Delta Temperature trip. When the axial power imbalance exceeds the tolerance (or deadband) of the  $f_1(\Delta I)$  trip reset function, the Overtemperature Delta Temperature trip setpoint is reduced by the values

BASES

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BACKGROUND (continued)

in the COLR to provide protection required by the core safety limits.

Similarly, the limiting linear heat generation rate conditions for CFM are higher than those calculated for the range of all control rods from fully withdrawn to the maximum allowable control rod insertion assuming the axial power imbalance or Delta-I ( $\Delta I$ ) is within the limits of the  $f_2(\Delta I)$  function of the Overpower-Delta Temperature trip. When the axial power imbalance exceeds the tolerance (or deadband) of the  $f_2(\Delta I)$  trip reset function, the Overpower-Delta Temperature trip setpoint is reduced by the values specified in the COLR to provide protection required by the core safety limits.

The proper functioning of the Reactor Protection System (RPS) and steam generator safety valves prevents violation of the reactor core SLs.

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APPLICABLE  
SAFETY  
ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and AOOs. The reactor core SLs are established to preclude violation of the following fuel design criteria:

- a. There must be at least 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB and
- b. The hot fuel pellet in the core must not experience centerline fuel melting.

The Reactor Trip System setpoints (Ref. 2), in combination with the LCOs, are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System (RCS) temperature, pressure, RCS Flow,  $\Delta I$ , and THERMAL POWER level that would result in a departure from nucleate boiling ratio (DNBR) of less than the DNBR limit and preclude the existence of flow instabilities.

Automatic enforcement of these reactor core SLs is provided by the appropriate operation of the RPS and the steam generator safety valves.

The SLs represent a design requirement for establishing the RPS trip setpoints identified previously. LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," or the assumed initial conditions of the safety analyses (as indicated in the UFSAR, Ref. 2) provide more restrictive limits to ensure that the SLs are not exceeded.

## BASES

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### SAFETY LIMITS

Figure 2.1.1-1 shows the loci of points of THERMAL POWER, RCS pressure, and average temperature for which the minimum DNBR is not less than the safety analyses limit, that fuel centerline temperature remains below melting, that the average enthalpy in the hot leg is less than or equal to the enthalpy of saturated liquid, or that the exit quality is within the limits defined by the DNBR correlation.

The reactor core SLs are established to preclude violation of the following fuel design criteria:

- a. There must be at least a 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB and
- b. There must be at least a 95% probability at a 95% confidence level that the hot fuel pellet in the core does not experience centerline fuel melting.

The reactor core SLs are used to define the various RPS functions such that the above criteria are satisfied during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). To ensure that the RPS precludes the violation of the above criteria, additional criteria are applied to the Overtemperature and Overpower  $\Delta T$  reactor trip functions. That is, it must be demonstrated that the average enthalpy in the hot leg is less than or equal to the saturation enthalpy and that the core exit quality is within the limits defined by the DNBR correlation. Appropriate functioning of the RPS ensures that for variations in the THERMAL POWER, RCS Pressure, RCS average temperature, RCS flow rate, and  $\Delta I$  that the reactor core SLs will be satisfied during steady state operation, normal operational transients, and AOOs.

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### APPLICABILITY

SL 2.1.1 only applies in MODES 1 and 2 because these are the only MODES in which the reactor is critical. Automatic protection functions are required to be OPERABLE during MODES 1 and 2 to ensure operation within the reactor core SLs. The steam generator safety valves or automatic protection actions serve to prevent RCS heatup to the reactor core SL conditions or to initiate a reactor trip function, which forces the unit into MODE 3. Setpoints for the reactor trip functions are specified in LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation." In MODES 3, 4, 5, and 6, Applicability is not required since the reactor is not generating significant THERMAL POWER.

BASES

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SAFETY LIMIT  
VIOLATIONS

The following SL violation responses are applicable to the reactor core SLs. If SL 2.1.1 is violated, the requirement to go to MODE 3 places the unit in a MODE in which this SL is not applicable.

The allowed Completion Time of 1 hour recognizes the importance of bringing the unit to a MODE of operation where this SL is not applicable, and reduces the probability of fuel damage.

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REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.
  2. UFSAR, Section 7.2.
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## B 2.0 SAFETY LIMITS (SLs)

### B 2.1.2 Reactor Coolant System (RCS) Pressure SL

#### BASES

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**BACKGROUND** The SL on RCS pressure protects the integrity of the RCS against overpressurization. In the event of fuel cladding failure, fission products are released into the reactor coolant. The RCS then serves as the primary barrier in preventing the release of fission products into the atmosphere. By establishing an upper limit on RCS pressure, the continued integrity of the RCS is ensured. According to 10 CFR 50, Appendix A, GDC 14, "Reactor Coolant Pressure Boundary," and GDC 15, "Reactor Coolant System Design" (Ref. 1), the reactor coolant pressure boundary (RCPB) design conditions are not to be exceeded during normal operation and anticipated operational occurrences (AOOs). Also, in accordance with GDC 28, "Reactivity Limits" (Ref. 1), reactivity accidents, including rod ejection, do not result in damage to the RCPB greater than limited local yielding.

The design pressure of the RCS is 2500 psia. During normal operation and AOOs, RCS pressure is limited from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME Code (Ref. 2). To ensure system integrity, all RCS components are hydrostatically tested at 125% of design pressure, according to the ASME Code requirements prior to initial operation when there is no fuel in the core. Following inception of unit operation, RCS components shall be pressure tested, in accordance with the requirements of ASME Code, Section XI (Ref. 3).

Overpressurization of the RCS could result in a breach of the RCPB. If such a breach occurs in conjunction with a fuel cladding failure, fission products could enter the containment atmosphere, raising concerns relative to limits on radioactive releases specified in 10 CFR 100, "Reactor Site Criteria" (Ref. 4).

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**APPLICABLE SAFETY ANALYSES** The RCS pressurizer safety valves, the main steam safety valves (MSSVs), and the reactor high pressure trip have settings established to ensure that the RCS pressure SL will not be exceeded.

The RCS pressurizer safety valves are sized to prevent system pressure from exceeding the design pressure by more than 10%, as specified in Section III of the ASME Code for Nuclear Power Plant Components (Ref. 2). The transient that establishes the required relief capacity, and hence valve size requirements and lift settings, is a complete loss of

## BASES

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### APPLICABLE SAFETY ANALYSES (continued)

external load without a direct reactor trip. During the transient, no control actions are assumed, except that the safety valves on the secondary plant are assumed to open when the steam pressure reaches the secondary plant safety valve settings, and nominal feedwater supply is maintained.

The Reactor Trip System setpoints (Ref. 5), together with the settings of the MSSVs, provide pressure protection for normal operation and AOOs. The reactor high pressure trip setpoint is specifically set to provide protection against overpressurization (Ref. 5). The safety analyses for both the high pressure trip and the RCS pressurizer safety valves are performed using conservative assumptions relative to pressure control devices.

More specifically, no credit is taken for operation of any of the following:

- a. Pressurizer power operated relief valves (PORVs);
- b. Steam Dump System;
- c. Reactor Control System;
- d. Pressurizer Level Control System; or
- e. Pressurizer spray valve.

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### SAFETY LIMITS

The maximum transient pressure allowed in the RCS pressure vessel under the ASME Code, Section III, is 110% of design pressure. The maximum transient pressure allowed in the RCS piping, valves, and fittings under USAS, Section B31.1 (Ref. 6) is 120% of design pressure. The most limiting of these two allowances is the 110% of design pressure; therefore, the SL on maximum allowable RCS pressure is 2735 psig.

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APPLICABILITY SL 2.1.2 applies in MODES 1, 2, 3, 4, and 5 because this SL could be approached or exceeded in these MODES due to overpressurization events. The SL is not applicable in MODE 6 because the reactor vessel head closure bolts are not fully tightened, making it unlikely that the RCS can be pressurized.

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SAFETY LIMIT VIOLATIONS If the RCS pressure SL is violated when the reactor is in MODE 1 or 2, the requirement is to restore compliance and be in MODE 3 within 1 hour.

Exceeding the RCS pressure SL may cause immediate RCS failure and create a potential for radioactive releases in excess of 10 CFR 100, "Reactor Site Criteria," limits (Ref. 4).

The allowable Completion Time of 1 hour recognizes the importance of reducing power level to a MODE of operation where the potential for challenges to safety systems is minimized.

If the RCS pressure SL is exceeded in MODE 3, 4, or 5, RCS pressure must be restored to within the SL value within 5 minutes. Exceeding the RCS pressure SL in MODE 3, 4, or 5 is more severe than exceeding this SL in MODE 1 or 2, since the reactor vessel temperature may be lower and the vessel material, consequently, less ductile. As such, pressure must be reduced to less than the SL within 5 minutes. The action does not require reducing MODES, since this would require reducing temperature, which would compound the problem by adding thermal gradient stresses to the existing pressure stress.

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- REFERENCES
1. 10 CFR 50, Appendix A, GDC 14, GDC 15, and GDC 28.
  2. ASME, Boiler and Pressure Vessel Code, Section III, Article NB-7000, 1971.
  3. ASME, Boiler and Pressure Vessel Code, Section XI, Article IWX-5000.
  4. 10 CFR 100.
  5. UFSAR, Section 7.2.
  6. USAS B31.1, Standard Code for Pressure Piping, American Society of Mechanical Engineers, 1967.
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## B 3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

### BASES

LCOs	LCO 3.0.1 through LCO 3.0.9 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.
LCO 3.0.1	LCO 3.0.1 establishes the Applicability statement within each individual Specification as the requirement for when the LCO is required to be met (i.e., when the unit is in the MODES or other specified conditions of the Applicability statement of each Specification).
LCO 3.0.2	<p>LCO 3.0.2 establishes that upon discovery of a failure to meet an LCO, the associated ACTIONS shall be met. The Completion Time of each Required Action for an ACTIONS Condition is applicable from the point in time that an ACTIONS Condition is entered. The Required Actions establish those remedial measures that must be taken within specified Completion Times when the requirements of an LCO are not met. This Specification establishes that:</p> <ol style="list-style-type: none"> <li>a. Completion of the Required Actions within the specified Completion Times constitutes compliance with a Specification; and</li> <li>b. Completion of the Required Actions is not required when an LCO is met within the specified Completion Time, unless otherwise specified.</li> </ol> <p>There are two basic types of Required Actions. The first type of Required Action specifies a time limit in which the LCO must be met. This time limit is the Completion Time to restore an inoperable system or component to OPERABLE status or to restore variables to within specified limits. If this type of Required Action is not completed within the specified Completion Time, a shutdown may be required to place the unit in a MODE or condition in which the Specification is not applicable. (Whether stated as a Required Action or not, correction of the entered Condition is an action that may always be considered upon entering ACTIONS.) The second type of Required Action specifies the remedial measures that permit continued operation of the unit that is not further restricted by the Completion Time. In this case, compliance with the Required Actions provides an acceptable level of safety for continued operation.</p> <p>Completing the Required Actions is not required when an LCO is met or is no longer applicable, unless otherwise stated in the individual Specifications.</p>

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LCO 3.0.2 (continued)

The nature of some Required Actions of some Conditions necessitates that, once the Condition is entered, the Required Actions must be completed even though the associated Conditions no longer exist. The individual LCO's ACTIONS specify the Required Actions where this is the case. An example of this is in LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits."

The Completion Times of the Required Actions are also applicable when a system or component is removed from service intentionally. The reasons for intentionally relying on the ACTIONS include, but are not limited to, performance of Surveillances, preventive maintenance, corrective maintenance, or investigation of operational problems. Entering ACTIONS for these reasons must be done in a manner that does not compromise safety. Intentional entry into ACTIONS should not be made for operational convenience. Additionally, if intentional entry into ACTIONS would result in redundant equipment being inoperable, alternatives should be used instead. Doing so limits the time both subsystems/trains of a safety function are inoperable and limits the time conditions exist which may result in LCO 3.0.3 being entered. Individual Specifications may specify a time limit for performing an SR when equipment is removed from service or bypassed for testing. In this case, the Completion Times of the Required Actions are applicable when this time limit expires, if the equipment remains removed from service or bypassed.

When a change in MODE or other specified condition is required to comply with Required Actions, the unit may enter a MODE or other specified condition in which another Specification becomes applicable. In this case, the Completion Times of the associated Required Actions would apply from the point in time that the new Specification becomes applicable, and the ACTIONS Condition(s) are entered.

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LCO 3.0.3

LCO 3.0.3 establishes the actions that must be implemented when an LCO is not met and:

- a. An associated Required Action and Completion Time is not met and no other Condition applies; or
- b. The condition of the unit is not specifically addressed by the associated ACTIONS. This means that no combination of Conditions stated in the ACTIONS can be made that exactly corresponds to the actual condition of the unit. Sometimes, possible combinations of Conditions are such that entering LCO 3.0.3 is warranted; in such cases, the ACTIONS specifically state a Condition corresponding to such combinations and also that LCO 3.0.3 be entered immediately.

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LCO 3.0.3 (continued)

This Specification delineates the time limits for placing the unit in a safe MODE or other specified condition when operation cannot be maintained within the limits for safe operation as defined by the LCO and its ACTIONS. It is not intended to be used as an operational convenience that permits routine voluntary removal of redundant systems or components from service in lieu of other alternatives that would not result in redundant systems or components being inoperable.

Upon entering LCO 3.0.3, 1 hour is allowed to prepare for an orderly shutdown before initiating a change in unit operation. This includes time to permit the operator to coordinate the reduction in electrical generation with the load dispatcher to ensure the stability and availability of the electrical grid. The time limits specified to reach lower MODES of operation permit the shutdown to proceed in a controlled and orderly manner that is well within the specified maximum cooldown rate and within the capabilities of the unit, assuming that only the minimum required equipment is OPERABLE. This reduces thermal stresses on components of the Reactor Coolant System and the potential for a plant upset that could challenge safety systems under conditions to which this Specification applies. The use and interpretation of specified times to complete the actions of LCO 3.0.3 are consistent with the discussion of Section 1.3, Completion Times.

A unit shutdown required in accordance with LCO 3.0.3 may be terminated and LCO 3.0.3 exited if any of the following occurs:

- a. The LCO is now met;
- b. A Condition exists for which the Required Actions have now been performed; or
- c. ACTIONS exist that do not have expired Completion Times. These Completion Times are applicable from the point in time that the Condition is initially entered and not from the time LCO 3.0.3 is exited.

The time limits of LCO 3.0.3 allow 37 hours for the unit to be in MODE 5 when a shutdown is required during MODE 1 operation. If the unit is in a lower MODE of operation when a shutdown is required, the time limit for reaching the next lower MODE applies. If a lower MODE is reached in less time than allowed, however, the total allowable time to reach MODE 5, or other applicable MODE, is not reduced. For example, if MODE 3 is reached in 2 hours, then the time allowed for reaching MODE 4 is the next 11 hours, because the total time for reaching

## BASES

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### LCO 3.0.3 (continued)

MODE 4 is not reduced from the allowable limit of 13 hours. Therefore, if remedial measures are completed that would permit a return to MODE 1, a penalty is not incurred by having to reach a lower MODE of operation in less than the total time allowed.

In MODES 1, 2, 3, and 4, LCO 3.0.3 provides actions for Conditions not covered in other Specifications. The requirements of LCO 3.0.3 do not apply in MODES 5 and 6 because the unit is already in the most restrictive Condition required by LCO 3.0.3. The requirements of LCO 3.0.3 do not apply in other specified conditions of the Applicability (unless in MODE 1, 2, 3, or 4) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken.

Exceptions to LCO 3.0.3 are provided in instances where requiring a unit shutdown, in accordance with LCO 3.0.3, would not provide appropriate remedial measures for the associated condition of the unit. An example of this is in LCO 3.7.13, "Spent Fuel Pool Water Level." LCO 3.7.13 has an Applicability of "Whenever irradiated fuel assemblies are in the spent fuel pool." Therefore, this LCO can be applicable in any or all MODES. If the LCO and the Required Actions of LCO 3.7.13 are not met while in MODE 1, 2, or 3, there is no safety benefit to be gained by placing the unit in a shutdown condition. The Required Actions of LCO 3.7.13 of "Suspend all movement of fuel assemblies and crane operations with loads in the fuel storage areas" and "Restore spent fuel pool water level to within limit" are the appropriate Required Actions to complete in lieu of the actions of LCO 3.0.3. These exceptions are addressed in the individual Specifications.

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### LCO 3.0.4

LCO 3.0.4 establishes limitations on changes in MODES or other specified conditions in the Applicability when an LCO is not met. It allows placing the unit in a MODE or other specified condition stated in that Applicability (e.g., the Applicability desired to be entered) when unit conditions are such that the requirements of the LCO would not be met, in accordance with LCO 3.0.4.a, LCO 3.0.4.b, or LCO 3.0.4.c.

LCO 3.0.4.a allows entry into a MODE or other specified condition in the Applicability with the LCO not met when the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time. Compliance with Required Actions that permit continued operation of the unit for an unlimited period of time in a MODE or other specified condition provides an acceptable level of safety for continued operation. This is without regard to the status of the unit before or after the MODE change. Therefore, in such cases, entry into a MODE or other specified condition

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### LCO 3.0.4 (continued)

in the Applicability may be made in accordance with the provisions of the Required Actions.

LCO 3.0.4.b allows entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering the MODE or other specified condition in the Applicability, and establishment of risk management actions, if appropriate.

The risk assessment may use quantitative, qualitative, or blended approaches, and the risk assessment will be conducted using the plant program, procedures, and criteria in place to implement 10 CFR 50.65(a)(4), which requires that risk impacts of maintenance activities be assessed and managed. The risk assessment, for the purposes of LCO 3.0.4.b, must take into account all inoperable Technical Specification equipment regardless of whether the equipment is included in the normal 10 CFR 50.65(a)(4) risk assessment scope. The risk assessments will be conducted using the procedures and guidance endorsed by Regulatory Guide 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants." Regulatory Guide 1.182 endorses the guidance in Section 11 of NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." These documents address general guidance for conduct of the risk assessment, quantitative and qualitative guidelines for establishing risk management actions, and example risk management actions. These include actions to plan and conduct other activities in a manner that controls overall risk, increased risk awareness by shift and management personnel, actions to reduce the duration of the condition, actions to minimize the magnitude of risk increases (establishment of backup success paths or compensatory measures), and determination that the proposed MODE change is acceptable. Consideration should also be given to the probability of completing restoration such that the requirements of the LCO would be met prior to the expiration of ACTIONS Completion Times that would require exiting the Applicability.

LCO 3.0.4.b may be used with single, or multiple systems and components unavailable. NUMARC 93-01 provides guidance relative to consideration of simultaneous unavailability of multiple systems and components.

The results of the risk assessment shall be considered in determining the acceptability of entering the MODE or other specified condition in the Applicability, and any corresponding risk management actions. The LCO 3.0.4.b risk assessments do not have to be documented.

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## LCO 3.0.4 (continued)

The Technical Specifications allow continued operation with equipment unavailable in MODE 1 for the duration of the Completion Time. Since this is allowable, and since in general the risk impact in that particular MODE bounds the risk of transitioning into and through the applicable MODES or other specified conditions in the Applicability of the LCO, the use of the LCO 3.0.4.b allowance should be generally acceptable, as long as the risk is assessed and managed as stated above. However, there is a small subset of systems and components that have been determined to be more important to risk and use of the LCO 3.0.4.b allowance is prohibited. The LCOs governing these systems and components contain Notes prohibiting the use of LCO 3.0.4.b by stating that LCO 3.0.4.b is not applicable.

LCO 3.0.4.c allows entry into a MODE or other specified condition in the Applicability with the LCO not met based on a Note in the Specification which states LCO 3.0.4.c is applicable. These specific allowances permit entry into MODES or other specified conditions in the Applicability when the associated ACTIONS to be entered do not provide for continued operation for an unlimited period of time and a risk assessment has not been performed. This allowance may apply to all the ACTIONS or to a specific Required Action of a Specification. The risk assessments performed to justify the use of LCO 3.0.4.b usually only consider systems and components. For this reason, LCO 3.0.4.c is typically applied to Specifications which describe values and parameters (e.g., Containment Air Temperature, Containment Pressure, and Moderator Temperature Coefficient), and may be applied to other Specifications based on NRC plant specific approval.

The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to OPERABLE status before entering an associated MODE or other specified condition in the Applicability.

The provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any unit shutdown. In this context, a unit shutdown is defined as a change in MODE or other specified condition in the Applicability associated with transitioning from MODE 1 to MODE 2, MODE 2 to MODE 3, MODE 3 to MODE 4, and MODE 4 to MODE 5.

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### LCO 3.0.4 (continued)

Upon entry into a MODE or other specified condition in the Applicability with the LCO not met, LCO 3.0.1 and LCO 3.0.2 require entry into the applicable Conditions and Required Actions until the Condition is resolved, until the LCO is met, or until the unit is not within the Applicability of the Technical Specification.

Surveillances do not have to be performed on the associated inoperable equipment (or on variables outside the specified limits), as permitted by SR 3.0.1. Therefore, utilizing LCO 3.0.4 is not a violation of SR 3.0.1 or SR 3.0.4 for any Surveillances that have not been performed on inoperable equipment. However, SRs must be met to ensure OPERABILITY prior to declaring the associated equipment OPERABLE (or variable within limits) and restoring compliance with the affected LCO.

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### LCO 3.0.5

LCO 3.0.5 establishes the allowance for restoring equipment to service under administrative controls when it has been removed from service or declared inoperable to comply with ACTIONS. The sole purpose of this Specification is to provide an exception to LCO 3.0.2 (e.g., to not comply with the applicable Required Action(s)) to allow the performance of required testing to demonstrate:

- a. The OPERABILITY of the equipment being returned to service; or
- b. The OPERABILITY of other equipment.

The administrative controls ensure the time the equipment is returned to service in conflict with the requirements of the ACTIONS is limited to the time absolutely necessary to perform the required testing to demonstrate OPERABILITY. This Specification does not provide time to perform any other preventive or corrective maintenance.

An example of demonstrating the OPERABILITY of the equipment being returned to service is reopening a containment isolation valve that has been closed to comply with Required Actions and must be reopened to perform the required testing.

An example of demonstrating the OPERABILITY of other equipment is taking an inoperable channel or trip system out of the tripped condition to prevent the trip function from occurring during the performance of required testing on another channel in the other trip system. A similar example of demonstrating the OPERABILITY of other equipment is taking an inoperable channel or trip system out of the tripped condition to permit the logic to function and indicate the appropriate response during the performance of required testing on another channel in the same trip system.

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### LCO 3.0.6

LCO 3.0.6 establishes an exception to LCO 3.0.2 for supported systems that have a support system LCO specified in the Technical Specifications (TS). This exception is provided because LCO 3.0.2 would require that the Conditions and Required Actions of the associated inoperable supported system LCO be entered solely due to the inoperability of the support system. This exception is justified because the actions that are required to ensure the unit is maintained in a safe condition are specified in the support system LCO's Required Actions. These Required Actions may include entering the supported system's Conditions and Required Actions or may specify other Required Actions.

When a support system is inoperable and there is an LCO specified for it in the TS, the supported system(s) are required to be declared inoperable if determined to be inoperable as a result of the support system inoperability. However, it is not necessary to enter into the supported systems' Conditions and Required Actions unless directed to do so by the support system's Required Actions. The potential confusion and inconsistency of requirements related to the entry into multiple support and supported systems' LCOs' Conditions and Required Actions are eliminated by providing all the actions that are necessary to ensure the unit is maintained in a safe condition in the support system's Required Actions.

However, there are instances where a support system's Required Action may either direct a supported system to be declared inoperable or direct entry into Conditions and Required Actions for the supported system. This may occur immediately or after some specified delay to perform some other Required Action. Regardless of whether it is immediate or after some delay, when a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.

Specification 5.5.13, "Safety Function Determination Program (SFDP)," ensures loss of safety function is detected and appropriate actions are taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other limitations, remedial actions, or compensatory actions may be identified as a result of the support system inoperability and corresponding exception to entering supported system Conditions and Required Actions. The SFDP implements the requirements of LCO 3.0.6.

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### LCO 3.0.6 (continued)

The following examples use Figure B 3.0-1 to illustrate loss of safety function conditions that may result when a TS support system is inoperable. In this figure, the fifteen systems that comprise Train A are independent and redundant to the fifteen systems that comprise Train B. To correctly use the figure to illustrate the SFDP provisions for a cross train check, the figure establishes a relationship between support and supported systems as follows: the figure shows System 1 as a support system for System 2 and System 3; System 2 as a support system for System 4 and System 5; and System 4 as a support system for System 8 and System 9. Specifically, a loss of safety function may exist when a support system is inoperable and:

- a. A system redundant to system(s) supported by the inoperable support system is also inoperable (EXAMPLE B 3.0.6-1);
- b. A system redundant to system(s) in turn supported by the inoperable supported system is also inoperable (EXAMPLE B 3.0.6-2); or
- c. A system redundant to support system(s) for the supported systems (a) and (b) above is also inoperable (EXAMPLE B 3.0.6-3).

For the following examples, refer to Figure B 3.0-1.

#### EXAMPLE B 3.0.6-1

If System 2 of Train A is inoperable and System 5 of Train B is inoperable, a loss of safety function exists in Systems 5, 10, and 11.

#### EXAMPLE B 3.0.6-2

If System 2 of Train A is inoperable, and System 11 of Train B is inoperable, a loss of safety function exists in System 11.

#### EXAMPLE B 3.0.6-3

If System 2 of Train A is inoperable, and System 1 of Train B is inoperable, a loss of safety function exists in Systems 2, 4, 5, 8, 9, 10 and 11.

If an evaluation determines that a loss of safety function exists, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

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LCO 3.0.6 (continued)

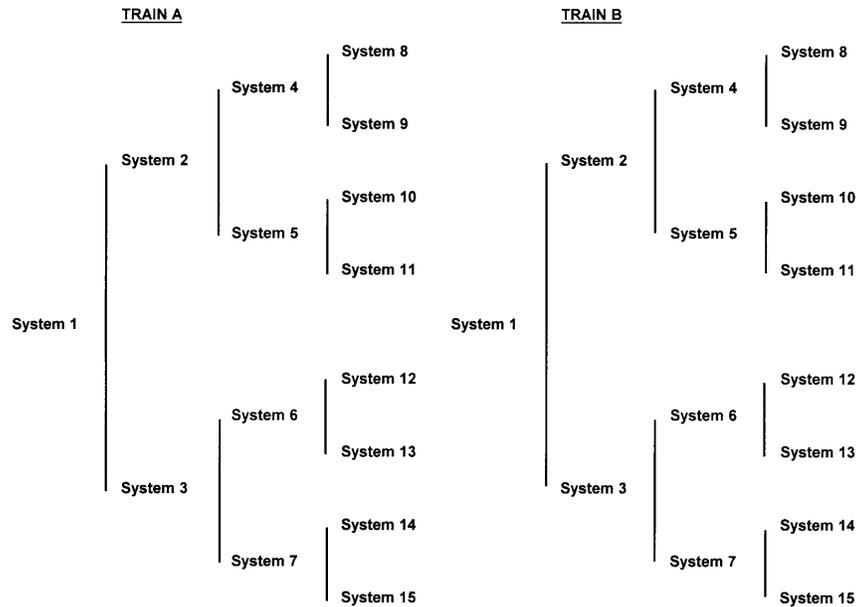


Figure B 3.0-1  
Configuration of Trains and Systems

This loss of safety function does not require the assumption of additional single failures or loss of offsite power. Since operations are being restricted in accordance with the ACTIONS of the support system, any resulting temporary loss of redundancy or single failure protection is taken into account. Similarly, the ACTIONS for inoperable offsite circuit(s) and inoperable diesel generator(s) provide the necessary restriction for cross train inoperabilities. This explicit cross train verification for inoperable AC electrical power sources also acknowledges that supported system(s) are not declared inoperable solely as a result of inoperability of a normal or emergency electrical power source (refer to the definition of OPERABILITY).

When loss of safety function is determined to exist, and the SFDP requires entry into the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists, consideration must be given to the specific type of function affected. Where a loss of function is solely due to a single Technical Specification support system (e.g., loss of automatic start due to inoperable instrumentation, or loss of pump suction

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LCO 3.0.6 (continued)

source due to low tank level) the appropriate LCO is the LCO for the support system. The ACTIONS for a support system LCO adequately address the inoperabilities of that system without reliance on entering its supported system LCO. When the loss of function is the result of multiple support systems, the appropriate LCO is the LCO for the supported system.

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LCO 3.0.7

There are certain special tests and operations required to be performed at various times over the life of the unit. These special tests and operations are necessary to demonstrate select unit performance characteristics, to perform special maintenance activities, and to perform special evolutions. Test Exception LCO 3.1.8, "PHYSICS TEST Exceptions - Mode 2," allows specified Technical Specification (TS) requirements to be changed to permit performances of these special tests and operations, which otherwise could not be performed if required to comply with the requirements of these TS. Unless otherwise specified, all the other TS requirements remain unchanged. This will ensure all appropriate requirements of the MODE or other specified condition not directly associated with or required to be changed to perform the special test or operation will remain in effect.

The Applicability of a Test Exception LCO represents a condition not necessarily in compliance with the normal requirements of the TS. Compliance with Test Exception LCOs is optional. A special operation may be performed either under the provisions of the appropriate Test Exception LCO or under the other applicable TS requirements. If it is desired to perform the special operation under the provisions of the Test Exception LCO, the requirements of the Test Exception LCO shall be followed.

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LCO 3.0.8

LCO 3.0.8 establishes conditions under which systems are considered to remain capable of performing their intended safety function when associated snubbers are not capable of providing their associated support function(s). This LCO states that the supported system is not considered to be inoperable solely due to one or more snubbers not capable of performing their associated support function(s). This is appropriate because a limited length of time is allowed for maintenance, testing, or repair of one or more snubbers not capable of performing their associated support function(s) and appropriate compensatory measures are specified in the snubber requirements, which are located outside of the Technical Specifications (TS) under licensee control. The snubber requirements do not meet the criteria in 10 CFR 50.36(c)(2)(ii), and, as such, are appropriate for control by the licensee.

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### LCO 3.0.8 (continued)

If the allowed time expires and the snubber(s) are unable to perform their associated support function(s), the affected supported system's LCO(s) must be declared not met and the Conditions and Required Actions entered in accordance with LCO 3.0.2.

LCO 3.0.8.a applies when one or more snubbers are not capable of providing their associated support function(s) to a single train or subsystem of a multiple train or subsystem supported system or to a single train or subsystem supported system. LCO 3.0.8.a allows 72 hours to restore the snubber(s) before declaring the supported system inoperable. The 72 hour Completion Time is reasonable based on the low probability of a seismic event concurrent with an event that would require operation of the supported system occurring while the snubber(s) are not capable of performing their associated support function and due to the availability of the redundant train of the supported system.

LCO 3.0.8.b applies when one or more snubbers are not capable of providing their associated support function(s) to more than one train or subsystem of a multiple train or subsystem supported system. LCO 3.0.8.b allows 12 hours to restore the snubber(s) before declaring the supported system inoperable. The 12 hour Completion Time is reasonable based on the low probability of a seismic event concurrent with an event that would require operation of the supported system occurring while the snubber(s) are not capable of performing their associated support function.

LCO 3.0.8 requires that risk be assessed and managed. Industry and NRC guidance on the implementation of 10 CFR 50.65(a)(4) (the Maintenance Rule) does not address seismic risk. However, use of LCO 3.0.8 should be considered with respect to other plant maintenance activities, and integrated into the existing Maintenance Rule process to the extent possible so that maintenance on any unaffected train or subsystem is properly controlled, and emergent issues are properly addressed. The risk assessment need not be quantified, but may be a qualitative awareness of the vulnerability of systems and components when one or more snubbers are not able to perform their associated support function.

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### LCO 3.0.9

LCO 3.0.9 establishes conditions under which systems described in the Technical Specifications are considered to remain OPERABLE when required barriers are not capable of providing their related support function(s).

Barriers are doors, walls, floor plugs, curbs, hatches, installed structures or components, or other devices, not explicitly described in Technical Specifications, that support the performance of the safety function of systems described in the Technical Specifications. This LCO states that the supported system is not considered to be inoperable solely because required barriers not capable of performing their related support function(s) under the described conditions. LCO 3.0.9 allows 30 days before declaring the supported system(s) inoperable and the LCO(s) associated with the supported system(s) not met. A maximum time is placed on each use of this allowance to ensure that required barriers are restored. However, the allowable duration may be less than the specified maximum time based on the risk assessment.

If the allowed time expires and the barriers are unable to perform their related support function(s), the supported system's LCO(s) must be declared not met and the Conditions and Required Actions entered in accordance with LCO 3.0.2.

This provision does not apply to barriers which support ventilation systems or to fire barriers. The Technical Specifications for ventilation systems provide specific Conditions for inoperable barriers. Fire barriers are addressed by other regulatory requirements and associated plant programs. This provision does not apply to barriers that are not required to support system OPERABILITY (see NRC Regulatory Issue Summary 2001-09, "Control of Hazard Barriers," dated April 2, 2001).

The provisions of LCO 3.0.9 are justified because of the low risk associated with required barriers not being capable of performing their related support function. This provision is based on consideration of the following initiating event categories:

- Loss of coolant accidents;
- High energy line breaks;
- Feedwater line breaks;
- Internal flooding;
- External flooding;
- Turbine missile ejection; and
- Tornado or high wind.

BASES

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## LCO 3.0.9 (continued)

The risk impact of the barriers which cannot perform their related support function(s) must be addressed pursuant to the risk assessment and management provision of the Maintenance Rule, 10 CFR 50.65 (a)(4), and the associated implementation guidance, Regulatory Guide 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants." Regulatory Guide 1.182 endorses the guidance in Section 11 of NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." This guidance provides for the consideration of dynamic plant configuration issues, emergent conditions, and other aspects pertinent to plant operation with the barriers unable to perform their related support function(s). These considerations may result in risk management and other compensatory actions being required during the period that barriers are unable to perform their related support function(s).

LCO 3.0.9 may be applied to one or more trains or subsystems of a system supported by barriers that cannot provide their related support function(s), provided that risk is assessed and managed (including consideration of the effects on Large Early Release and from external events). If applied concurrently to more than one train or subsystem of a multiple train or subsystem supported system, the barriers supporting each of these trains or subsystems must provide their related support function(s) for different categories of initiating events. For example, LCO 3.0.9 may be applied for up to 30 days for more than one train of a multiple train supported system if the affected barrier for one train protects against internal flooding and the affected barrier for the other train protects against tornado missiles. In this example, the affected barrier may be the same physical barrier but serve different protection functions for each train.

If during the time that LCO 3.0.9 is being used, the required OPERABLE train or subsystem becomes inoperable, it must be restored to OPERABLE status within 24 hours. Otherwise, the train(s) or subsystem(s) supported by barriers that cannot perform their related support function(s) must be declared inoperable and the associated LCOs declared not met. This 24 hour period provides time to respond to emergent conditions that would otherwise likely lead to entry into LCO 3.0.3 and a rapid plant shutdown, which is not justified given the low probability of an initiating event which would require the barrier(s) not capable of performing their related support function(s). During this 24 hour period, the plant risk associated with the existing conditions is assessed and managed in accordance with 10 CFR 50.65(a)(4).

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B 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

BASES

SRs	SR 3.0.1 through SR 3.0.4 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated. SR 3.0.2 and SR 3.0.3 apply in Chapter 5 only when invoked by a Chapter 5 Specification.
SR 3.0.1	<p>SR 3.0.1 establishes the requirement that SRs must be met during the MODES or other specified conditions in the Applicability for which the requirements of the LCO apply, unless otherwise specified in the individual SRs. This Specification is to ensure that Surveillances are performed to verify the OPERABILITY of systems and components, and that variables are within specified limits. Failure to meet a Surveillance within the specified Frequency, in accordance with SR 3.0.2, constitutes a failure to meet an LCO. Surveillances may be performed by means of any series of sequential, overlapping, or total steps provided the entire Surveillance is performed within the specified Frequency. Additionally, the definitions related to instrument testing (e.g., CHANNEL CALIBRATION) specify that these tests are performed by means of any series of sequential, overlapping, or total steps.</p> <p>Systems and components are assumed to be OPERABLE when the associated SRs have been met. Nothing in this Specification, however, is to be construed as implying that systems or components are OPERABLE when:</p> <ol style="list-style-type: none"> <li>a. The systems or components are known to be inoperable, although still meeting the SRs; or</li> <li>b. The requirements of the Surveillance(s) are known not to be met between required Surveillance performances.</li> </ol> <p>Surveillances do not have to be performed when the unit is in a MODE or other specified condition for which the requirements of the associated LCO are not applicable, unless otherwise specified. The SRs associated with a test exception are only applicable when the test exception is used as an allowable exception to the requirements of a Specification.</p> <p>Unplanned events may satisfy the requirements (including applicable acceptance criteria) for a given SR. In this case, the unplanned event may be credited as fulfilling the performance of the SR. This allowance includes those SRs whose performance is normally precluded in a given MODE or other specified condition.</p> <p>Surveillances, including Surveillances invoked by Required Actions, do not have to be performed on inoperable equipment because the ACTIONS define the remedial measures that apply. Surveillances have</p>

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SR 3.0.1 (continued)

to be met and performed in accordance with SR 3.0.2, prior to returning equipment to OPERABLE status.

Upon completion of maintenance, appropriate post maintenance testing is required to declare equipment OPERABLE. This includes ensuring applicable Surveillances are not failed and their most recent performance is in accordance with SR 3.0.2. Post maintenance testing may not be possible in the current MODE or other specified conditions in the Applicability due to the necessary unit parameters not having been established. In these situations, the equipment may be considered OPERABLE provided testing has been satisfactorily completed to the extent possible and the equipment is not otherwise believed to be incapable of performing its function. This will allow operation to proceed to a MODE or other specified condition where other necessary post maintenance tests can be completed.

Some examples of this process are:

- a. Auxiliary feedwater (AFW) pump turbine maintenance during refueling that requires testing at steam pressures greater than 800 psi. However, if other appropriate testing is satisfactorily completed, the AFW System can be considered OPERABLE. This allows startup and other necessary testing to proceed until the plant reaches the steam pressure required to perform the testing.
- b. High pressure safety injection (SI) maintenance during shutdown that requires system functional tests at a specified pressure. Provided other appropriate testing is satisfactorily completed, startup can proceed with SI considered OPERABLE. This allows operation to reach the specified pressure to complete the necessary post maintenance testing.

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SR 3.0.2

SR 3.0.2 establishes the requirements for meeting the specified Frequency for Surveillances and any Required Action with a Completion Time that requires the periodic performance of the Required Action on a "once per . . ." interval.

SR 3.0.2 permits a 25 percent extension of the interval specified in the Frequency. This extension facilitates Surveillance scheduling and considers plant operating conditions that may not be suitable for conducting the Surveillance (e.g., transient conditions or other ongoing Surveillance or maintenance activities).

When a Section 5.5, "Programs and Manuals," specification states that the provisions of SR 3.0.2 are applicable, a 25 percent extension of the

BASES

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SR 3.0.2 (continued)

testing interval, whether stated in the specification or incorporated by reference, is permitted.

The 25 percent extension does not significantly degrade the reliability that results from performing the Surveillance at its specified Frequency. This is based on the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the SRs. The exceptions to SR 3.0.2 are those Surveillances for which the 25 percent extension of the interval specified in the Frequency does not apply. These exceptions are stated in the individual Specifications. The requirements of regulations take precedence over the TS. An example of where SR 3.0.2 does not apply is in the Containment Leakage Rate Testing Program. This program establishes testing requirements and Frequencies in accordance with the requirements of regulations. The TS cannot in and of themselves extend a test interval specified in the regulations. As stated in SR 3.0.2, the 25 percent extension also does not apply to the initial portion of a periodic Completion Time that requires performance on a "once per ..." basis. The 25 percent extension applies to each performance after the initial performance. The initial performance of the Required Action, whether it is a particular Surveillance or some other remedial action, is considered a single action with a single Completion Time. One reason for not allowing the 25 percent extension to this Completion Time is that such an action usually verifies that no loss of function has occurred by checking the status of redundant or diverse components or accomplishes the function of the inoperable equipment in an alternative manner.

The provisions of SR 3.0.2 are not intended to be used repeatedly merely as an operational convenience to extend Surveillance intervals (other than those consistent with refueling intervals) or periodic Completion Time intervals beyond those specified.

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SR 3.0.3

SR 3.0.3 establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a Surveillance has not been completed within the specified Frequency. A delay period of up to 24 hours or up to the limit of the specified Frequency, whichever is greater, applies from the point in time that it is discovered that the Surveillance has not been performed in accordance with SR 3.0.2, and not at the time that the specified Frequency was not met.

When a Section 5.5, "Programs and Manuals," specification states that the provisions of SR 3.0.3 are applicable, it permits the flexibility to defer declaring the testing requirement not met in accordance with SR 3.0.3 when the testing has not been completed within the testing interval

## BASES

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### SR 3.0.3 (continued)

(including the allowance of SR 3.0.2 if invoked by the Section 5.5 specification).

This delay period provides adequate time to complete Surveillances that have been missed. This delay period permits the completion of a Surveillance before complying with Required Actions or other remedial measures that might preclude completion of the Surveillance.

The basis for this delay period includes consideration of unit conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, the safety significance of the delay in completing the required Surveillance, and the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the requirements.

When a Surveillance with a Frequency based not on time intervals, but upon specified unit conditions, operating situations, or requirements of regulations (e.g., prior to entering MODE 1 after each fuel loading, or in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions, etc.) is discovered to not have been performed when specified, SR 3.0.3 allows for the full delay period of up to the specified Frequency to perform the Surveillance. However, since there is not a time interval specified, the missed Surveillance should be performed at the first reasonable opportunity.

SR 3.0.3 provides a time limit for, and allowances for the performance of, Surveillances that become applicable as a consequence of MODE changes imposed by Required Actions.

Failure to comply with specified Frequencies for SRs is expected to be an infrequent occurrence. Use of the delay period established by SR 3.0.3 is a flexibility which is not intended to be used as an operational convenience to extend Surveillance intervals. While up to 24 hours or the limit of the specified Frequency is provided to perform the missed Surveillance, it is expected that the missed Surveillance will be performed at the first reasonable opportunity. The determination of the first reasonable opportunity should include consideration of the impact on plant risk (from delaying the Surveillance as well as any plant configuration changes required or shutting the plant down to perform the Surveillance) and impact on any analysis assumptions, in addition to unit conditions, planning, availability of personnel, and the time required to perform the Surveillance. This risk impact should be managed through the program in place to implement 10 CFR 50.65(a)(4) and its implementation guidance, NRC Regulatory Guide 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants." This Regulatory Guide addresses consideration of temporary and

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SR 3.0.3 (continued)

aggregate risk impacts, determination of risk management action thresholds, and risk management action up to and including plant shutdown. The missed Surveillance should be treated as an emergent condition as discussed in the Regulatory Guide. The risk evaluation may use quantitative, qualitative, or blended methods. The degree of depth and rigor of the evaluation should be commensurate with the importance of the component. Missed Surveillances for important components should be analyzed quantitatively. If the results of the risk evaluation determine the risk increase is significant, this evaluation should be used to determine the safest course of action. All missed Surveillances will be placed in the licensee's Corrective Action Program.

If a Surveillance is not completed within the allowed delay period, then the equipment is considered inoperable or the variable is considered outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon expiration of the delay period. If a Surveillance is failed within the delay period, then the equipment is inoperable, or the variable is outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon the failure of the Surveillance.

Completion of the Surveillance within the delay period allowed by this Specification, or within the Completion Time of the ACTIONS, restores compliance with SR 3.0.1.

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SR 3.0.4

SR 3.0.4 establishes the requirement that all applicable SRs must be met before entry into a MODE or other specified condition in the Applicability.

This Specification ensures that system and component OPERABILITY requirements and variable limits are met before entry into MODES or other specified conditions in the Applicability for which these systems and components ensure safe operation of the unit. The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to OPERABLE status before entering an associated MODE or other specified condition in the Applicability.

A provision is included to allow entry into a MODE or other specified condition in the Applicability when an LCO is not met due to a Surveillance not being met in accordance with LCO 3.0.4.

However, in certain circumstances, failing to meet an SR will not result in SR 3.0.4 restricting a MODE change or other specified condition change. When a system, subsystem, division, component, device, or variable is inoperable or outside its specified limits, the associated SR(s) are not

BASES

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## SR 3.0.4 (continued)

required to be performed, per SR 3.0.1, which states that surveillances do not have to be performed on inoperable equipment. When equipment is inoperable, SR 3.0.4 does not apply to the associated SR(s) since the requirement for the SR(s) to be performed is removed. Therefore, failing to perform the Surveillance(s) within the specified Frequency does not result in an SR 3.0.4 restriction to changing MODES or other specified conditions of the Applicability. However, since the LCO is not met in this instance, LCO 3.0.4 will govern any restrictions that may (or may not) apply to MODE or other specified condition changes. SR 3.0.4 does not restrict changing MODES or other specified conditions of the Applicability when a Surveillance has not been performed within the specified Frequency, provided the requirement to declare the LCO not met has been delayed in accordance with SR 3.0.3.

The provisions of SR 3.0.4 shall not prevent entry into MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of SR 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any unit shutdown. In this context, a unit shutdown is defined as a change in MODE or other specified condition in the Applicability associated with transitioning from MODE 1 to MODE 2, MODE 2 to MODE 3, MODE 3 to MODE 4, and MODE 4 to MODE 5.

The precise requirements for performance of SRs are specified such that exceptions to SR 3.0.4 are not necessary. The specific time frames and conditions necessary for meeting the SRs are specified in the Frequency, in the Surveillance, or both. This allows performance of Surveillances when the prerequisite condition(s) specified in a Surveillance procedure require entry into the MODE or other specified condition in the Applicability of the associated LCO prior to the performance or completion of a Surveillance. A Surveillance that could not be performed until after entering the LCO's Applicability, would have its Frequency specified such that it is not "due" until the specific conditions needed are met. Alternately, the Surveillance may be stated in the form of a Note, as not required (to be met or performed) until a particular event, condition, or time has been reached. Further discussion of the specific formats of SRs' annotation is found in Section 1.4, Frequency.

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## B 3.1 REACTIVITY CONTROL SYSTEMS

### B 3.1.1 SHUTDOWN MARGIN (SDM)

#### BASES

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BACKGROUND	<p>According to GDC 26 (Ref. 1), the reactivity control systems must be redundant and capable of holding the reactor core subcritical when shut down under cold conditions. Maintenance of the SDM ensures that postulated reactivity events will not damage the fuel.</p> <p>SDM requirements provide sufficient reactivity margin to ensure that acceptable fuel design limits will not be exceeded for normal shutdown and anticipated operational occurrences (AOOs). As such, the SDM defines the degree of subcriticality that would be obtained immediately following the insertion or scram of all shutdown and control rods, assuming that the single rod cluster assembly of highest reactivity worth is fully withdrawn.</p> <p>The system design requires that two independent reactivity control systems be provided, and that one of these systems be capable of maintaining the core subcritical under cold conditions. These requirements are provided by the use of movable control assemblies and soluble boric acid in the Reactor Coolant System (RCS). The Control Rod System can compensate for the reactivity effects of the fuel and water temperature changes accompanying power level changes over the range from full load to no load. In addition, the Control Rod System, together with the boration system, provides the SDM during power operation and is capable of making the core subcritical rapidly enough to prevent exceeding acceptable fuel damage limits, assuming that the rod of highest reactivity worth remains fully withdrawn. The soluble boron system can compensate for fuel depletion during operation and all xenon burnout reactivity changes and maintain the reactor subcritical under cold conditions.</p> <p>During power operation, SDM control is ensured by operating with the shutdown banks fully withdrawn and the control banks within the limits of LCO 3.1.6, "Control Bank Insertion Limits." When the unit is in the shutdown and refueling modes, the SDM requirements are met by means of adjustments to the RCS boron concentration.</p>
APPLICABLE SAFETY ANALYSES	<p>The minimum required SDM is assumed as an initial condition in safety analyses. The safety analysis (Ref. 2) establishes a SDM that ensures specified acceptable fuel design limits are not exceeded for normal operation and AOOs, with the assumption of the highest worth rod stuck out on scram. For MODE 5, the primary safety analysis that relies on the SDM limits is the boron dilution analysis.</p>

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BASES

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## APPLICABLE SAFETY ANALYSES (continued)

The acceptance criteria for the SDM requirements are that specified acceptable fuel design limits are maintained. This is done by ensuring that:

- a. The reactor can be made subcritical from all operating conditions, transients, and Design Basis Events,
- b. The reactivity transients associated with postulated accident conditions are controllable within acceptable limits (departure from nucleate boiling ratio (DNBR), fuel centerline temperature limits for AOOs, and  $\leq 280$  cal/gm fuel energy deposition for the rod ejection accident), and
- c. The reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

The most limiting accident for the SDM requirements is based on a main steam line break (MSLB), as described in the accident analysis (Ref. 2). The increased steam flow resulting from a pipe break in the main steam system causes an increased energy removal from the affected steam generator (SG), and consequently the RCS. This results in a reduction of the reactor coolant temperature. The resultant coolant shrinkage causes a reduction in pressure. In the presence of a negative moderator temperature coefficient, this cooldown causes an increase in core reactivity. As RCS temperature decreases, the severity of an MSLB decreases until the MODE 5 value is reached. The most limiting MSLB, with respect to potential fuel damage before a reactor trip occurs, is a double ended break of a main steam line inside containment initiated at the end of core life. The positive reactivity addition from the moderator temperature decrease will terminate when the affected SG boils dry, thus terminating RCS heat removal and cooldown. Following the MSLB, a post trip return to power may occur; however, no fuel damage occurs as a result of the post trip return to power, and THERMAL POWER does not violate the Safety Limit (SL) requirement of SL 2.1.1.

In addition to the limiting MSLB transient, the SDM requirements must also protect against:

- a. Inadvertent boron dilution,
- b. An uncontrolled rod withdrawal from subcritical or low power condition,

## BASES

## APPLICABLE SAFETY ANALYSES (continued)

- c. Startup of an inactive reactor coolant pump (RCP), and
- d. Rod ejection.

Each of these events is discussed below.

In the boron dilution analysis, the required SDM defines the reactivity difference between an initial subcritical boron concentration and the corresponding critical boron concentration. These values, in conjunction with the configuration of the RCS and the assumed dilution flow rate, directly affect the results of the analysis. This event is most limiting at the beginning of core life, when critical boron concentrations are highest.

Depending on the system initial conditions and reactivity insertion rate, the uncontrolled rod withdrawal transient is terminated by either a high power level trip or an overtemperature  $\Delta T$  trip. In all cases, power level, RCS pressure, linear heat rate, and the DNBR do not exceed allowable limits.

The startup of an inactive RCP will not result in a "cold water" criticality, even if the maximum difference in temperature exists between the SG and the core. The maximum positive reactivity addition that can occur due to an inadvertent RCP start is less than half the minimum required SDM. Startup of an idle RCP cannot, therefore, produce a return to power from the hot standby condition.

The ejection of a control rod rapidly adds reactivity to the reactor core, causing both the core power level and heat flux to increase with corresponding increases in reactor coolant temperatures and pressure. The ejection of a rod also produces a time dependent redistribution of core power.

SDM satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii). Even though it is not directly observed from the control room, SDM is considered an initial condition process variable because it is periodically monitored to ensure that the unit is operating within the bounds of accident analysis assumptions.

## LCO

SDM is a core design condition that can be ensured during operation through control rod positioning (control and shutdown banks) and through the soluble boron concentration.

The MSLB (Ref. 2) and the boron dilution (Ref. 3) accidents are the most limiting analyses that establish the SDM value of the LCO. For MSLB accidents, if the LCO is violated, there is a potential to exceed the DNBR limit and to exceed 10 CFR 100, "Reactor Site Criteria," limits (Ref. 4).

BASES

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## LCO (continued)

For the boron dilution accident, if the LCO is violated, the minimum required time assumed for operator action to terminate dilution may no longer be applicable.

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## APPLICABILITY

In MODE 2 with  $k_{\text{eff}} < 1.0$  and in MODES 3, 4, and 5, the SDM requirements are applicable to provide sufficient negative reactivity to meet the assumptions of the safety analyses discussed above. In MODE 6, the shutdown reactivity requirements are given in LCO 3.9.1, "Boron Concentration." In MODES 1 and 2, SDM is ensured by complying with LCO 3.1.5, "Shutdown Bank Insertion Limits," and LCO 3.1.6.

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## ACTIONS

A.1

If the SDM requirements are not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. It is assumed that boration will be continued until the SDM requirements are met.

In the determination of the required combination of boration flow rate and boron concentration, there is no unique requirement that must be satisfied. Since it is imperative to raise the boron concentration of the RCS as soon as possible, the boron concentration should be a highly concentrated solution, such as that normally found in the boric acid tank, or the refueling water storage tank. The operator should borate with the best source available for the plant conditions.

In determining the boration flow rate, the time in core life must be considered. For instance, the most difficult time in core life to increase the RCS boron concentration is at the beginning of cycle when the boron concentration may approach or exceed 2000 ppm. Assuming that a value of 1%  $\Delta k/k$  (1000 pcm) must be recovered and a boration flow rate of 50 gpm, it is possible to increase the boron concentration of the RCS by 156 ppm in approximately 48 minutes. If a boron worth of 6.4 pcm/ppm is assumed, this combination will increase the SDM by 1%  $\Delta k/k$  or 1000 pcm. These boration parameters represent Sequoyah typical values and are provided for the purpose of offering a specific example.

BASES

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SURVEILLANCE  
REQUIREMENTSSR 3.1.1.1

In MODES 1 and 2 with  $k_{\text{eff}} \geq 1.0$ , SDM is verified by observing that the requirements of LCO 3.1.5 and LCO 3.1.6 are met. In the event that a rod is known to be untrippable, however, SDM verification must account for the worth of the untrippable rod as well as another rod of maximum worth.

In MODE 2 with  $k_{\text{eff}} < 1.0$  and in MODES 3, 4, and 5, the SDM is verified by performing a reactivity balance calculation, considering the listed reactivity effects:

- a. RCS boron concentration,
- b. Control bank position,
- c. RCS average temperature,
- d. Fuel burnup based on gross thermal energy generation,
- e. Xenon concentration,
- f. Samarium concentration, and
- g. Isothermal temperature coefficient (ITC).

Using the ITC accounts for Doppler reactivity in this calculation because the reactor is subcritical, and the fuel temperature will be changing at the same rate as the RCS.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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## REFERENCES

1. 10 CFR 50, Appendix A, GDC 26.
  2. UFSAR, Section 15.4.2.
  3. UFSAR, Section 15.2.4.
  4. 10 CFR 100.
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## B 3.1 REACTIVITY CONTROL SYSTEMS

### B 3.1.2 Core Reactivity

#### BASES

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**BACKGROUND** According to GDC 26, GDC 28, and GDC 29 (Ref. 1), reactivity shall be controllable, such that subcriticality is maintained under cold conditions, and acceptable fuel design limits are not exceeded during normal operation and anticipated operational occurrences. Therefore, reactivity balance is used as a measure of the predicted versus measured core reactivity during power operation. The periodic confirmation of core reactivity is necessary to ensure that Design Basis Accident (DBA) and transient safety analyses remain valid. A large reactivity difference could be the result of unanticipated changes in fuel, control rod worth, or operation at conditions not consistent with those assumed in the predictions of core reactivity, and could potentially result in a loss of SDM or violation of acceptable fuel design limits. Comparing predicted versus measured core reactivity validates the nuclear methods used in the safety analysis and supports the SDM demonstrations (LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") in ensuring the reactor can be brought safely to cold, subcritical conditions.

When the reactor core is critical or in normal power operation, a reactivity balance exists and the net reactivity is zero. A comparison of predicted and measured reactivity is convenient under such a balance, since parameters are being maintained relatively stable under steady state power conditions. The positive reactivity inherent in the core design is balanced by the negative reactivity of the control components, thermal feedback, neutron leakage, and materials in the core that absorb neutrons, such as burnable absorbers producing zero net reactivity. Excess reactivity can be inferred from the boron letdown curve (or critical boron curve), which provides an indication of the soluble boron concentration in the Reactor Coolant System (RCS) versus cycle burnup. Periodic measurement of the RCS boron concentration for comparison with the predicted value with specific variables (such as rod height, temperature, pressure, and power), provides a convenient method of ensuring that core reactivity is within design expectations and that the calculational models used to generate the safety analysis are adequate.

In order to achieve the required fuel cycle energy output, the uranium enrichment, in the new fuel loading and in the fuel remaining from the previous cycle, provides excess positive reactivity beyond that required to sustain steady state operation throughout the cycle. When the reactor is critical at RTP and moderator temperature, the excess positive reactivity is compensated by burnable absorbers (if any), control rods, whatever neutron poisons (mainly xenon and samarium) are present in the fuel, and the RCS boron concentration.

## BASES

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### BACKGROUND (continued)

When the core is producing THERMAL POWER, the fuel is being depleted and excess reactivity is decreasing. As the fuel depletes, the RCS boron concentration is reduced to decrease negative reactivity and maintain constant THERMAL POWER. The boron letdown curve is based on steady state operation at RTP. Therefore, deviations from the predicted boron letdown curve may indicate deficiencies in the design analysis, deficiencies in the calculational models, or abnormal core conditions, and must be evaluated.

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### APPLICABLE SAFETY ANALYSES

The acceptance criteria for core reactivity are that the reactivity balance limit ensures plant operation is maintained within the assumptions of the safety analyses.

Accurate prediction of core reactivity is either an explicit or implicit assumption in the accident analysis evaluations. Every accident evaluation (Ref. 2) is, therefore, dependent upon accurate evaluation of core reactivity. In particular, SDM and reactivity transients, such as control rod withdrawal accidents or rod ejection accidents, are very sensitive to accurate prediction of core reactivity. These accident analysis evaluations rely on computer codes that have been qualified against available test data, operating plant data, and analytical benchmarks. Monitoring reactivity balance additionally ensures that the nuclear methods provide an accurate representation of the core reactivity.

Design calculations and safety analyses are performed for each fuel cycle for the purpose of predetermining reactivity behavior and the RCS boron concentration requirements for reactivity control during fuel depletion.

The comparison between measured and predicted initial core reactivity provides a normalization for the calculational models used to predict core reactivity. If the measured and predicted RCS boron concentrations for identical core conditions at beginning of cycle life (BOL) do not agree, then the assumptions used in the reload cycle design analysis or the calculational models used to predict soluble boron requirements may not be accurate. If reasonable agreement between measured and predicted core reactivity exists at BOL, then the prediction may be normalized to the measured boron concentration. Thereafter, any significant deviations in the measured boron concentration from the predicted boron letdown curve that develop during fuel depletion may be an indication that the calculational model is not adequate for core burnups beyond BOL, or that an unexpected change in core conditions has occurred.

BASES

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APPLICABLE SAFETY ANALYSES (continued)

The normalization of predicted RCS boron concentration to the measured value is typically performed after reaching RTP following startup from a refueling outage, with the control rods in their normal positions for power operation. The normalization is performed at BOL conditions, so that core reactivity relative to predicted values can be continually monitored and evaluated as core conditions change during the cycle.

Core reactivity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

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LCO

Long term core reactivity behavior is a result of the core physics design and cannot be easily controlled once the core design is fixed. During operation, therefore, the LCO can only be ensured through measurement and tracking, and appropriate actions taken as necessary. Large differences between actual and predicted core reactivity may indicate that the assumptions of the DBA and transient analyses are no longer valid, or that the uncertainties in the Nuclear Design Methodology are larger than expected. A limit on the reactivity balance of  $\pm 1\% \Delta k/k$  has been established based on engineering judgment. A 1% deviation in reactivity from that predicted is larger than expected for normal operation and should therefore be evaluated.

When measured core reactivity is within  $1\% \Delta k/k$  of the predicted value at steady state thermal conditions, the core is considered to be operating within acceptable design limits. Since deviations from the limit are normally detected by comparing predicted and measured steady state RCS critical boron concentrations, the difference between measured and predicted values would be approximately 100 ppm (depending on the boron worth) before the limit is reached. These values are well within the uncertainty limits for analysis of boron concentration samples, so that spurious violations of the limit due to uncertainty in measuring the RCS boron concentration are unlikely.

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APPLICABILITY

The limits on core reactivity must be maintained during MODES 1 and 2 because a reactivity balance must exist when the reactor is critical or producing THERMAL POWER. As the fuel depletes, core conditions are changing, and confirmation of the reactivity balance ensures the core is operating as designed. This Specification does not apply in MODES 3, 4, and 5 because the reactor is shut down and the reactivity balance is not changing.

In MODE 6, fuel loading results in a continually changing core reactivity. Boron concentration requirements (LCO 3.9.1, "Boron Concentration") ensure that fuel movements are performed within the bounds of the safety analysis. A SDM demonstration is required during the first startup following operations that could have altered core reactivity (e.g., fuel movement, control rod replacement, control rod shuffling).

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BASES

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ACTIONS

A.1 and A.2

Should an anomaly develop between measured and predicted core reactivity, an evaluation of the core design and safety analysis must be performed. Core conditions are evaluated to determine their consistency with input to design calculations. Measured core and process parameters are evaluated to determine that they are within the bounds of the safety analysis, and safety analysis calculational models are reviewed to verify that they are adequate for representation of the core conditions. The required Completion Time of 7 days is based on the low probability of a DBA occurring during this period, and allows sufficient time to assess the physical condition of the reactor and complete the evaluation of the core design and safety analysis.

Following evaluations of the core design and safety analysis, the cause of the reactivity anomaly may be resolved. If the cause of the reactivity anomaly is a mismatch in core conditions at the time of RCS boron concentration sampling, then a recalculation of the RCS boron concentration requirements may be performed to demonstrate that core reactivity is behaving as expected. If an unexpected physical change in the condition of the core has occurred, it must be evaluated and corrected, if possible. If the cause of the reactivity anomaly is in the calculation technique, then the calculational models must be revised to provide more accurate predictions. If any of these results are demonstrated, and it is concluded that the reactor core is acceptable for continued operation, then the boron letdown curve may be renormalized and power operation may continue. If operational restriction or additional SRs are necessary to ensure the reactor core is acceptable for continued operation, then they must be defined.

The required Completion Time of 7 days is adequate for preparing whatever operating restrictions or Surveillances that may be required to allow continued reactor operation.

B.1

If the core reactivity cannot be restored to within the 1%  $\Delta k/k$  limit, 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. If the SDM for MODE 3 is not met, then the boration required by SR 3.1.1.1 would occur. The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.1.2.1

Core reactivity is verified by periodic comparisons of measured and predicted RCS boron concentrations. The comparison is made, considering that other core conditions are fixed or stable, including control rod position, moderator temperature, fuel temperature, fuel depletion, xenon concentration, and samarium concentration. The Surveillance is performed prior to entering MODE 1 as an initial check on core conditions and design calculations at BOL. The SR is modified by a Note. The Note indicates that the normalization of predicted core reactivity to the measured value may take place, if required, within the first 60 effective full power days (EFPD) after each fuel loading. This allows sufficient time for core conditions to reach steady state, but prevents operation for a large fraction of the fuel cycle without establishing a benchmark for the design calculations. The required subsequent Frequency, following the initial 60 EFPD after entering MODE 1, is acceptable, based on the slow rate of core changes due to fuel depletion and the presence of other indicators (QPTR, AFD, etc.) for prompt indication of an anomaly.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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REFERENCES

1. 10 CFR 50, Appendix A, GDC 26, GDC 28, and GDC 29.
  2. UFSAR, Chapter 15.
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## B 3.1 REACTIVITY CONTROL SYSTEMS

### B 3.1.3 Moderator Temperature Coefficient (MTC)

#### BASES

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**BACKGROUND** According to GDC 11 (Ref. 1), the reactor core and its interaction with the Reactor Coolant System (RCS) must be designed for inherently stable power operation, even in the possible event of an accident. In particular, the net reactivity feedback in the system must compensate for any unintended reactivity increases.

The MTC relates a change in core reactivity to a change in reactor coolant temperature (a positive MTC means that reactivity increases with increasing moderator temperature; conversely, a negative MTC means that reactivity decreases with increasing moderator temperature). The reactor is designed to operate with a negative MTC over the largest possible range of fuel cycle operation. Therefore, a coolant temperature increase will cause a reactivity decrease, so that the coolant temperature tends to return toward its initial value. Reactivity increases that cause a coolant temperature increase will thus be self limiting, and stable power operation will result.

MTC values are predicted at selected burnups during the safety evaluation analysis and are confirmed to be acceptable by measurements. Both initial and reload cores are designed so that the beginning of cycle life (BOL) MTC is less than zero when THERMAL POWER is at RTP. The actual value of the MTC is dependent on core characteristics, such as fuel loading and reactor coolant soluble boron concentration. The core design may require additional fixed distributed poisons to yield an MTC at BOL within the range analyzed in the plant accident analysis. The end of cycle life (EOL) MTC is also limited by the requirements of the accident analysis. Fuel cycles that are designed to achieve high burnups or that have changes to other characteristics are evaluated to ensure that the MTC does not exceed the EOL limit.

The limitations on MTC are provided to ensure that the value of this coefficient remains within the limiting conditions assumed in the UFSAR accident and transient analyses.

BASES

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## BACKGROUND (continued)

If the LCO limits are not met, the unit response during transients may not be as predicted. The core could violate criteria that prohibit a return to criticality, or the departure from nucleate boiling ratio criteria of the approved correlation may be violated, which could lead to a loss of the fuel cladding integrity.

The SRs for measurement of the MTC at the beginning and near the end of the fuel cycle are adequate to confirm that the MTC remains within its limits, since this coefficient changes slowly, due principally to the reduction in RCS boron concentration associated with fuel burnup.

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APPLICABLE  
SAFETY  
ANALYSES

The acceptance criteria for the specified MTC are:

- a. The MTC values must remain within the bounds of those used in the accident analysis (Ref. 2) and
- b. The MTC must be such that inherently stable power operations result during normal operation and accidents, such as overheating and overcooling events.

The UFSAR, Chapter 15 (Ref. 2), contains analyses of accidents that result in both overheating and overcooling of the reactor core. MTC is one of the controlling parameters for core reactivity in these accidents. Both the most positive value and most negative value of the MTC are important to safety, and both values must be bounded. Values used in the analyses consider worst case conditions to ensure that the accident results are bounding (Ref. 3).

The consequences of accidents that cause core overheating must be evaluated when the MTC is positive. Such accidents include the rod withdrawal transient from either zero (Ref. 4) or RTP, loss of main feedwater flow, and loss of forced reactor coolant flow. The consequences of accidents that cause core overcooling must be evaluated when the MTC is negative. Such accidents include sudden feedwater flow increase and sudden decrease in feedwater temperature.

BASES

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## APPLICABLE SAFETY ANALYSES (continued)

In order to ensure a bounding accident analysis, the MTC is assumed to be its most limiting value for the analysis conditions appropriate to each accident. The bounding value is determined by considering rodged and unrodged conditions, whether the reactor is at full or zero power, and whether it is the BOL or EOL. The most conservative combination appropriate to the accident is then used for the analysis (Ref. 2).

MTC values are bounded in reload safety evaluations assuming steady state conditions at BOL and EOL. An EOL measurement is conducted at conditions when the RCS boron concentration reaches approximately 300 ppm. The measured value may be extrapolated to project the EOL value, in order to confirm reload design predictions.

MTC satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii). Even though it is not directly observed and controlled from the control room, MTC is considered an initial condition process variable because of its dependence on boron concentration.

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## LCO

LCO 3.1.3 requires the MTC to be maintained within specified limits of the COLR to ensure that the core operates within the assumptions of the accident analysis. During the reload core safety evaluation, the MTC is analyzed to determine that its values remain within the bounds of the original accident analysis during operation.

Assumptions made in safety analyses require that the MTC be less positive than a given upper bound and more positive than a given lower bound. The MTC is most positive at BOL; this upper bound must not be exceeded. This maximum upper limit occurs at BOL, all rods out (ARO), hot zero power conditions. At EOL the MTC takes on its most negative value, when the lower bound becomes important. This LCO exists to ensure that both the upper and lower bounds are not exceeded.

During operation, therefore, the conditions of the LCO can only be ensured through measurement. The Surveillance checks at BOL and EOL on MTC provide confirmation that the MTC is behaving as anticipated so that the acceptance criteria are met.

The LCO establishes a maximum positive value that cannot be exceeded. The BOL positive limit and the EOL negative limit are established in the COLR to allow specifying limits for each particular cycle. This permits the unit to take advantage of improved fuel management and changes in unit operating schedule.

BASES

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APPLICABILITY      Technical Specifications place both LCO and SR values on MTC, based on the safety analysis assumptions described above.

In MODE 1, the limits on MTC must be maintained to ensure that any accident initiated from THERMAL POWER operation will not violate the design assumptions of the accident analysis. In MODE 2 with the reactor critical, the BOL limit must be maintained to ensure that startup and subcritical accidents (such as the uncontrolled control rod assembly or group withdrawal) will not violate the assumptions of the accident analysis. The EOL MTC limit must be maintained in MODES 2 and 3, in addition to MODE 1, to ensure that cooldown accidents will not violate the assumptions of the accident analysis. In MODES 4, 5, and 6, this LCO is not applicable, since no Design Basis Accidents using the MTC as an analysis assumption are initiated from these MODES.

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## ACTIONS

A.1

If the BOL MTC limit is violated, administrative withdrawal limits for control banks must be established to maintain the MTC within its limits. The MTC becomes more negative with control bank insertion and decreased boron concentration. A Completion Time of 24 hours provides enough time for evaluating the MTC measurement and computing the required bank withdrawal limits.

As cycle burnup is increased, the RCS boron concentration will be reduced. The reduced boron concentration causes the MTC to become more negative. Using physics calculations, the time in cycle life at which the calculated MTC will meet the LCO requirement can be determined. At this point in core life Condition A no longer exists. The unit is no longer in the Required Action, so the administrative withdrawal limits are no longer in effect.

B.1

If the required administrative withdrawal limits at BOL are not established within 24 hours, the unit must be brought to at least MODE 2 with  $k_{\text{eff}} < 1.0$  to prevent operation with an MTC that is more positive than that assumed in safety analyses.

The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

BASES

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ACTIONS (continued)

C.1

Exceeding the EOL MTC limit means that the safety analysis assumptions for the EOL accidents that use a bounding negative MTC value may be invalid. If the EOL MTC limit is exceeded, the plant must be brought to a MODE or condition in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least MODE 4 within 12 hours.

The allowed Completion Time is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

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SURVEILLANCE  
REQUIREMENTS

SR 3.1.3.1

This SR requires measurement of the MTC at BOL prior to entering MODE 1 in order to demonstrate compliance with the most positive MTC LCO. Meeting the limit prior to entering MODE 1 ensures that the limit will also be met at higher power levels.

The BOL MTC value for ARO will be inferred from isothermal temperature coefficient measurements obtained during the physics tests after refueling. The ARO value can be directly compared to the BOL MTC limit of the LCO. If required, measurement results and predicted design values can be used to establish administrative withdrawal limits for control banks.

## BASES

## SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.3.2

In similar fashion, the LCO demands that the MTC be less negative than the specified value for EOL full power conditions. This measurement may be performed at any THERMAL POWER, but its results must be extrapolated to the conditions of RTP and all banks withdrawn in order to make a proper comparison with the LCO value. Because the RTP MTC value will gradually become more negative with further core depletion and boron concentration reduction, a 300 ppm SR value of MTC should necessarily be less negative than the EOL LCO limit. The 300 ppm SR value is sufficiently less negative than the EOL LCO limit value to ensure that the LCO limit will be met when the 300 ppm Surveillance criterion is met.

SR 3.1.3.2 is modified by three Notes that include the following requirements:

- a. The SR is not required to be performed until 7 effective full power days (EFPDs) after reaching the equivalent of an equilibrium RTP all rods out (ARO) boron concentration of 300 ppm.
- b. If the 300 ppm Surveillance limit is not met, it is possible that the EOL limit on MTC could be reached before the planned EOL. Because the MTC changes slowly with core depletion, the Frequency of 14 effective full power days is sufficient to avoid exceeding the EOL limit on MTC.
- c. If the measured MTC at 60 ppm is more positive than the 60 ppm Surveillance limit, the EOL limit on MTC will not be exceeded because of the gradual manner in which MTC changes with core burnup. The 60 ppm Surveillance is only performed if the 300 ppm Surveillance limit was not met (see note b). If the 60 ppm Surveillance limit is met, no further Surveillance of EOL MTC is required for the remainder of the fuel cycle. If the 60 ppm Surveillance limit is not met, then Surveillance of EOL MTC is required for the remainder of the fuel cycle as described in note b.

## REFERENCES

1. 10 CFR 50, Appendix A, GDC 11.
2. UFSAR, Chapter 15.
3. BAW 10169P-A, "B&W Safety Analysis Methodology for Recirculating Steam Generator Plants," October 1989.
4. UFSAR, Section 15.2.1.

## B 3.1 REACTIVITY CONTROL SYSTEMS

### B 3.1.4 Rod Group Alignment Limits

#### BASES

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**BACKGROUND** The OPERABILITY (i.e., trippability) of the shutdown and control rods is an initial assumption in all safety analyses that assume rod insertion upon reactor trip. Maximum rod misalignment is an initial assumption in the safety analysis that directly affects core power distributions and assumptions of available SDM.

The applicable criteria for these reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10, "Reactor Design," GDC 26, "Reactivity Control System Redundancy and Capability" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plants" (Ref. 2).

Mechanical or electrical failures may cause a control or shutdown rod to become inoperable or to become misaligned from its group. Rod inoperability or misalignment may cause increased power peaking, due to the asymmetric reactivity distribution and a reduction in the total available rod worth for reactor shutdown. Therefore, rod alignment and OPERABILITY are related to core operation in design power peaking limits and the core design requirement of a minimum SDM.

Limits on rod alignment have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

Rod cluster control assemblies (RCCAs), or rods, are moved by their control rod drive mechanisms (CRDMs). Each CRDM moves its RCCA one step (approximately 5/8 inch) at a time, but at varying rates (steps per minute) depending on the signal output from the Rod Control System.

The RCCAs are divided among control banks and shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control. A group consists of two or more RCCAs that are electrically paralleled to step simultaneously. If a bank of RCCAs consists of two groups, the groups are moved in a staggered fashion, but always within one step of each other. Each unit has four control banks and four shutdown banks.

The shutdown banks are maintained either in the fully inserted or fully withdrawn position. The control banks are moved in an overlap pattern, using the following withdrawal sequence: When Control Bank A reaches a predetermined height in the core, Control Bank B begins to move out with Control Bank A. Control Bank A stops at the position of maximum

## BASES

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### BACKGROUND (continued)

withdrawal, and Control Bank B continues to move out. When Control Bank B reaches a predetermined height, Control Bank C begins to move out with Control Bank B. This sequence continues until Control Banks A, B, and C are at the fully withdrawn position, and Control Bank D is approximately halfway withdrawn. The insertion sequence is the opposite of the withdrawal sequence. The control rods are arranged in a radially symmetric pattern, so that control bank motion does not introduce radial asymmetries in the core power distributions.

The axial position of shutdown rods and control rods is indicated by two separate and independent systems that are the Demand Position Indication System (commonly called group step counters) and the Rod Position Indication System.

The Demand Position Indication System counts the pulses from the rod control system that moves the rods. There is one step counter for each group of rods. Individual rods in a group receive the same signal to move and should, therefore, all be at the same position indicated by the group step counter for that group. The Demand Position Indication System is considered highly precise ( $\pm 1$  step or  $\pm 5/8$  inch). If a rod does not move one step for each demand pulse, the step counter will still count the pulse and incorrectly reflect the position of the rod.

The Rod Position Indication System provides an indication of actual rod position, but at a lower precision than the step counters. This system is based on inductive analog signals from a series of coils spaced along a hollow tube. The Rod Position Indication System is capable of monitoring rod position within at least  $\pm 12$  steps.

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### APPLICABLE SAFETY ANALYSES

Control rod misalignment accidents are analyzed in the safety analysis (Ref. 3). The acceptance criteria for addressing control rod inoperability or misalignment are that:

- a. There be no violations of:
  1. Specified acceptable fuel design limits; or
  2. Reactor Coolant System (RCS) pressure boundary integrity; and
- b. The core remains subcritical after accident transients.

## BASES

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### APPLICABLE SAFETY ANALYSES (continued)

There are three RCCA misalignment accidents which are analyzed. They include one or more dropped RCCAs, a dropped RCCA bank, and a statically misaligned RCCA (Ref. 4). A different type of misalignment occurs if one rod fails to insert upon a reactor trip and remains stuck fully withdrawn. This condition requires an evaluation to determine that sufficient reactivity worth is held in the control rods to meet the SDM requirement, with the maximum worth rod stuck fully withdrawn.

For the dropped RCCA(s) misalignment accident, a negative reactivity insertion will result. For those dropped RCCA(s) that do not result in a reactor trip, power may be reestablished either by reactivity feedback or control bank withdrawal. Following a dropped rod event in manual rod control, the plant will establish a new equilibrium condition. The equilibrium process without control system interaction is monotonic, thus removing power overshoot as a concern and establishing the automatic rod control mode of operation as the limiting case.

For the dropped RCCA bank misalignment accident, a reactivity insertion of greater than 500 pcm which will be detected by the power range negative neutron flux rate trip circuitry. The reactor is then tripped. The core is not adversely affected during this period since power is decreasing rapidly. Following the reactor trip, normal shutdown procedures are followed to further cool down the plant.

Two types of analysis are performed in regard to static rod misalignment (Ref. 3). With control banks at their insertion limits, one type of analysis considers the case when any one rod is completely inserted into the core. The second type of analysis considers the case of a completely withdrawn single rod and Control Bank D is fully inserted to its insertion limit. Satisfying limits on departure from nucleate boiling ratio in both of these cases bounds the situation when a rod is misaligned from its group by  $\pm 12$  steps.

Another type of misalignment occurs if one RCCA fails to insert upon a reactor trip and remains stuck fully withdrawn. This condition is assumed in the evaluation to determine that the required SDM is met with the maximum worth RCCA also fully withdrawn (Ref. 4).

Shutdown and control rod OPERABILITY and alignment are directly related to power distributions and SDM, which are initial conditions assumed in safety analyses. Therefore they satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

BASES

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LCO

The limits on shutdown or control rod alignments ensure that the assumptions in the safety analysis will remain valid. The requirements on control rod OPERABILITY ensure that upon reactor trip, the assumed reactivity will be available and will be inserted. The control rod OPERABILITY requirements (i.e., trippability) are separate from the alignment requirements, which ensure that the RCCAs and banks maintain the correct power distribution and rod alignment. The rod OPERABILITY requirement is satisfied provided the rod will fully insert in the required rod drop time assumed in the safety analysis. Rod control malfunctions that result in the inability to move a rod (e.g., rod lift coil failures), but that do not impact trippability, do not result in rod inoperability.

The requirement to maintain the rod alignment to within plus or minus 12 steps is conservative. The minimum misalignment assumed in safety analysis is 10% of span (14.4 inches), and in some cases a total misalignment from fully withdrawn to fully inserted is assumed.

Failure to meet the requirements of this LCO may produce unacceptable power peaking factors and linear heat rates (LHRs), or unacceptable SDMs, that may constitute initial conditions inconsistent with the safety analysis.

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APPLICABILITY

The requirements on RCCA OPERABILITY and alignment are applicable in MODES 1 and 2 because these are the only MODES in which neutron (or fission) power is generated, and the OPERABILITY (i.e., trippability) and alignment of rods have the potential to affect the safety of the plant. In MODES 3, 4, 5, and 6, the alignment limits do not apply because, except for control rod OPERABILITY testing, the control rods are bottomed and the reactor is shut down and not producing fission power. In the shutdown MODES, the OPERABILITY of the shutdown and control rods has the potential to affect the required SDM, but this effect can be compensated for by an increase in the boron concentration of the RCS. See LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," for SDM in MODES 3, 4, and 5 and LCO 3.9.1, "Boron Concentration," for boron concentration requirements during refueling.

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ACTIONS

A.1.1 and A.1.2

When one or more rods are inoperable (i.e., untrippable), there is a possibility that the required SDM may be adversely affected. Under these conditions, it is important to determine the SDM, and if it is less than the required value, initiate boration until the required SDM is recovered. The Completion Time of 1 hour is adequate for determining SDM and, if necessary, for initiating emergency boration and restoring SDM.

In this situation, SDM verification must include the worth of the untrippable rod, as well as a rod of maximum worth.

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## BASES

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### ACTIONS (continued)

#### A.2

If the inoperable rod(s) cannot be restored to OPERABLE status, the plant must be brought to a MODE or condition in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours.

The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

#### B.1

When a rod becomes misaligned, it can usually be moved and is trippable. If the rod can be realigned within the Completion Time of 1 hour, local xenon redistribution during this short interval will not be significant, and operation may proceed without further restriction.

An alternative to realigning a single misaligned RCCA to the group average position is to align the remainder of the group to the position of the misaligned RCCA. However, this must be done without violating the bank sequence, overlap, and insertion limits specified in LCO 3.1.5, "Shutdown Bank Insertion Limits," and LCO 3.1.6, "Control Bank Insertion Limits." The Completion Time of 1 hour gives the operator sufficient time to adjust the rod positions in an orderly manner.

#### B.2.1.1 and B.2.1.2

With a misaligned rod, SDM must be verified to be within limit or boration must be initiated to restore SDM to within limit.

In many cases, realigning the remainder of the group to the misaligned rod may not be desirable.

Power operation may continue with one RCCA misaligned but trippable (OPERABLE), provided that SDM is verified within 1 hour. The Completion Time of 1 hour represents the time necessary for determining the actual unit SDM and, if necessary, aligning and starting the necessary systems and components to initiate boration.

## BASES

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### ACTIONS (continued)

#### B.2.2, B.2.3, B.2.4, B.2.5, and B.2.6

For continued operation with a misaligned rod, RTP must be reduced, SDM must periodically be verified within limits, hot channel factors ( $F_Q(X,Y,Z)$  and  $F_{\Delta H}(X,Y)$ ) must be verified within limits, and the safety analyses must be re-evaluated to confirm continued operation is permissible.

Reduction of power to 75% RTP ensures that local LHR increases resulting from a misaligned RCCA will not cause the core design criteria to be exceeded (Ref. 5). The Completion Time of 2 hours gives the operator sufficient time to accomplish an orderly power reduction without challenging the Reactor Protection System.

When a rod is known to be misaligned, there is a potential to impact the SDM. Since the core conditions can change with time, periodic verification of SDM is required. A Frequency of 12 hours is sufficient to ensure this requirement continues to be met.

Verifying that  $F_Q(X,Y,Z)$  and  $F_{\Delta H}(X,Y)$  are within the required limits ensures that current operation at 75% RTP with a rod misaligned is not resulting in power distributions that may invalidate safety analysis assumptions at full power. The Completion Time of 72 hours allows sufficient time to obtain flux maps of the core power distribution using the incore flux mapping system and to calculate  $F_Q(X,Y,Z)$  and  $F_{\Delta H}(X,Y)$ .

Once current conditions have been verified acceptable, time is available to perform evaluations of accident analysis to determine that core limits will not be exceeded during a Design Basis Event for the duration of operation under these conditions. The accident analyses presented in UFSAR Chapter 15 (Ref. 5) that may be adversely affected will be evaluated to ensure that the analysis results remain valid for the duration of continued operation under these conditions. A Completion Time of 5 days is sufficient time to obtain the required input data and to perform the analysis.

#### C.1

When Required Actions cannot be completed within their Completion Time, the unit must be brought to a MODE in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours, which

## BASES

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### ACTIONS (continued)

obviates concerns about the development of undesirable xenon or power distributions. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging the plant systems.

#### D.1.1 and D.1.2

More than one control rod becoming misaligned from its group average position is not expected, and has the potential to reduce SDM. Therefore, SDM must be evaluated. One hour allows the operator adequate time to determine SDM. Restoration of the required SDM, if necessary, requires increasing the RCS boron concentration to provide negative reactivity, as described in the Bases of LCO 3.1.1. The required Completion Time of 1 hour for initiating boration is reasonable, based on the time required for potential xenon redistribution, the low probability of an accident occurring, and the steps required to complete the action. This allows the operator sufficient time to align the required valves and start the boric acid pumps. Boration will continue until the required SDM is restored.

#### D.2

If more than one rod is found to be misaligned or becomes misaligned because of bank movement, the unit conditions fall outside of the accident analysis assumptions. Since automatic sequencing of the control banks would continue to cause misalignment, the unit must be brought to a MODE or Condition in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours.

The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

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### SURVEILLANCE REQUIREMENTS

#### SR 3.1.4.1

Verification that individual rod positions are within alignment limits allows the operator to detect a rod that is beginning to deviate from its expected position.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.4.2

Verifying each control rod is OPERABLE would require that each rod be tripped. However, in MODES 1 and 2, tripping each control rod would result in radial or axial power tilts, or oscillations. Exercising each individual control rod provides increased confidence that all rods continue to be OPERABLE without exceeding the alignment limit, even if they are not regularly tripped. Moving each control rod by greater than or equal to 10 steps in either direction will not cause radial or axial power tilts, or oscillations, to occur.

To ensure minimum  $F_{\Delta H}$  peaking factor margins are maintained in accordance with TS 3.2.2 during SR 3.1.4.2 rod testing, margin penalties are typically assigned during the test, as described in the Nuclear Design Report. The minimum predicted  $F_{\Delta H}$  future margin, including penalties, should be verified prior to performing the test to ensure adequate margin will be maintained. In the event that a potential negative margin condition exists, compliance with the associated Conditions and Required Actions of TS 3.2.2 should be verified prior to performing rod testing.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

Between required performances of SR 3.1.4.2 (determination of control rod OPERABILITY by movement), if a control rod(s) is discovered to be immovable, but remains trippable, the control rod(s) is considered to be OPERABLE. At any time, if a control rod(s) is immovable, a determination of the trippability (OPERABILITY) of the control rod(s) must be made, and appropriate action taken.

SR 3.1.4.3

Verification of rod drop times allows the operator to determine that the maximum rod drop time permitted is consistent with the assumed rod drop time used in the safety analysis. Measuring rod drop times prior to reactor criticality, after reactor vessel head installation, ensures that the reactor internals and rod drive mechanism will not interfere with rod motion or rod drop time, and that no degradation in these systems has occurred that would adversely affect control rod motion or drop time. This testing is performed with all RCPs operating and the average moderator temperature  $\geq 500^{\circ}\text{F}$  to simulate a reactor trip under actual conditions. Fully withdrawn shall be the condition where shutdown and control banks are at a position within the interval of  $\geq 222$  and  $\leq 231$  steps withdrawn, inclusive.

BASES

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SURVEILLANCE REQUIREMENTS (continued)

This Surveillance is performed during a plant outage, due to the plant conditions needed to perform the SR and the potential for an unplanned plant transient if the Surveillance were performed with the reactor at power.

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REFERENCES

1. 10 CFR 50, Appendix A, GDC 10 and GDC 26.
  2. 10 CFR 50.46.
  3. UFSAR, Section 15.2.3.
  4. UFSAR, Section 15.4.2.
  5. UFSAR, Chapter 15.
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## B 3.1 REACTIVITY CONTROL SYSTEMS

### B 3.1.5 Shutdown Bank Insertion Limits

#### BASES

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**BACKGROUND** The insertion limits of the shutdown and control rods are initial assumptions in all safety analyses that assume rod insertion upon reactor trip. The insertion limits directly affect core power and fuel burnup distributions and assumptions of available ejected rod worth, SDM and initial reactivity insertion rate.

The applicable criteria for these reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10, "Reactor Design," GDC 26, "Reactivity Control System Redundancy and Protection," GDC 28, "Reactivity Limits" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 2). Limits on control rod insertion have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

The rod cluster control assemblies (RCCAs) are divided among control banks and shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control. A group consists of two or more RCCAs that are electrically paralleled to step simultaneously. A bank of RCCAs consists of two groups that are moved in a staggered fashion, but always within one step of each other. Each unit has four control banks and four shutdown banks. See LCO 3.1.4, "Rod Group Alignment Limits," for control and shutdown rod OPERABILITY and alignment requirements, and LCO 3.1.7, "Rod Position Indication," for position indication requirements.

The control banks are used for precise reactivity control of the reactor. The positions of the control banks are normally automatically controlled by the Rod Control System, but they can also be manually controlled. They are capable of adding negative reactivity very quickly (compared to borating). The control banks must be maintained above designed insertion limits and are typically near the fully withdrawn position during normal full power operations.

Hence, they are not capable of adding a large amount of positive reactivity. Boration or dilution of the Reactor Coolant System (RCS) compensates for the reactivity changes associated with large changes in RCS temperature. The design calculations are performed with the assumption that the shutdown banks are withdrawn first. The shutdown banks can be fully withdrawn without the core going critical. This provides available negative reactivity in the event of boration errors. The

BASES

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BACKGROUND (continued)

shutdown banks are controlled manually by the control room operator. During normal unit operation, the shutdown banks are either fully withdrawn or fully inserted. They are moved quarterly or following maintenance to ensure trippability but are returned to the withdrawn position when the testing is completed. The shutdown banks must be withdrawn from the core, prior to withdrawing any control banks during an approach to criticality. They affect core power and burnup distribution, and add negative reactivity to shut down the reactor upon receipt of a reactor trip signal.

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APPLICABLE  
SAFETY  
ANALYSES

On a reactor trip, all RCCAs (shutdown banks and control banks), except the most reactive RCCA, are assumed to insert into the core. The shutdown banks shall be at or above their insertion limits and available to insert the maximum amount of negative reactivity on a reactor trip signal. The control banks may be partially inserted in the core, as allowed by LCO 3.1.6, "Control Bank Insertion Limits." The shutdown bank and control bank insertion limits are established to ensure that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM (see LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") following a reactor trip from full power. The combination of control banks and shutdown banks (less the most reactive RCCA, which is assumed to be fully withdrawn) is sufficient to take the reactor from full power conditions at rated temperature to zero power, and to maintain the required SDM at rated no load temperature (Ref. 3). The shutdown bank insertion limit also limits the reactivity worth of an ejected shutdown rod.

The acceptance criteria for addressing shutdown and control rod bank insertion limits and inoperability or misalignment is that:

- a. There be no violations of:
  - 1. Specified acceptable fuel design limits or
  - 2. RCS pressure boundary integrity and
- b. The core remains subcritical after accident transients.

As such, the shutdown bank insertion limits affect safety analyses involving core reactivity and SDM (Ref. 3).

The shutdown bank insertion limits preserve an initial condition assumed in the safety analyses and, as such, satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

BASES

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LCO The shutdown banks must be within their insertion limits any time the reactor is critical or approaching criticality. This ensures that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM following a reactor trip.

The shutdown bank insertion limits are defined in the COLR.

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APPLICABILITY The shutdown banks must be within their insertion limits, with the reactor in MODES 1 and 2. This ensures that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM following a reactor trip. The shutdown banks do not have to be within their insertion limits in MODE 3, unless an approach to criticality is being made. In MODE 3, 4, 5, or 6, the shutdown banks, except for control rod OPERABILITY testing, are fully inserted in the core and contribute to the SDM. Refer to LCO 3.1.1 for SDM requirements in MODE 2  $k_{\text{eff}} < 1.0$ , MODES 3, 4, and 5. LCO 3.9.1, "Boron Concentration," ensures adequate SDM in MODE 6.

The Applicability requirements have been modified by a Note indicating the LCO requirement is suspended during SR 3.1.4.2. This SR verifies the freedom of the rods to move, and requires the shutdown bank to move below the LCO limits, which would normally violate the LCO.

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ACTIONS A.1, A.2, A.3, A.4, A.5, A.6, and A.7

When one shutdown bank is inserted beyond the insertion limit due to performance of trippability testing per SR 3.1.4.2 and is immovable due to a malfunction in the rod control system, 72 hours are provided to restore the shutdown banks to within limits. Additionally, immediate verification is required to prove that the shutdown bank is less than or equal to 18 steps below the insertion limit as measured by the group demand position indicators, the individual control rod alignment limits of LCOs 3.1.4 and 3.1.6 are met, there are no reactor coolant system boron dilution activities, and there are no power level increases taking place. Checks are performed for each reload core to ensure that bank insertions of up to 18 steps will not result in power distributions which violate the DNB criterion for ANS Condition II transients (moderate frequency transients analyzed in Section 15.2 of the UFSAR). Administrative requirements on the initial controlling bank position will ensure that this insertion and an additional controlling bank insertion of five steps or less will not violate the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 during the repair period. If the controlling bank is inserted more than five steps deeper than its initial position, a calculation will be performed to ensure that the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is met. Since no dilution or power level increases are allowed, shutdown margin will be maintained as long as the controlling bank is far enough above its insertion limit to compensate for the inserted worth of the bank that is beyond its insertion limit. Furthermore, a verification of SDM is

## BASES

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### ACTIONS (continued)

required within 12 hours and when the controlling bank is inserted more than 5 steps from the initial position. The requirement to be in compliance with LCOs 3.1.4 and 3.1.6 ensures that the rods are trippable, and power distribution is acceptable during the time allowed to restore the inserted rod. The 12 hour requirement to verify the SDM is within limits ensures the SDM requirements of LCO 3.1.1 are met during the repair period. Furthermore, the requirement to verify the SDM is within limits when a controlling bank is inserted five steps or more also ensures that SDM requirements of LCO 3.1.1 are met during the repair period. If any of these Required Actions are not met, Condition C must be applied.

The Completion Time of 72 hours is based on operating experience and provides an acceptable time for evaluating and repairing problems with the rod control system, while restricting the probability of a more severe (i.e., ANS Condition III or IV) accident or transient condition occurring concurrently with the insertion limit violation.

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

When one or more shutdown banks is not within insertion limits for reasons other than Condition A, 2 hours is allowed to restore the shutdown banks to within the insertion limits. This is necessary because the available SDM may be significantly reduced, with one or more of the shutdown banks not within their insertion limits. Also, verification of SDM or initiation of boration within 1 hour is required, since the SDM in MODES 1 and 2 is ensured by adhering to the control and shutdown bank insertion limits (see LCO 3.1.1). If shutdown banks are not within their insertion limits, then SDM will be verified by performing a reactivity balance calculation, considering the effects listed in the Bases for SR 3.1.1.1.

The allowed Completion Time of 2 hours provides an acceptable time for evaluating and repairing minor problems without allowing the plant to remain in an unacceptable condition for an extended period of time.

#### C.1

If the Required Action(s) of Condition A or B are not met within the associated Completion Times, the unit must be brought to a MODE where the LCO is not applicable. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.1.5.1

Verification that the shutdown banks are within their insertion limits prior to an approach to criticality ensures that when the reactor is critical, or being taken critical, the shutdown banks will be available to shut down the reactor, and the required SDM will be maintained following a reactor trip. This SR and Frequency ensure that the shutdown banks are withdrawn before the control banks are withdrawn during a unit startup.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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REFERENCES

1. 10 CFR 50, Appendix A, GDC 10, GDC 26, and GDC 28.
  2. 10 CFR 50.46.
  3. UFSAR, Chapter 15.
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## B 3.1 REACTIVITY CONTROL SYSTEMS

### B 3.1.6 Control Bank Insertion Limits

#### BASES

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**BACKGROUND** The insertion limits of the shutdown and control rods are initial assumptions in the safety analyses that assume rod insertion upon reactor trip. The insertion limits directly affect core power and fuel burnup distributions and assumptions of available SDM, and initial reactivity insertion rate.

The applicable criteria for these reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10, "Reactor Design," GDC 26, "Reactivity Control System Redundancy and Protection," GDC 28, "Reactivity Limits" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 2). Limits on control rod insertion have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

The rod cluster control assemblies (RCCAs) are divided among control banks and shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control. A group consists of two or more RCCAs that are electrically paralleled to step simultaneously. A bank of RCCAs consists of two groups that are moved in a staggered fashion, but always within one step of each other. Each unit has four control banks and four shutdown banks. See LCO 3.1.4, "Rod Group Alignment Limits," for control and shutdown rod OPERABILITY and alignment requirements, and LCO 3.1.7, "Rod Position Indication," for position indication requirements.

The control bank insertion limits are specified in the COLR. The control banks are required to be at or above the insertion limit lines.

Overlap is the distance travelled together by two control banks. The predetermined position of control bank C, at which control bank D will begin to move with bank C on a withdrawal is shown on the COLR Figure. The fully withdrawn position is defined in the COLR.

BASES

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BACKGROUND (continued)

The control banks are used for precise reactivity control of the reactor. The positions of the control banks are normally controlled automatically by the Rod Control System, but can also be manually controlled. They are capable of adding reactivity very quickly (compared to borating or diluting).

The power density at any point in the core must be limited, so that the fuel design criteria are maintained. Together, LCO 3.1.4, LCO 3.1.5, "Shutdown Bank Insertion Limits," LCO 3.1.6, LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," provide limits on control component operation and on monitored process variables, which ensure that the core operates within the fuel design criteria.

The shutdown and control bank insertion and alignment limits, AFD, and QPTR are process variables that together characterize and control the three dimensional power distribution of the reactor core. Additionally, the control bank insertion limits control the reactivity that could be added in the event of a rod ejection accident, and the shutdown and control bank insertion limits ensure the required SDM is maintained.

Operation within the subject LCO limits will prevent fuel cladding failures that would breach the primary fission product barrier and release fission products to the reactor coolant in the event of a loss of coolant accident (LOCA), loss of flow, ejected rod, or other accident requiring termination by a Reactor Trip System (RTS) trip function.

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APPLICABLE  
SAFETY  
ANALYSES

The shutdown and control bank insertion limits, AFD, and QPTR LCOs are required to prevent power distributions that could result in fuel cladding failures in the event of a LOCA, loss of flow, ejected rod, or other accident requiring termination by an RTS trip function.

The acceptance criteria for addressing shutdown and control bank insertion limits and inoperability or misalignment are that:

- a. There be no violations of:
  - 1. Specified acceptable fuel design limits or
  - 2. Reactor Coolant System pressure boundary integrity and
- b. The core remains subcritical after accident transients.

BASES

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APPLICABLE SAFETY ANALYSES (continued)

As such, the shutdown and control bank insertion limits affect safety analyses involving core reactivity and power distributions (Ref. 3).

The SDM requirement is ensured by limiting the control and shutdown bank insertion limits so that allowable inserted worth of the RCCAs is such that sufficient reactivity is available in the rods to shut down the reactor to hot zero power with a reactivity margin that assumes the maximum worth RCCA remains fully withdrawn upon trip (Ref. 3).

Operation at the insertion limits or AFD limits may approach the maximum allowable linear heat generation rate or peaking factor with the allowed QPTR present. Operation at the insertion limit may indicate the maximum ejected RCCA worth could be equal to the limiting value in the fuel cycle that has sufficiently high ejected RCCA worths.

The control and shutdown bank insertion limits ensure that safety analyses assumptions for SDM, ejected rod worth, and power distribution peaking factors are preserved (Ref. 3).

The insertion limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii), in that they are initial conditions assumed in the safety analyses.

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LCO

The limits on control banks sequence, overlap, and physical insertion, as defined in the COLR, must be maintained because they serve the function of preserving power distribution, ensuring that the SDM is maintained, ensuring that ejected rod worth is maintained, and ensuring adequate negative reactivity insertion is available on trip. The overlap between control banks provides more uniform rates of reactivity insertion and withdrawal and is imposed to maintain acceptable power peaking during control bank motion.

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APPLICABILITY

The control bank sequence, overlap, and physical insertion limits shall be maintained with the reactor in MODES 1 and 2 with  $k_{\text{eff}} \geq 1.0$ . These limits must be maintained, since they preserve the assumed power distribution, ejected rod worth, SDM, and reactivity rate insertion assumptions. Applicability in MODE 2 with  $k_{\text{eff}} < 1.0$ , MODES 3, 4, and 5 is not required, since neither the power distribution nor ejected rod worth assumptions would be exceeded in these MODES.

The applicability requirements have been modified by a Note indicating the LCO requirements are suspended during the performance of SR 3.1.4.2. This SR verifies the freedom of the rods to move, and requires the control bank to move below the LCO limits, which would violate the LCO.

BASES

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ACTIONS

A.1, A.2, A.3, A.4, A.5, A.6, and A.7

When one control bank is inserted beyond the insertion limit due to performance of trippability testing per SR 3.1.4.2 and is immovable due to malfunctions in the rod control system, 72 hours are provided to restore the control banks to within limits. Additionally, immediate verification is required to prove that the control bank is less than or equal to 18 steps below the insertion limit as measured by the group demand position indicators, the individual rod alignment limits of LCOs 3.1.4 and 3.1.5 are met, there are no reactor coolant system boron concentration dilution activities, and there are no power level increases taking place. Checks are performed for each reload core to ensure that bank insertions of up to 18 steps will not result in power distributions which violate the DNB criterion for ANS Condition II transients (moderate frequency transients analyzed in Section 15.2 of the UFSAR). Administrative requirements on the initial controlling bank position will ensure that this insertion and an additional controlling bank insertion of five steps or less will not violate the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 during the repair period. If the controlling bank is inserted more than five steps deeper than its initial position, a calculation will be performed to ensure that the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is met. Since no dilution or power level increases are allowed, shutdown margin will be maintained as long as the controlling bank is far enough above its insertion limit to compensate for the inserted worth of the bank that is beyond its insertion limit. Furthermore, a verification of SDM is required within 12 hours and when the controlling bank is inserted more than 5 steps from the initial position. The requirement to be in compliance with LCOs 3.1.4 and 3.1.5 ensures that the rods are trippable, and power distribution is acceptable during the time allowed to restore the inserted bank. The 12 hour requirement to verify the SDM is within limits ensures the SDM requirements of LCO 3.1.1 are met during the repair period. Furthermore, the requirement to verify the SDM is within limits when a controlling bank is inserted five steps or more also ensures that SDM requirements of LCO 3.1.1 are met during the repair period. If any of these Required Actions are not met, Condition D must be applied.

The Condition is modified by a Note that specifies it only applies to control banks inserted beyond the insertion limit that are not controlling banks. A controlling bank is defined as a control bank that is less than fully withdrawn as defined in the COLR, with the exception of fully withdrawn banks that have been inserted for the performance of SR 3.1.4.2 (rod freedom of movement Surveillance).

The Completion Time of 72 hours is based on operating experience and provides an acceptable time for evaluating and repairing problems with the rod control system, while restricting the probability of a more severe (i.e., ANS Condition III or IV) accident or transient condition occurring concurrently with the insertion limit violation.

## BASES

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### ACTIONS (continued)

#### B.1.1, B.1.2, B.2, C.1.1, C.1.2, and C.2

When the control banks are outside the acceptable insertion limits, they must be restored to within those limits. This restoration can occur in two ways:

- a. Reducing power to be consistent with rod position or
- b. Moving rods to be consistent with power.

Also, verification of SDM or initiation of boration to regain SDM is required within 1 hour, since the SDM in MODES 1 and 2 normally ensured by adhering to the control and shutdown bank insertion limits (see LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") has been upset. If control banks are not within their insertion limits, then SDM will be verified by performing a reactivity balance calculation, considering the effects listed in the Bases for SR 3.1.1.1.

Similarly, if the control banks are found to be out of sequence or in the wrong overlap configuration, they must be restored to meet the limits.

Operation beyond the LCO limits is allowed for a short time period in order to take conservative action because the simultaneous occurrence of either a LOCA, loss of flow accident, ejected rod accident, or other accident during this short time period, together with an inadequate power distribution or reactivity capability, has an acceptably low probability.

The allowed Completion Time of 2 hours for restoring the banks to within the insertion, sequence, and overlap limits provides an acceptable time for evaluating and repairing minor problems without allowing the plant to remain in an unacceptable condition for an extended period of time.

#### D.1

If Required Action(s) of Condition A, B, or C are not met within the associated Completion Times, the plant must be brought to at least MODE 2 with  $k_{\text{eff}} < 1.0$ , where the LCO is not applicable. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.1.6.1

This Surveillance is required to ensure that the reactor does not achieve criticality with the control banks below their insertion limits.

The estimated critical position (ECP) depends upon a number of factors, one of which is xenon concentration. If the ECP was calculated long before criticality, xenon concentration could change to make the ECP substantially in error. Conversely, determining the ECP immediately before criticality could be an unnecessary burden. There are a number of unit parameters requiring operator attention at that point. Performing the ECP calculation within 4 hours prior to criticality avoids a large error from changes in xenon concentration, but allows the operator some flexibility to schedule the ECP calculation with other startup activities.

SR 3.1.6.2

Verification of the control bank insertion limits is sufficient to detect control banks that may be approaching the insertion limits.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.1.6.3

When control banks are maintained within their insertion limits as checked by SR 3.1.6.2 above, it is unlikely that their sequence and overlap will not be in accordance with requirements provided in the COLR.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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REFERENCES

1. 10 CFR 50, Appendix A, GDC 10, GDC 26, GDC 28.
  2. 10 CFR 50.46.
  3. UFSAR, Chapter 15.
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## B 3.1 REACTIVITY CONTROL SYSTEMS

### B 3.1.7 Rod Position Indication

#### BASES

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**BACKGROUND** According to GDC 13 (Ref. 1), instrumentation to monitor variables and systems over their operating ranges during normal operation, anticipated operational occurrences, and accident conditions must be OPERABLE. LCO 3.1.7 is required to ensure OPERABILITY of the control and shutdown rod position indicators to determine rod positions and thereby ensure compliance with the control rod alignment and insertion limits.

The OPERABILITY, including position indication, of the shutdown and control rods is an initial assumption in all safety analyses that assume rod insertion upon reactor trip. Maximum rod misalignment is an initial assumption in the safety analysis that directly affects core power distributions and assumptions of available SDM. Rod position indication is required to assess OPERABILITY and misalignment.

Mechanical or electrical failures may cause a rod to become inoperable or to become misaligned from its group. Control rod inoperability or misalignment may cause increased power peaking, resulting from the asymmetric reactivity distribution and a reduction in the total available rod worth for reactor shutdown. Therefore, control rod alignment and OPERABILITY are related to core operation in design power peaking limits and the core design requirement of a minimum SDM.

Limits on control rod alignment and OPERABILITY have been established, and rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

Rod cluster control assemblies (RCCAs), or rods, are moved out of the core (up or withdrawn) or into the core (down or inserted) by their control rod drive mechanisms. The RCCAs are divided among control banks and shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control.

The axial position of shutdown rods and control rods are determined by two separate and independent systems: the Demand Position Indication System (commonly called group step counters) and the Rod Position Indication System.

BASES

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BACKGROUND (continued)

The Demand Position Indication System counts the pulses from the Rod Control System that move the rods. There is one step counter for each group of rods. Individual rods in a group receive the same signal to move and should, therefore, be at the same position indicated by the group step counter for that group. The Demand Position Indication System is considered highly precise ( $\pm 1$  step or  $\pm 5/8$  inch). If a rod does not move one step for each demand pulse, the step counter will still count the pulse and incorrectly reflect the position of the rod.

The Rod Position Indication System provides an indication of actual control rod position, but at a lower precision than the step counters. This system is based on inductive analog signals from a series of coils spaced along a hollow tube. A deviation of  $\pm 12$  steps between the group step counter and a rod position indication is based on normal Rod Position Indication System indication accuracy of  $\pm 5\%$  span with a maximum uncertainty of 10% span between the group step counter and the rod position indication.

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APPLICABLE  
SAFETY  
ANALYSES

Control and shutdown rod position accuracy is essential during power operation. Power peaking, ejected rod worth, or SDM limits may be violated in the event of a Design Basis Accident (Ref. 2), with control or shutdown rods operating outside their limits undetected. Therefore, the acceptance criteria for rod position indication are that rod positions must be known with sufficient accuracy in order to verify the core is operating within the group sequence, overlap, design peaking limits, ejected rod worth, and with minimum SDM (LCO 3.1.5, "Shutdown Bank Insertion Limits," and LCO 3.1.6, "Control Bank Insertion Limits"). The rod positions must also be known in order to verify the alignment limits are preserved (LCO 3.1.4, "Rod Group Alignment Limits"). Rod positions are continuously monitored to provide operators with information that ensures the plant is operating within the bounds of the accident analysis assumptions.

The rod position indicator channels satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii). The rod position indicators monitor rod position, which is an initial condition of the accident.

## BASES

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### LCO

LCO 3.1.7 specifies that one Rod Position Indication System shall be OPERABLE for each rod. Additionally, one Demand Position Indication System shall be OPERABLE for each group within a bank. For the rod position indicators to be OPERABLE requires meeting the SR of the LCO and the following:

- a. The Rod Position Indication System indicates within 12 steps of the group step counter demand position as required by LCO 3.1.4, "Rod Group Alignment Limits,"
- b. For the Rod Position Indication System there are no failed coils, and
- c. The Demand Position Indication System has been calibrated either in the fully inserted position or a check is performed between the two step counters in the same bank. Shutdown Banks C and D each contain a single group. Therefore, validation of movement for Shutdown Banks C and D can only be performed with a comparison of the single group to the corresponding RPI movement.

The 12 step agreement limit between the Demand Position Indication System and the Rod Position Indication System indicates that the Demand Position Indication System is adequately calibrated, and can be used for indication of the measurement of control rod bank position.

A deviation of less than the allowable limit, given in LCO 3.1.4, in position indication for a single control rod, ensures high confidence that the position uncertainty of the corresponding control rod group is within the assumed values used in the analysis (that specified control rod group insertion limits).

These requirements ensure that control rod position indication during power operation and PHYSICS TESTS is accurate, and that design assumptions are not challenged.

OPERABILITY of the position indicator channels ensures that inoperable, misaligned, or mispositioned control rods can be detected. Therefore, power peaking, ejected rod worth, and SDM can be controlled within acceptable limits.

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### APPLICABILITY

The requirements of the Rod Position Indication and step counters are only applicable in MODES 1 and 2 (consistent with LCO 3.1.4, LCO 3.1.5, and LCO 3.1.6), because these are the only MODES in which power is generated, and the OPERABILITY and alignment of rods have the potential to affect the safety of the plant. In the shutdown MODES, the OPERABILITY of the shutdown and control banks has the potential to affect the required SDM, but this effect can be compensated for by an increase in the boron concentration of the Reactor Coolant System.

## BASES

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### ACTIONS

The ACTIONS Table is modified by a Note indicating that a separate Condition entry is allowed for each inoperable rod position indicator and each demand position indicator. This is acceptable because the Required Actions for each Condition provide appropriate compensatory actions for each inoperable position indicator.

A second Note has been added to provide clarification that LCO 3.0.4.a and LCO 3.0.4.b are not applicable for Required Action A.2.1 and A.2.2 following startup from a refueling outage, or following entry into MODE 5 of sufficient duration to safely repair an inoperable rod position indication.

#### A.1

When one Rod Position Indication channel per bank fails, the position of the rod may be determined indirectly by use of the movable incore detectors. Based on experience, normal power operation does not require excessive movement of banks. If a bank has been significantly moved, the Required Action of C.1 or C.2 below is required. Therefore, verification of RCCA position within the Completion Time of 12 hours is adequate for allowing continued full power operation, since the probability of simultaneously having a rod significantly out of position and an event sensitive to that rod position is small.

#### A.2.1 and A.2.2

When one RPI channel per bank fails, the position of the rod may still be determined indirectly by use of the movable incore detectors and reviewing the parameters of the rod control system for indications of unintended rod movement for the rod with the inoperable position indication. Therefore, verification of RCCA position within 8 hours and every 31 days thereafter is adequate for allowing continued full power operation as long as a review of the parameters of the rod control system for indications of unintended rod movement for the rod with the inoperable position indication is performed within 16 hours and every 8 hours thereafter. Furthermore, if the rod control system parameters indicate unintended movement or if the rod with an inoperable position indicator is moved greater than 12 steps, then the verification of the RCCA position must be performed within 8 hours. As long as these compensatory actions are met, reactor operation can then continue until the end of the current cycle or until an entry into MODE 5 of sufficient duration that the repair of the inoperable rod position indication can safely be performed.

Required Actions A.2.1 and A.2.2 are modified by a Note directing that these Required Actions may only be applied to one inoperable rod position indicator.

BASES

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ACTIONS (continued)

A.3

Reduction of THERMAL POWER to < 50% RTP puts the core into a condition where rod position is not significantly affecting core peaking factors (Ref. 3).

The allowed Completion Time of 8 hours is reasonable, based on operating experience, for reducing power to < 50% RTP from full power conditions without challenging plant systems and allowing for rod position determination by Required Action A.1 above.

B.1, B.2, B.3, and B.4

When more than one Rod Position Indication per bank fails, additional actions are necessary to ensure that acceptable power distribution limits are maintained, minimum SDM is maintained, and the potential effects of rod misalignment on associated accident analyses are limited. Placing the Rod Control System in manual assures unplanned rod motion will not occur. Together with the indirect position determination available via movable incore detectors will minimize the potential for rod misalignment. The immediate Completion Time for placing the Rod Control System in manual reflects the urgency with which unplanned rod motion must be prevented while in this Condition.

Monitoring and recording reactor coolant  $T_{avg}$  helps assure that significant changes in power distribution and SDM are avoided. The once per hour Completion Time is acceptable because only minor fluctuations in RCS temperature are expected at steady state plant operating conditions.

The position of the rods may be determined indirectly by use of the movable incore detectors. Verification of control rod position once per 12 hours is adequate for allowing continued full power operation for a limited, 24 hour period, since the probability of simultaneously having a rod significantly out of position and an event sensitive to that rod position is small. The 24 hour Completion Time provides sufficient time to troubleshoot and restore the Rod Position Indication System to operation while avoiding the plant challenges associated with the shutdown without full rod position indication.

Based on operating experience, normal power operation does not require excessive rod movement. If one or more rods has been significantly moved, the Required Action of C.1 or C.2 below is required.

## BASES

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### ACTIONS (continued)

#### C.1 and C.2

These Required Actions clarify that when one or more rods with inoperable position indicators have been moved in excess of 24 steps in one direction, since the position was last determined, the Required Actions of A.1 and A.2, or B.1, as applicable are still appropriate but must be initiated promptly under Required Action C.1 to begin verifying that these rods are still properly positioned, relative to their group positions.

If, the rod positions have not been determined, THERMAL POWER must be reduced to  $< 50\%$  RTP within 8 hours to avoid undesirable power distributions that could result from continued operation at  $\geq 50\%$  RTP, if one or more rods are misaligned by more than 24 steps.

#### D.1.1 and D.1.2

With one demand position indicator per bank inoperable, the rod positions can be determined by the Rod Position Indication System. Since normal power operation does not require excessive movement of rods, verification by administrative means that the rod position indicators are OPERABLE and the most withdrawn rod and the least withdrawn rod are  $\leq 12$  steps apart within the allowed Completion Time of once every 12 hours is adequate.

#### D.2

Reduction of THERMAL POWER to  $< 50\%$  RTP puts the core into a condition where rod position is not significantly affecting core peaking factor limits (Ref. 3). The allowed Completion Time of 8 hours provides an acceptable period of time to verify the rod positions per Required Actions C.1 and C.2 or reduce power to  $< 50\%$  RTP.

#### E.1

If the Required Actions cannot be completed within the associated Completion Time, 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. The allowed Completion Time is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.1.7.1

Verification that the Rod Position Indication agrees with the demand position within 12 steps ensures that the Rod Position Indication is operating correctly. This verification will be performed at 20 steps and 215 steps of rod travel.

This Surveillance is performed prior to reactor criticality after each removal of the reactor head, as there is the potential for unnecessary plant transients if the SR were performed with the reactor at power.

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REFERENCES

1. 10 CFR 50, Appendix A, GDC 13.
  2. UFSAR, Section 7.7.1.
  3. UFSAR, Chapter 15.
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## B 3.1 REACTIVITY CONTROL SYSTEMS

### B 3.1.8 PHYSICS TESTS Exceptions - MODE 2

#### BASES

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##### BACKGROUND

The primary purpose of the MODE 2 PHYSICS TESTS exceptions is to permit relaxations of existing LCOs to allow certain PHYSICS TESTS to be performed.

Section XI of 10 CFR 50, Appendix B (Ref. 1), requires that a test program be established to ensure that structures, systems, and components will perform satisfactorily in service. The 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 plant. Requirements for notification of the NRC, for the purpose of conducting tests and experiments, are specified in 10 CFR 50.59 (Ref. 2).

The key objectives of a test program are to (Ref. 3):

- a. Ensure that the facility has been adequately designed,
- b. Validate the analytical models used in the design and analysis,
- c. Verify the assumptions used to predict unit response,
- d. Ensure that installation of equipment in the facility has been accomplished in accordance with the design, and
- e. Verify that the operating and emergency procedures are adequate.

To accomplish these objectives, testing is performed prior to initial criticality, during startup, during low power operations, during power ascension, at high power, and after each refueling. The PHYSICS TESTS requirements for reload fuel cycles ensure that the operating characteristics of the core are consistent with the design predictions and that the core can be operated as designed (Ref. 4).

PHYSICS TESTS procedures are written and approved in accordance with established formats. The procedures include the information necessary to permit a detailed execution of the testing required to ensure that the design intent is met. PHYSICS TESTS are performed in accordance with these procedures and test results are approved prior to continued power escalation and long term power operation.

## BASES

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### BACKGROUND (continued)

The PHYSICS TESTS required for reload fuel cycles (Ref. 4) in MODE 2 are listed below:

- a. Critical Boron Concentration - Control Rods Withdrawn;
- b. Critical Boron Concentration - Control Rods Inserted;
- c. Control Rod Worth; and
- d. Isothermal Temperature Coefficient (ITC).

These and other supplementary tests may be required to calibrate the nuclear instrumentation or to diagnose operational problems. These tests may cause the operating controls and process variables to deviate from their LCO requirements during their performance.

- a. The Critical Boron Concentration - Control Rods Withdrawn Test measures the critical boron concentration at hot zero power (HZP). With all rods out, the lead control bank is at or near its fully withdrawn position. HZP is where the core is critical ( $k_{\text{eff}} = 1.0$ ), and the Reactor Coolant System (RCS) is at design temperature and pressure for zero power. Performance of this test should not violate any of the referenced LCOs.
- b. The Critical Boron Concentration - Control Rods Inserted Test measures the critical boron concentration at HZP, with a bank having a worth of at least 1%  $\Delta k/k$  when fully inserted into the core. This test is used to measure the boron reactivity coefficient. With the core at HZP and all banks fully withdrawn, the boron concentration of the reactor coolant is gradually lowered in a continuous manner. The selected bank is then inserted to make up for the decreasing boron concentration until the selected bank has been moved over its entire range of travel. The reactivity resulting from each incremental bank movement is measured with a reactivity computer. The difference between the measured critical boron concentration with all rods fully withdrawn and with the bank inserted is determined. The boron reactivity coefficient is determined by dividing the measured bank worth by the measured boron concentration difference. Performance of this test could violate LCO 3.1.4, "Rod Group Alignment Limits," LCO 3.1.5, "Shutdown Bank Insertion Limit," or LCO 3.1.6, "Control Bank Insertion Limits."

BASES

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BACKGROUND (continued)

- c. The Control Rod Worth Test is used to measure the reactivity worth of selected control banks. This test is performed at HZP and has three alternative methods of performance. The first method, the Boron Exchange Method, varies the reactor coolant boron concentration and moves the selected control bank in response to the changing boron concentration. The reactivity changes are measured with a reactivity computer. This sequence is repeated for the remaining control banks. The second method, the Rod Swap Method, measures the worth of a predetermined reference bank using the Boron Exchange Method above. The reference bank is then nearly fully inserted into the core. The selected bank is then inserted into the core as the reference bank is withdrawn. The HZP critical conditions are then determined with the selected bank fully inserted into the core. The worth of the selected bank is inferred, based on the position of the reference bank with respect to the selected bank. This sequence is repeated as necessary for the remaining control banks. The third method, the Boron Endpoint Method, moves the selected control bank over its entire length of travel and then varies the reactor coolant boron concentration to achieve HZP criticality again. The difference in boron concentration is the worth of the selected control bank. This sequence is repeated for the remaining control banks. Performance of this test could violate LCO 3.1.4, LCO 3.1.5, or LCO 3.1.6.
- d. The ITC Test measures the ITC of the reactor. This test is performed at HZP and has two methods of performance. The first method, the Slope Method, varies RCS temperature in a slow and continuous manner. The reactivity change is measured with a reactivity computer as a function of the temperature change. The ITC is the slope of the reactivity versus the temperature plot. The test is repeated by reversing the direction of the temperature change, and the final ITC is the average of the two calculated ITCs. The second method, the Endpoint Method, changes the RCS temperature and measures the reactivity at the beginning and end of the temperature change. The ITC is the total reactivity change divided by the total temperature change. The test is repeated by reversing the direction of the temperature change, and the final ITC is the average of the two calculated ITCs. Performance of this test could violate LCO 3.4.2, "RCS Minimum Temperature for Criticality."

BASES

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APPLICABLE  
SAFETY  
ANALYSES

The fuel is protected by LCOs that preserve the initial conditions of the core assumed during the safety analyses. The methods for development of the LCOs that are excepted by this LCO are described in the Core Operating Limit Methodology for Westinghouse Designed PWRs (Ref. 5). The above mentioned PHYSICS TESTS, and other tests that may be required to calibrate nuclear instrumentation or to diagnose operational problems, may require the operating control or process variables to deviate from their LCO limitations.

The UFSAR defines requirements for initial testing of the facility, including PHYSICS TESTS. Table 14.1-3 summarizes the zero, low power, and power tests. Requirements for reload fuel cycle PHYSICS TESTS are defined in ANSI/ANS-19.6.1-1997 (Ref. 4). Although these PHYSICS TESTS are generally accomplished within the limits for the LCOs, conditions may occur when one or more LCOs must be suspended to make completion of PHYSICS TESTS possible or practical. This is acceptable as long as the fuel design criteria are not violated. When one or more of the requirements specified in LCO 3.1.3, "Moderator Temperature Coefficient (MTC)," LCO 3.1.4, LCO 3.1.5, LCO 3.1.6, and LCO 3.4.2 are suspended for PHYSICS TESTS, the fuel design criteria are preserved as long as the power level is limited to  $\leq 5\%$  RTP, the reactor coolant temperature is kept  $\geq 531^\circ\text{F}$ , and SDM is within the limits provided in the COLR.

The PHYSICS TESTS include measurement of core nuclear parameters or the exercise of control components that affect process variables. Among the process variables involved are AFD and QPTR, representing initial conditions of the unit safety analyses. Also involved are the movable control components (control and shutdown rods), that are required to shut down the reactor. The limits for these variables are specified for each fuel cycle in the COLR.

As described in LCO 3.0.7, compliance with Test Exception LCOs is optional, and therefore no criteria of 10 CFR 50.36(c)(2)(ii) apply. Test Exception LCOs provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs. A discussion of the criteria satisfied for the other LCOs is provided in their respective Bases.

BASES

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LCO This LCO allows the reactor parameters of MTC and minimum temperature for criticality to be outside their specified limits. In addition, it allows selected control and shutdown rods to be positioned outside of their specified alignment and insertion limits. One power range neutron flux channel may be bypassed, reducing the number of required channels from 4 to 3. Operation beyond specified limits is permitted for the purpose of performing PHYSICS TESTS and poses no threat to fuel integrity, provided the SRs are met.

The requirements of LCO 3.1.3, LCO 3.1.4, LCO 3.1.5, LCO 3.1.6, and LCO 3.4.2 may be suspended and the number of required channels for LCO 3.3.1, "RTS Instrumentation," Functions 2, 3, 6 and 16.e may be reduced to 3 required channels during the performance of PHYSICS TESTS provided:

- a. RCS lowest loop average temperature is  $\geq 531^{\circ}\text{F}$ ,
- b. SDM is within the limits provided in the COLR, and
- c. THERMAL POWER is  $\leq 5\%$  RTP.

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APPLICABILITY This LCO is applicable when performing low power PHYSICS TESTS. The Applicability is stated as "during PHYSICS TESTS initiated in MODE 2" to ensure that the 5% RTP maximum power level is not exceeded. Should the THERMAL POWER exceed 5% RTP, and consequently the unit enter MODE 1, this Applicability statement prevents exiting this Specification and its Required Actions.

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ACTIONS A.1 and A.2

If the SDM requirement is not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. The operator should begin boration with the best source available for the plant conditions. Boration will be continued until SDM is within limit.

Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable LCOs to within specification.

B.1

When THERMAL POWER is  $> 5\%$  RTP, the only acceptable action is to open the reactor trip breakers (RTBs) to prevent operation of the reactor beyond its design limits. Immediately opening the RTBs will shut down the reactor and prevent operation of the reactor outside of its design limits.

BASES

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ACTIONS (continued)

C.1

When the RCS lowest  $T_{avg}$  is  $< 531^{\circ}\text{F}$ , the appropriate action is to restore  $T_{avg}$  to within its specified limit. The allowed Completion Time of 15 minutes provides time for restoring  $T_{avg}$  to within limits without allowing the plant to remain in an unacceptable condition for an extended period of time. Operation with the reactor critical and with temperature below  $531^{\circ}\text{F}$  could violate the assumptions for accidents analyzed in the safety analyses.

D.1

If the Required Actions cannot be completed within the associated Completion Time, 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 an additional 15 minutes. The Completion Time of 15 additional minutes is reasonable, based on operating experience, for reaching MODE 3 in an orderly manner and without challenging plant systems.

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SURVEILLANCE  
REQUIREMENTS

SR 3.1.8.1

The power range and intermediate range neutron detectors must be verified to be OPERABLE in MODE 2 by LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation." A CHANNEL OPERATIONAL TEST is performed on each power range and intermediate range channel prior to initiation of the PHYSICS TESTS. This will ensure that the RTS is properly aligned to provide the required degree of core protection during the performance of the PHYSICS TESTS.

SR 3.1.8.2

Verification that the RCS lowest loop  $T_{avg}$  is  $\geq 531^{\circ}\text{F}$  will ensure that the unit is not operating in a condition that could invalidate the safety analyses. Verification of the RCS temperature during the performance of the PHYSICS TESTS will ensure that the initial conditions of the safety analyses are not violated.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.8.3

Verification that the THERMAL POWER is  $\leq 5\%$  RTP will ensure that the plant is not operating in a condition that could invalidate the safety analyses. Verification of the THERMAL POWER during the performance of the PHYSICS TESTS will ensure that the initial conditions of the safety analyses are not violated.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.1.8.4

The SDM is verified by performing a reactivity balance calculation, considering the following reactivity effects:

- a. RCS boron concentration,
- b. Control bank position,
- c. RCS average temperature,
- d. Fuel burnup based on gross thermal energy generation,
- e. Xenon concentration,
- f. Samarium concentration,
- g. Isothermal temperature coefficient (ITC), when below the point of adding heat (POAH),
- h. Moderator temperature defect, when above the POAH, and
- i. Doppler defect, when above the POAH.

Using the ITC accounts for Doppler reactivity in this calculation when the reactor is subcritical or critical but below the POAH, and the fuel temperature will be changing at the same rate as the RCS.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

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- REFERENCES
1. 10 CFR 50, Appendix B, Section XI.
  2. 10 CFR 50.59.
  3. Regulatory Guide 1.68, Revision 2, August, 1978.
  4. ANSI/ANS-19.6.1-1997.
  5. BAW-10163P-A, "Core Operating Limit Methodology for Westinghouse Designed PWRs," June 1989.
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## B 3.2 POWER DISTRIBUTION LIMITS

### B 3.2.1 Heat Flux Hot Channel Factor (F<sub>Q</sub>(X,Y,Z))

#### BASES

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**BACKGROUND** The purpose of the limits on the values of F<sub>Q</sub>(X,Y,Z) is to limit the local (i.e., pellet) peak power density. The value of F<sub>Q</sub>(X,Y,Z) varies along the axial height (Z) of the core and by assembly location, X, Y.

F<sub>Q</sub>(X,Y,Z) is defined as the maximum local fuel rod linear power density divided by the average fuel rod linear power density. Therefore, F<sub>Q</sub>(X,Y,Z) is a measure of the peak fuel pellet power within the reactor core.

During power operation, the global power distribution is limited by LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO(QPTR)," which are directly and continuously measured process variables. These LCOs, along with LCO 3.1.6, "Control Bank Insertion Limits," maintain the core limits on power distributions on a continuous basis.

F<sub>Q</sub>(X,Y,Z) varies with fuel loading patterns, control bank insertion, fuel burnup, and changes in axial power distribution.

F<sub>Q</sub>(X,Y,Z) is measured periodically using the incore detector system. These measurements are generally taken with the core at or near equilibrium conditions.

Using the measured three dimensional power distributions, it is possible to derive a measured value for F<sub>Q</sub>(X,Y,Z). However, because this value represents an equilibrium condition, it does not include the variations in the value of F<sub>Q</sub>(X,Y,Z) which are present during nonequilibrium situations such as load following or power ascension.

To account for these possible variations, "the F<sub>Q</sub>(X,Y,Z) limits, BQDES(X,Y,Z) and BCDES(X,Y,Z), have been adjusted by pre-calculated factors (MQ(X,Y,Z) and MC(X,Y,Z) respectively) to account for the calculated worst case transient conditions."

Core monitoring and control under non-equilibrium conditions are accomplished by operating the core within the limits of the appropriate LCOs, including the limits on AFD, QPTR, and control rod insertion.

BASES

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APPLICABLE  
SAFETY  
ANALYSES

This LCO precludes core power distributions that violate the following fuel design criteria:

- a. During a large break loss of coolant accident (LOCA), the peak cladding temperature must not exceed 2200°F (Ref. 1),
- b. During a loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a departure from nucleate boiling (DNB) condition,
- c. During an ejected rod accident, the energy deposition to the fuel must not exceed 280 cal/gm (Ref. 2), and
- d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 3).

Limits on F<sub>Q</sub>(X,Y,Z) ensure that the value of the initial total peaking factor assumed in the accident analyses remains valid. Other criteria must also be met (e.g., maximum cladding oxidation, maximum hydrogen generation, coolable geometry, and long term cooling). However, the peak cladding temperature is typically most limiting.

F<sub>Q</sub>(X,Y,Z) limits assumed in the LOCA analysis are typically limiting relative to (i.e., lower than) the F<sub>Q</sub>(X,Y,Z) limit assumed in safety analyses for other postulated accidents. Therefore, this LCO provides conservative limits for other postulated accidents.

F<sub>Q</sub>(X,Y,Z) satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

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LCO

The Heat Flux Hot Channel Factor, F<sub>Q</sub>(X,Y,Z), shall be limited by the following relationships:

$$F_Q(X,Y,Z) \leq (F_Q^{RTP} / P) K(Z) \text{ for } P > 0.5$$

$$F_Q(X,Y,Z) \leq (F_Q^{RTP} / 0.5) K(Z) \text{ for } P \leq 0.5$$

where: F<sub>Q</sub><sup>RTP</sup> is the F<sub>Q</sub>(X,Y,Z) limit at RTP provided in the COLR,

K(Z) is the normalized F<sub>Q</sub>(X,Y,Z) as a function of core height provided in the COLR, and

$$P = \text{THERMAL POWER} / \text{RTP}$$

## BASES

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### LCO (continued)

For SQN, the actual values of  $F_Q^{RTP}$  and  $K(Z)$  are given in the COLR; however,  $F_Q^{RTP}$  is normally a number on the order of 2.62.

Measured  $F_Q(X,Y,Z)$  is compared against three limits:

- Steady state limit,  $(F_Q^{RTP}/P) * K(Z)$ ,
- Limiting condition LOCA limit,  $BQDES(X,Y,Z)$ , and
- Limiting condition centerline fuel melt limit,  $BCDES(X,Y,Z)$ .

$F_Q(X,Y,Z)$  is approximated by  $F_Q^C(X, Y, Z)$  for the steady state limit on  $F_Q(X,Y,Z)$ . An  $F_Q^C(X, Y, Z)$  evaluation requires using the moveable incore detectors to obtain a power distribution map in MODE 1. From the incore flux map results we obtain the measured value ( $F_Q^M(X,Y,Z)$ ) of  $F_Q(X,Y,Z)$ . Then,

$$F_Q^C(X, Y, Z) = \text{overall measured } F_Q(X,Y,Z) * 1.05 * 1.03$$

where, 1.05 is the measurement reliability factor that accounts for flux map measurement uncertainty (Reference 5) and 1.03 is the local engineering heat flux hot channel factor to account for fuel rod manufacturing tolerance (Reference 4).

$BQDES(X,Y,Z)$  and  $BCDES(X,Y,Z)$  are cycle dependent design limits to ensure the  $F_Q(X,Y,Z)$  limit is met during transients. An evaluation of these limits requires obtaining an incore flux map in MODE 1. From the incore flux map results we obtain the assembly nodal measured value ( $F_Q^M(X, Y, Z)$ ) of  $F_Q(X,Y,Z)$ .  $F_Q^M(X, Y, Z)$  is compared directly to the limits  $BQDES(X,Y,Z)$  and  $BCDES(X,Y,Z)$ . This is appropriate since  $BQDES(X,Y,Z)$  and  $BCDES(X,Y,Z)$  have been adjusted for uncertainties.

The expression for  $BQDES(X,Y,Z)$  is:  $BQDES(X,Y,Z) = P^d(X,Y,Z) * MQ(X,Y,Z) * NRF * F1 / MRF$

## BASES

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### LCO (continued)

where:

- BQDES(X,Y,Z) is the cycle dependent maximum allowable design peaking factor for fuel assembly X,Y at axial location Z. BQDES(X,Y,Z) ensures that the LOCA limit will be preserved for operation within the LCO limits, including allowances for calculational and measurement uncertainties;
- P<sup>d</sup>(X,Y,Z) is the design power distribution for fuel assembly X,Y at axial location Z, including the operational flexibility factor;
- MQ(X,Y,Z) is the minimum available margin ratio for the LOCA limit at assembly X,Y and axial location Z;
- NRF is the nuclear reliability factor;
- F1 is the spacer grid factor;
- MRF is measurement reliability factor.

The expression for BCDES(X,Y,Z) is:  $BCDES(X,Y,Z) = P^d(X,Y,Z) * MC(X,Y,Z) * NRF * F1 / MRF$

where:

- BCDES(X,Y,Z) is the cycle dependent maximum allowable design peaking factor for fuel assembly X,Y, at axial location Z. BCDES(X,Y,Z) ensures that the centerline fuel melt limit will be preserved for operation within the LCO limits, including allowances for calculational and measurement uncertainties;
- MC(X,Y,Z) is the minimum available margin ratio for the centerline fuel melt limit at assembly X,Y and axial location Z;

The reactor core is operating as designed if the measured steady state core power distribution agrees with prediction within statistical variation. This guarantees that the operating limits will preserve the thermal criteria in the applicable safety analyses. The core is operating as designed if the following relationship is satisfied:

$$F_Q^M(X, Y, Z) \leq BQNOM(X,Y,Z)$$

where:

- BQNOM(X,Y,Z) is the nominal design peaking factor for assembly X,Y at axial location Z increased by an allowance for the expected deviation between the measured and predicted design power distribution.

BASES

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LCO (continued)

The F<sub>Q</sub>(X,Y,Z) limits define limiting values for core power peaking that precludes peak cladding temperatures above 2200°F during either a large or small break LOCA.

BQNOM (X,Y,Z), BQDES(X,Y,Z), and BCDES(X,Y,Z) Data bases are provided for the plant power distribution analysis computer codes on a cycle specific basis and are determined using the methodology for core limit generation described in the references in the COLR.

This LCO requires operation within the bounds assumed in the safety analyses. Calculations are performed in the core design process to confirm that the core can be controlled in such a manner during operation that it can stay within the LOCA F<sub>Q</sub>(X,Y,Z) limits. If F<sub>Q</sub><sup>C</sup>(X,Y,Z) cannot be maintained within the LCO limits, reduction of the core power is required. Note that sufficient reduction of the AFD limits will also result in a reduction of the core power.

Violating the LCO limits for F<sub>Q</sub>(X,Y,Z) produces unacceptable consequences if a design basis event occurs while F<sub>Q</sub>(X,Y,Z) is outside its specified limits.

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APPLICABILITY

The F<sub>Q</sub>(X,Y,Z) limits must be maintained in MODE 1 to prevent core power distributions from exceeding the limits assumed in the safety analyses. Applicability in other MODES is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the reactor coolant to require a limit on the distribution of core power.

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ACTIONS

A.1

Reducing THERMAL POWER by ≥ 1% RTP for each 1% by which F<sub>Q</sub><sup>C</sup>(X,Y,Z) exceeds its limit, maintains an acceptable absolute power density. F<sub>Q</sub><sup>C</sup>(X,Y,Z) is F<sub>Q</sub><sup>M</sup>(X,Y,Z) multiplied by a factor accounting for manufacturing tolerances and measurement uncertainties. F<sub>Q</sub><sup>M</sup>(X,Y,Z) is the measured value of F<sub>Q</sub>(X,Y,Z). The Completion Time of 15 minutes provides an acceptable time to reduce power in an orderly manner and without allowing the plant to remain in an unacceptable condition for an extended period of time. The maximum allowable power level initially determined by Required Action A.1 may be affected by subsequent

BASES

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## ACTIONS (continued)

determinations of  $F_Q^C(X,Y,Z)$  and would require power reductions within 15 minutes of the  $F_Q^C(X,Y,Z)$  determination, if necessary to comply with the decreased maximum allowable power level. Decreases in  $F_Q^C(X,Y,Z)$  would allow increasing the maximum allowable power level and increasing power up to this revised limit.

A.2

Required Action A.2 requires an administrative reduction of the AFD limits by  $\geq 1\%$  for each 1% by which  $F_Q^C(X,Y,Z)$  exceeds the steady state limit. The allowed Completion Time of 2 hours, restricts the axial flux distribution such that even if a transient occurred, core peaking factor limits are not exceeded. The maximum allowable AFD limits initially determined by Required Action A.2 may be affected by subsequent determinations of  $F_Q^C(X,Y,Z)$  and would require further AFD limit reductions within 2 hours of the  $F_Q^C(X,Y,Z)$  determination, if necessary to comply with the decreased maximum allowable AFD limits. Decreases in  $F_Q^C(X,Y,Z)$  would allow increasing the maximum allowable AFD limits.

A.3

Reduction in the Overpower  $\Delta T$  trip setpoints (value of  $K_4$ ) by  $\geq 1\%$  in  $\Delta T$  span for each 1% by which  $F_Q^C(X,Y,Z)$  exceeds its limit, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 48 hours is sufficient considering the small likelihood of a severe transient in this time period, and the preceding prompt reduction in THERMAL POWER in accordance with Required Action A.1. The maximum allowable Overpower  $\Delta T$  trip setpoints initially determined by Required Action A.3 may be affected by subsequent determinations of  $F_Q^C(X,Y,Z)$  and would require Overpower  $\Delta T$  trip setpoint reductions within 48 hours of the  $F_Q^C(X,Y,Z)$  determination, if necessary to comply with the decreased maximum allowable Overpower  $\Delta T$  trip setpoints. Decreases in  $F_Q^C(X,Y,Z)$  would allow increasing the maximum allowable Overpower  $\Delta T$  trip setpoints.

BASES

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## ACTIONS (continued)

A.4

A reduction of the Power Range Neutron Flux - High trip setpoints by  $\geq 1\%$  for each 1% by which  $F_Q^C(X,Y,Z)$  exceeds its limit, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 72 hours is sufficient considering the small likelihood of a severe transient in this time period and the preceding prompt reduction in THERMAL POWER in accordance with Required Action A.1. The maximum allowable Power Range Neutron Flux - High trip setpoints initially determined by Required Action A.4 may be affected by subsequent determinations of  $F_Q^C(X,Y,Z)$  and would require Power Range Neutron Flux - High trip setpoint reductions within 72 hours of the  $F_Q^C(X,Y,Z)$  determination, if necessary to comply with the decreased maximum allowable Power Range Neutron Flux - High trip setpoints. Decreases in  $F_Q^C(X,Y,Z)$  would allow increasing the maximum allowable Power Range Neutron Flux - High trip setpoints.

A.5

Verification that  $F_Q^C(X,Y,Z)$  has been restored to within its steady state and transient limit, by performing SR 3.2.1.1, SR 3.2.1.2 and SR 3.2.1.3 prior to increasing THERMAL POWER above the limit imposed by Required Action A.1, ensures that core conditions during operation at higher power levels and future operation are consistent with safety analyses assumptions.

Since  $F_Q^C(X,Y,Z)$  exceeds the steady state limit, the limiting condition operational limit (BQDES) and the limiting condition Reactor Protection System limit (BCDES) may also be exceeded. By performing SR 3.2.1.2 and SR 3.2.1.3, appropriate actions with respect to reductions in AFD limits and OPΔT trip setpoints will be performed, ensuring that core conditions during operational and Condition II transients are maintained within the bounds of the safety analysis.

Condition A is modified by a Note that requires Required Action A.5 to be performed whenever the Condition is entered. This ensures that SR 3.2.1.1, SR 3.2.1.2 and SR 3.2.1.3 will be performed prior to increasing THERMAL POWER above the limit of Required Action A.1, even when Condition A is exited prior to performing Required Action A.5. Performance of SR 3.2.1.1, SR 3.2.1.2 and SR 3.2.1.3 are necessary to assure  $F_Q^C(X,Y,Z)$  is properly evaluated prior to increasing THERMAL POWER.

## BASES

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### ACTIONS (continued)

#### B.1 and B.2

The F<sub>Q</sub>(X,Y,Z) margin supporting AFD operational limits (AFD margin) during transient operations is based on the relationship between F<sub>Q</sub><sup>M</sup>(X,Y,Z) and the limiting condition operational limit, BQDES (X,Y,Z), as follows:

$$\%AFD \text{ margin} = \left( 1 - \frac{F_Q^M(X,Y,Z)}{BQDES(X,Y,Z)} \right) * 100\%$$

The AFD min margin = minimum % margin value of all locations examined. If the reactor core is operating as designed, then F<sub>Q</sub><sup>M</sup>(X,Y,Z) is less than BQDES (X,Y,Z) and calculation of %AFD margin is not required. If the AFD margin is less than zero, then F<sub>Q</sub><sup>M</sup>(X,Y,Z) is greater than BQDES (X,Y,Z) and the AFD limits may not be adequate to prevent exceeding the peaking criteria for a LOCA if a normal operational transient occurs.

Required Actions B.1 and B.2 require reducing the AFD limit lines as follows. The AFD limit reduction is from the full power AFD limits. The adjusted AFD limits must be used until a new measurement shows that a smaller adjustment can be made to the AFD limits, or that no adjustment is necessary:

APL = PAFDL – Absolute Value of (PSLOPE<sup>AFD</sup> \* % AFD Margin)  
ANL = NAFDL + Absolute Value of (NSLOPE<sup>AFD</sup> \* % AFD Margin)

where,

- APL is the adjusted positive AFD limit.
- ANL is the adjusted negative AFD limit.
- PAFDL is the positive AFD limit defined in the COLR.
- NAFDL is the negative AFD limit defined in the COLR.
- PSLOPE<sup>AFD</sup> is the adjustment to the positive AFD limit required to compensate for each 1% that F<sub>Q</sub><sup>M</sup>(X,Y,Z) exceeds BQDES (X,Y,Z) as defined in the COLR.
- NSLOPE<sup>AFD</sup> is the adjustment to the negative AFD limit required to compensate for each 1% that F<sub>Q</sub><sup>M</sup>(X,Y,Z) exceeds BQDES (X,Y,Z) as defined in the COLR.
- % AFD Margin is the most negative margin determined above.

BASES

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ACTIONS (continued)

Completing Required Actions B.1 and B.2 within the allowed Completion Time of 2 hours, restricts the axial flux distribution such that even if a transient occurred, core peaking factor limits are not exceeded.

C.1 and C.2

The F<sub>Q</sub>(X,Y,Z) margin supporting the Overpower ΔT f<sub>2</sub>(ΔI) breakpoints (f<sub>2</sub>(ΔI) margin) during transient operations is based on the relationship between F<sub>Q</sub><sup>M</sup>(X,Y,Z) and the limit, BCDES(X,Y,Z), as follows:

$$\% f_2(\Delta I) \text{ margin} = \left( 1 - \frac{F_Q^M(X,Y,Z)}{BCDES(X,Y,Z)} \right) * 100\%$$

The f<sub>2</sub>(ΔI) min margin = minimum % margin value of all locations examined. If the reactor core is operating as designed, then F<sub>Q</sub><sup>M</sup>(X,Y,Z) is less than BCDES(X,Y,Z) and calculation of % f<sub>2</sub>(ΔI) margin is not required. If the f<sub>2</sub>(ΔI) margin is less than zero, then F<sub>Q</sub><sup>M</sup>(X,Y,Z) is greater than BCDES(X,Y,Z) and there is a potential that the f<sub>2</sub>(ΔI) limits are insufficient to preclude centerline fuel melt during certain transients.

Required Actions C.1 and C.2 require reducing the f<sub>2</sub>(ΔI) breakpoint limits as follows. The f<sub>2</sub>(ΔI) breakpoint limit reduction is always from the full power f<sub>2</sub>(ΔI) breakpoint limits. The adjusted f<sub>2</sub>(ΔI) breakpoint limits must be used until a new measurement shows that a smaller adjustment can be made to the f<sub>2</sub>(ΔI) breakpoint limits, or that no adjustment is necessary.

$$\text{Pos}f_2(\Delta I)_{\text{Adjusted}} = \text{Pos}f_2(\Delta I)^{\text{Limit}} - \text{Absolute Value of (PSLOPE}^{f_2(\Delta I)} * \% f_2(\Delta I) \text{ Margin)}$$

$$\text{Neg}f_2(\Delta I)_{\text{Adjusted}} = \text{Neg}f_2(\Delta I)^{\text{Limit}} + \text{Absolute Value of (NSLOPE}^{f_2(\Delta I)} * \% f_2(\Delta I) \text{ Margin)}$$

where:

- Posf<sub>2</sub>(ΔI)<sub>Adjusted</sub> is the adjusted OPΔT positive f<sub>2</sub>(ΔI) breakpoint limit.
- Negf<sub>2</sub>(ΔI)<sub>Adjusted</sub> is the adjusted OPΔT negative f<sub>2</sub>(ΔI) breakpoint limit.
- Posf<sub>2</sub>(ΔI)<sup>Limit</sup> is the OPΔT positive f<sub>2</sub>(ΔI) breakpoint limit defined in the COLR.
- Negf<sub>2</sub>(ΔI)<sup>Limit</sup> is the OPΔT negative f<sub>2</sub>(ΔI) breakpoint limit defined in the COLR.

BASES

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ACTIONS (continued)

- PSLOPE<sup>f<sub>2</sub>(ΔI)</sup> is the adjustment to the positive OPΔT f<sub>2</sub>(ΔI) limit required to compensate for each 1% that F<sub>Q</sub><sup>M</sup>(X,Y,Z) exceeds BCDES(X,Y,Z) as defined in the COLR.
- NSLOPE<sup>f<sub>2</sub>(ΔI)</sup> is the adjustment to the negative OPΔT f<sub>2</sub>(ΔI) limit required to compensate for each 1% that F<sub>Q</sub><sup>M</sup>(X,Y,Z) exceeds BCDES(X,Y,Z) as defined in the COLR.
- % f<sub>2</sub>(ΔI) Margin is the most negative margin determined above.

Completing Required Actions C.1 and C.2 is a conservative action for protection against the consequences of transients since this adjustment limits the peak transient power level which can be achieved during an anticipated operational occurrence. Completing Required Actions C.1 and C.2 within the allowed Completion Time of 48 hours is sufficient considering the small likelihood of a limiting transient in this time period.

D.1

If Required Actions A.1 through A.5, B.1, B.2, C.1 or C.2 are not met within their associated Completion Times, the plant must be placed in a mode or condition in which the LCO requirements are not applicable. This is done by placing the plant in at least MODE 2 within 6 hours.

This allowed Completion Time is reasonable based on operating experience regarding the amount of time it takes to reach MODE 2 from full power operation in an orderly manner and without challenging plant systems.

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SURVEILLANCE  
REQUIREMENTS

SR 3.2.1.1, SR 3.2.1.2 and SR 3.2.1.3 are modified by a Note. It states that Surveillance performance is not required until 12 hours after an equilibrium power level has been achieved at which a power distribution map can be obtained. This allowance is modified, however, by one of the Frequency conditions that requires verification that F<sub>Q</sub><sup>C</sup>(X,Y,Z) and F<sub>Q</sub><sup>M</sup>(X,Y,Z) are within their specified limits after a power rise of more than 10% RTP over the THERMAL POWER at which they were last verified to be within specified limits. Because F<sub>Q</sub><sup>C</sup>(X,Y,Z) and F<sub>Q</sub><sup>M</sup>(X,Y,Z) could not have previously been measured in this reload core, there is a second Frequency condition, applicable only for reload cores, that requires determination of these parameters before exceeding 75% RTP. This ensures that some determination of F<sub>Q</sub><sup>C</sup>(X,Y,Z) and F<sub>Q</sub><sup>M</sup>(X,Y,Z) are made at a lower power level at which adequate margin is available before going to 100% RTP. Also, this Frequency condition, together with the Frequency condition requiring verification of F<sub>Q</sub><sup>C</sup>(X,Y,Z) and F<sub>Q</sub><sup>M</sup>(X,Y,Z) following a power increase of more than 10%, ensures that

## BASES

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### SURVEILLANCE REQUIREMENTS (continued)

they are verified as soon as RTP (or any other level for extended operation) is achieved. In the absence of these Frequency conditions, it is possible to increase power to RTP and operate for 31 days without verification of  $F_Q^C(X,Y,Z)$  and  $F_Q^M(X,Y,Z)$ . The Frequency condition is not intended to require verification of these parameters after every 10% increase in power level above the last verification. It only requires verification after a power level is achieved for extended operation that is 10% higher than that power at which  $F_Q(X,Y,Z)$  was last measured.

#### SR 3.2.1.1

Direct verification that  $F_Q^C(X,Y,Z)$  is within its specified limits involves increasing the overall measured  $F_Q(X,Y,Z)$  to allow for manufacturing tolerance and measurement uncertainties in order to obtain  $F_Q^C(X,Y,Z)$ .  $F_Q^C(X,Y,Z)$  is then compared to its specified limits.

The limit with which  $F_Q^C(X,Y,Z)$  is compared varies inversely with power above 50% RTP and directly with a function called  $K(Z)$  provided in the COLR.

The surveillance has been modified by a Note providing an allowance to not perform SR 3.2.1.1 if the Surveillance has been determined to be met based on the performance results of both SR 3.2.1.2 and SR 3.2.1.3. If both the AFD min margin and the  $f_2(\Delta I)$  min margin are  $\geq 0$ , then the steady state limit is met because these margins represent bounding limiting conditions. However, if AFD min margin or  $f_2(\Delta I)$  min margin is negative then a direct evaluation of the steady state limit is required to satisfy this surveillance requirement.

Performing this Surveillance in MODE 1 prior to exceeding 75% RTP ensures that the  $F_Q^C(X,Y,Z)$  limit is met when RTP is achieved, because peaking factors generally decrease as power level is increased.

If THERMAL POWER has been increased by  $\geq 10\%$  RTP since the last determination of  $F_Q^C(X,Y,Z)$ , another evaluation of this factor is required 12 hours after achieving equilibrium conditions at this higher power level (to ensure that  $F_Q^C(X,Y,Z)$  values are being reduced sufficiently with power increase to stay within the LCO limits).

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

## BASES

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### SURVEILLANCE REQUIREMENTS (continued)

#### SR 3.2.1.2 and SR 3.2.1.3

The nuclear design process includes calculations performed to determine that the core can be operated within the  $F_Q(X,Y,Z)$  limits. Because flux maps are taken in steady state conditions, the variations in power distribution resulting from normal operational maneuvers are not present in the flux map data. These variations are, however, conservatively calculated by considering a wide range of unit maneuvers in normal operation. The maximum peaking factor increase over steady state values, calculated as a function of both assembly and axial location  $(X,Y,Z)$ , has been included in the cycle specific limits  $BQDES(X,Y,Z)$  and  $BCDES(X,Y,Z)$  using margin factors  $MQ(X,Y,Z)$  and  $MC(X,Y,Z)$ , respectively (Reference 5).

No uncertainties are applied to  $F_Q^M(X,Y,Z)$  because the limits,  $BQDES(X,Y,Z)$  and  $BCDES(X,Y,Z)$ , have been adjusted for uncertainties.

$BQDES(X,Y,Z)$  and  $BCDES(X,Y,Z)$  limits are not applicable for the following axial core regions, measured in percent of core height:

- a. Lower core region, from 0 to 15% inclusive and
- b. Upper core region, from 85 to 100% inclusive.

The top and bottom 15% of the core are excluded from the evaluation because of the low probability that these regions would be more limiting in the safety analyses and because of the difficulty of making a precise measurement in these regions.

This Surveillance has been modified by a Note that may require that more frequent surveillances be performed based on future projections. If  $F_Q^M(X,Y,Z)$  is evaluated and found to be within the applicable limiting condition limits, an evaluation is required to account for any increase to  $F_Q^M(X,Y,Z)$  that may occur and cause the  $F_Q(X,Y,Z)$  limit to be exceeded before the next required  $F_Q(X,Y,Z)$  evaluation.

In addition to ensuring via surveillance that the heat flux hot channel factor is within its limits when a measurement is taken, there are also requirements to extrapolate trends in  $F_Q^M(X,Y,Z)$  for the last two measurements out to 31 EFPD beyond the most recent measurement. If the extrapolation yields an  $F_Q^M(X,Y,Z) > BQNOM(X,Y,Z)$ , further consideration is required.

BASES

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SURVEILLANCE REQUIREMENTS (continued)

The implications of these extrapolations are considered separately for both the operational and RPS heat flux hot channel factor limits. If the extrapolations of  $F_Q^M(X,Y,Z)$  are unfavorable, additional actions must be taken. These actions are to meet the  $F_Q(X,Y,Z)$  limit with the last  $F_Q^M(X,Y,Z)$  increased by the appropriate factor specified in the COLR or to evaluate  $F_Q^M(X,Y,Z)$  prior to the projected point in time when the extrapolated values are expected to exceed the extrapolated limits. These alternative requirements prevent  $F_Q(X,Y,Z)$  from exceeding its limit for any significant period of time without detection using the best available data.

Extrapolation is not required for the initial flux map taken after reaching equilibrium conditions following a refueling outage since the initial flux map establishes the baseline measurement for future trending.

$F_Q(X,Y,Z)$  is verified at power levels  $\geq 10\%$  RTP above the THERMAL POWER of its last verification within 12 hours after achieving equilibrium conditions to ensure that  $F_Q(X,Y,Z)$  is within its limit at higher power levels.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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REFERENCES	<ol style="list-style-type: none"> <li>1. 10 CFR 50.46, 1974.</li> <li>2. Regulatory Guide 1.77, Rev. 0, May 1974.</li> <li>3. 10 CFR 50, Appendix A, GDC 26.</li> <li>4. WCAP-7308-L-P-A, "Evaluation of Nuclear Hot Channel Factor Uncertainties," June 1988.</li> <li>5. BAW-10163PA "Core Operating Limit Methodology for Westinghouse-Designed PWRs" June 1989.</li> </ol>
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## B 3.2 POWER DISTRIBUTION LIMITS

### B 3.2.2 Nuclear Enthalpy Rise Hot Channel Factor (F<sub>ΔH</sub>(X,Y))

#### BASES

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**BACKGROUND** The purpose of this LCO is to establish limits on the power density at any point in the core so that the fuel design criteria are not exceeded and the accident analysis assumptions remain valid. The design limits on local (pellet) and integrated fuel rod peak power density are expressed in terms of hot channel factors. Control of the core power distribution with respect to these factors ensures that local conditions in the fuel rods and coolant channels do not challenge core integrity at any location during either normal operation or a postulated accident analyzed in the safety analyses.

F<sub>ΔH</sub>(X,Y) is defined as the ratio of the integral of the linear power along the fuel rod with the highest integrated power to the average integrated fuel rod power. Therefore, F<sub>ΔH</sub>(X,Y) is a measure of the maximum total power produced in a fuel rod.

F<sub>ΔH</sub>(X,Y) is sensitive to fuel loading patterns, bank insertion, and fuel burnup. F<sub>ΔH</sub>(X,Y) typically increases with control bank insertion and typically decreases with fuel burnup.

F<sub>ΔH</sub>(X,Y) is not directly measurable but is inferred from a power distribution map obtained with the movable incore detector system. Specifically, the results of the three dimensional power distribution map are analyzed by a computer to determine F<sub>ΔH</sub>(X,Y). An F<sub>ΔH</sub>(X,Y) evaluation requires obtaining an incore flux map in MODE 1. The incore flux map results provide the measured value ( F<sub>ΔH</sub><sup>M</sup>(X, Y) ) of F<sub>ΔH</sub>(X,Y) for each assembly location (X,Y). The F<sub>ΔH</sub> ratio (FDHR) is used in order to determine the F<sub>ΔH</sub> limit for the measured and design power distributions (Ref. 4). Then,

$$F_{\Delta H}^M(X,Y) = \frac{F_{\Delta H}^M(X, Y)}{MAP^M / AXIAL^M(X, Y)}$$

where MAP<sup>M</sup> is the maximum allowable peak from the COLR for the measured assembly power distribution at assembly location (X,Y) which accounts for calculational and measurement uncertainties, and AXIAL<sup>M</sup>(X, Y) is the measured ratio of the peak-to-average axial power at assembly location (X,Y).

BHDES(X,Y) is a cycle dependent design limit to preserve Departure from Nucleate Boiling(DNB) assumed for initial conditions at the time of limiting transients such as a Loss of Flow Accident (LOFA). BRDES(X,Y) is a

BASES

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BACKGROUND (continued)

cycle dependent design limit to preserve reactor protection system safety limits for DNB requirements (Ref. 4).

The expression for BHDES(X,Y) is:

$$\text{BHDES}(X,Y) = F\Delta\text{HR}^d(X,Y) * \text{MH}(X,Y)$$

where:  $F\Delta\text{HR}^d(X,Y) = \frac{F_{\Delta H}^d(X,Y)}{\text{MAP}^d / \text{AXIAL}^d(X,Y)}$

- MAP<sup>d</sup> is the maximum allowable peak from the COLR for the design assembly power distribution at assembly location (X,Y) which accounts for calculational and measurement uncertainties,
- AXIAL<sup>d</sup> (X, Y) is the design ratio of the peak-to-average axial power at assembly location (X,Y),
- F<sub>ΔH</sub><sup>d</sup> (X, Y) is the design F<sub>ΔH</sub> assembly location (X, Y), and
- MH(X,Y) is the minimum available margin ratio for initial condition DNB at the limiting conditions at assembly location (X,Y).

The expression for BRDES(X,Y) is:

$$\text{BRDES}(X,Y) = F\Delta\text{HR}^d(X,Y) * \text{MH}^s(X,Y)$$

where: MH<sup>s</sup>(X,Y) is the minimum available margin ratio for steady state DNB at the limiting conditions at assembly location (X,Y).

The reactor core is "operating as designed" if the measured steady state core power distribution agrees with prediction within statistical variation. This guarantees that the operating limits will preserve the thermal criteria in the applicable safety analyses. The core is "operating as designed" if the following relationship is satisfied:

$$F\Delta\text{HR}^M(X,Y) \leq \text{BHNOM}(X,Y)$$

where: BHNOM(X,Y) is the nominal design radial peaking factor for an assembly at core location (X,Y) increased by an allowance for the expected deviation between the measured and predicted design power distribution. This factor is calculated at least every 31 EFPD. However, during power operation, the global power distribution is monitored by LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," which address directly and continuously measured process variables.

BASES

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BACKGROUND (continued)

The COLR provides peaking factor limits that ensure that the design basis value of the DNB is met for normal operation, operational transients, and any transient condition arising from events of moderate frequency. The DNB design basis precludes DNB and is met by limiting the minimum local DNB heat flux ratio to the design limit value using an NRC approved critical heat flux correlation. All DNB limited transient events are assumed to begin with an F<sub>ΔH</sub>(X,Y) value that satisfies the LCO requirements.

Operation outside the LCO limits may produce unacceptable consequences if a DNB limiting event occurs. The DNB design basis ensures that there is no overheating of the fuel that results in possible cladding perforation with the release of fission products to the reactor coolant.

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APPLICABLE  
SAFETY  
ANALYSES

Limits on F<sub>ΔH</sub>(X,Y) preclude core power distributions that exceed the following fuel design limits:

- a. There must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hottest fuel rod in the core does not experience a DNB condition,
- b. During a large break loss of coolant accident (LOCA), peak cladding temperature (PCT) must not exceed 2200°F,
- c. During an ejected rod accident, the energy deposition to the fuel must not exceed 280 cal/gm (Ref. 1), and
- d. Fuel design limits required by GDC 26 (Ref. 2) for the condition when control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn.

For transients that may be DNB limited, the Reactor Coolant System flow and F<sub>ΔH</sub>(X,Y) are the core parameters of most importance. The limits on F<sub>ΔH</sub>(X,Y) ensure that the DNB design basis is met for normal operation, operational transients, and any transients arising from events of moderate frequency. The DNB design basis is met by limiting the minimum local DNB heat flux ratio to the design limit value using an NRC approved critical heat flux correlation. This value provides a high degree of assurance that the hottest fuel rod in the core does not experience a DNB.

The allowable F<sub>ΔH</sub>(X,Y), F<sub>ΔH</sub> min margin and f<sub>1</sub>(ΔI) min margin, increase with decreasing power level. This functionality in F<sub>ΔH</sub>(X,Y) is

BASES

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APPLICABLE SAFETY ANALYSES (continued)

included in the analyses that provide the Reactor Core Safety Limits (SLs) of SL 2.1.1. Therefore, any DNB events in which the calculation of the core limits is modeled implicitly use this variable value of F<sub>ΔH</sub>(X,Y) in the analyses. Likewise, all transients that may be DNB limited are assumed to begin with F<sub>ΔH</sub> min margin and f<sub>1</sub> (ΔI) min margin.

The LOCA safety analysis indirectly models F<sub>ΔH</sub>(X,Y) as an input parameter. The Nuclear Heat Flux Hot Channel Factor (F<sub>Q</sub>(X,Y,Z)) and the axial peaking factors are inserted directly into the LOCA safety analyses that verify the acceptability of the resulting peak cladding temperature (Ref. 3).

The fuel is protected in part by Technical Specifications, which ensure that the initial conditions assumed in the safety and accident analyses remain valid. The following LCOs ensure this: LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," LCO 3.1.6, "Control Bank Insertion Limits," LCO 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor F<sub>ΔH</sub>(X,Y)," and LCO 3.2.1, "Heat Flux Hot Channel Factor (F<sub>Q</sub>(X,Y,Z))."

F<sub>ΔH</sub>(X,Y) and F<sub>Q</sub>(X,Y,Z) are indirectly measured periodically using the movable incore detector system. Measurements are generally taken with the core at, or near, steady state conditions. Core monitoring and control under transient conditions (Condition 1 events) are accomplished by operating the core within the limits of the LCOs on AFD, QPTR, and Bank Insertion Limits.

F<sub>ΔH</sub>(X,Y) satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

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LCO

The LCO states that F<sub>ΔH</sub>(X,Y) shall be less than the limits provided in the COLR. This LCO relationship must be satisfied even if the core is operating at limiting conditions. This requires adjustment to the measured F<sub>ΔH</sub>(X,Y) to account for limiting conditions and the differences between design and measured conditions. The adjustments are accounted for by comparing F<sub>ΔHR</sub><sup>M</sup>(X,Y) to the limits BHDES(X,Y) and BRDES(X,Y). Therefore, if the F<sub>ΔH</sub> min margin is ≥ 0 and f<sub>1</sub>(ΔI) min margin ≥ 0 the LCO is satisfied.

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APPLICABILITY

The F<sub>ΔH</sub>(X,Y) limits must be maintained in MODE 1 to preclude core power distributions from exceeding the fuel design limits for DNBR and PCT. Applicability in other modes is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the coolant to require a limit on the distribution of core power. Specifically, the design bases events that are sensitive to

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BASES

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APPLICABILITY (continued)

F<sub>ΔH</sub>(X,Y) in other modes (MODES 2 through 5) have significant margin to DNB, and therefore, there is no need to restrict F<sub>ΔH</sub>(X,Y) in these modes.

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ACTIONS

The % F<sub>ΔH</sub> margin is based on the relationship between F<sub>ΔHR</sub><sup>M</sup>(X,Y) and the limit, BHDES (X,Y), as follows:

$$\% F_{\Delta H} \text{ Margin} = \left( 1 - \frac{F_{\Delta HR}^M(X,Y)}{BHDES(X,Y)} \right) \times 100\%$$

If the reactor core is "operating as designed", then F<sub>ΔHR</sub><sup>M</sup>(X,Y) is less than BHDES (X,Y) and calculation of %F<sub>ΔH</sub> margin is not required. If the %F<sub>ΔH</sub> margin is less than zero, then F<sub>ΔHR</sub><sup>M</sup>(X,Y) is greater than BHDES (X, Y) and the F<sub>ΔH</sub>(X,Y) limits may not be adequate to prevent exceeding the initial DNB conditions assumed for transients such as a LOFA. BHDES (X,Y) represents the maximum allowable design radial peaking factors which ensures that the initial condition DNB will be preserved for operation within the LCO limits, and includes allowances for calculational and measurement uncertainties. The F<sub>ΔH</sub> min margin is the minimum for all core locations examined.

Condition A is modified by a Note that requires that Required Actions A.3 and A.5 must be completed whenever Condition A is entered. If F<sub>ΔH</sub> min margin < 0 is restored to within limits prior to completion of the THERMAL POWER reduction in Required Action A.1, compliance with Required Actions A.3 and A.5 must be met.

However, if power is reduced below 50% RTP, Required Action A.5 requires that another determination of F<sub>ΔH</sub> min margin must be verified prior to exceeding 50% RTP, prior to exceeding 75% RTP, and within 24 hours after reaching or exceeding 95% RTP.

A.1 and A.2

If the value of F<sub>ΔH</sub> min margin is not restored to within its specified limit either by adjusting a misaligned rod or by reducing THERMAL POWER, the alternative option is to reduce allowable THERMAL POWER from RTP by at least RRH% (where RRH = Thermal power reduction required to compensate for each 1% that F<sub>ΔH</sub>(X,Y) exceeds its limit) multiplied by the F<sub>ΔH</sub> min margin in accordance with Required Action A.1 and reduce the Power Range Neutron Flux - High trip setpoints, as specified in TS Table 3.3.1-1 by ≥ RRH% multiplied times the F<sub>ΔH</sub> min margin in accordance with Required Action A.2. Reducing allowable RTP by at least RRH% multiplied by the F<sub>ΔH</sub> min margin increases the DNB

BASES

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## ACTIONS (continued)

margin and does not likely cause the DNBR limit to be violated in steady state operation. The reduction in trip setpoints ensures that continuing operation remains at an acceptable low power level with adequate DNBR margin. The allowed Completion Time of 2 hours for Required Action A.1 provides an acceptable time to reach the required power level from full power operation without allowing the plant to remain in an unacceptable condition for an extended period of time.

The allowed Completion Time of 72 hours to reset the trip setpoints per Required Action A.2 recognizes that, once power is reduced, the safety analysis assumptions are satisfied and there is no urgent need to reduce the trip setpoints. This is a sensitive operation that may inadvertently trip the Reactor Protection System.

A.3

Once the allowable power level has been reduced by at least RRH% multiplied by the  $F_{\Delta H}$  min margin per Required Action A.1, an incore flux map (SR 3.2.2.1) must be obtained and the  $F_{\Delta H}$  min margin is verified  $\geq 0$  at the lower power level. The unit is provided 22 additional hours to perform this task over and above the 2 hours allowed by Action A.1. The Completion Time of 24 hours is acceptable because of the increase in the DNB margin, which is obtained at lower power levels, and the low probability of having a DNB limiting event within this 24 hour period. Additionally, operating experience has indicated that this Completion Time is sufficient to obtain the incore flux map, perform the required calculations, and evaluate  $F_{\Delta H}$  min margin.

A.4

If the value of  $F\Delta HR^M(X,Y)$  is not restored to within its specified limit, Overtemperature  $\Delta T$  K1 (OT $\Delta T$  K1) term is required to be reduced by at least TRH multiplied by the  $F\Delta H$  min margin. The value of TRH is provided in the COLR. Completing Required Action A.4 ensures protection against the consequences of transients since this adjustment limits the peak transient power level which can be achieved during an anticipated operational occurrence. Also, completing Required Action A.4 within the allowed Completion Time of 48 hours is sufficient considering the small likelihood of a limiting transient in this time period.

BASES

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ACTIONS (continued)

A.5

Verification that F<sub>ΔH</sub> min margin is ≥ 0 after an out of limit occurrence ensures that the cause that led to the F<sub>ΔH</sub> min margin exceeding its limit is corrected, and that subsequent operation proceeds within the LCO limit. This Action demonstrates that the F<sub>ΔH</sub> min margin limit is ≥ 0 prior to exceeding 50% RTP, again prior to exceeding 75% RTP, and within 24 hours after THERMAL POWER is ≥ 95% RTP.

This Required Action is modified by a Note that states that THERMAL POWER does not have to be reduced prior to performing this Action.

B.1

The %f<sub>1</sub>(ΔI) margin is based on the relationship between FΔHR<sup>M</sup>(X,Y) and the limit, BRDES (X,Y), as follows:

$$\% f_1(\Delta I)Margin = \left( 1 - \frac{F\Delta HR^M(X,Y)}{BRDES(X,Y)} \right) \times 100\%$$

If the reactor core is "operating as designed", then FΔHR<sup>M</sup>(X,Y) is less than BRDES (X,Y) and calculation of %f<sub>1</sub>(ΔI) margin is not required. If the %f<sub>1</sub>(ΔI) margin is less than zero, then FΔHR<sup>M</sup>(X,Y) is greater than BRDES (X, Y) and the OTΔT setpoint limits may not be adequate to prevent exceeding DNB requirements.

BRDES (X,Y) represents the maximum allowable design radial peaking factors which ensure that the steady state DNBR limit will be preserved for operation within the LCO limits, including allowances for calculational and measurement uncertainties.

Required Action B.1 requires the reduction of the OTΔT K1 term by at least TRH multiplied by the f<sub>1</sub>(ΔI) min margin. TRH is the amount of OTΔT K1 setpoint reduction required to compensate for each 1% that F<sub>ΔH</sub>(X,Y) exceeds the limit provided in the COLR. Completing Required Action B.1 within the allowed Completion Time of 48 hours, restricts F<sub>ΔH</sub>(X,Y) such that even if a transient occurred, DNB requirements are met. The f<sub>1</sub>(ΔI) min margin is the minimum % of f<sub>1</sub>(ΔI) margin for all core locations examined.

BASES

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ACTIONS (continued)

C.1

When Required Actions A.1 through A.5, and B.1, cannot be completed within their required Completion Times, the plant must be placed in a mode in which the LCO requirements are not applicable. This is done by placing the plant in at least MODE 2 within 6 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience regarding the time required to reach MODE 2 from full power conditions in an orderly manner and without challenging plant systems.

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SURVEILLANCE  
REQUIREMENTS

SR 3.2.2.1 and SR 3.2.2.2 are modified by a Note. It states that, "Not required to be performed until 12 hours after an equilibrium power level has been achieved at which a power distribution map can be obtained." SR 3.2.2.1 and SR 3.2.2.2 require using the incore detector system to provide the necessary data to create a power distribution map. To provide the necessary data, MODE 1 needs to be entered, power escalated, stabilized and equilibrium conditions established at some higher power level. These surveillances could not be satisfactorily performed if the requirement for performance of the Surveillances was included in MODE 2 prior to entering MODE 1.

In a reload core,  $F_{\Delta H}^M(X,Y)$  could not have previously been measured, therefore, there is a Frequency condition, applicable only for reload cores, that requires determination of these parameters before exceeding 75% RTP. This ensures that some determination of  $F_{\Delta H}^M(X,Y)$  is made at a lower power level at which adequate margin is available before going to 100% RTP.

SR 3.2.2.1 and SR 3.2.2.2

In addition to ensuring via Surveillance that the nuclear enthalpy rise hot channel factor is within its limits when a measurement is taken, there are also requirements to extrapolate trends in  $F_{\Delta H}^M(X,Y)$  for the last two measurements out to 31 EFPD beyond the most recent measurement. If the extrapolation yields an  $F_{\Delta HR}^M(X,Y) > BHNOM(X,Y)$ , further consideration is required.

The implications of these extrapolations are considered separately for BHDES(X,Y) and BRDES(X,Y) limits. If the extrapolations of  $F_{\Delta H}^M(X,Y)$  are unfavorable, additional actions must be taken. These actions are to meet the  $F_{\Delta H}(X,Y)$  limit with the last  $F_{\Delta H}^M(X,Y)$  increased by the appropriate factor specified in the COLR or to evaluate  $F_{\Delta H}^M(X,Y)$  prior to the projected

## BASES

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### SURVEILLANCE REQUIREMENTS (continued)

point in time when the extrapolated values are expected to exceed the extrapolated limits. These alternative requirements attempt to prevent F<sub>ΔH</sub>(X,Y) from exceeding its limit for any significant period of time without detection using the best available data.

Extrapolation is not required for the initial flux map taken after reaching equilibrium conditions following a refueling outage since the initial flux map establishes the baseline measurement for future trending.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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### REFERENCES

1. Regulatory Guide 1.77, Rev. 0, May 1974.
  2. 10 CFR 50, Appendix A, GDC 26.
  3. 10 CFR 50.46.
  4. BAW-10163P-A, Revision 0, "Core Operating Limit Methodology for Westinghouse-Designed PWRs," June 1989.
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## B 3.2 POWER DISTRIBUTION LIMITS

### B 3.2.3 AXIAL FLUX DIFFERENCE (AFD)

#### BASES

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**BACKGROUND** The purpose of this LCO is to establish limits on the values of the AFD in order to limit the amount of axial power distribution skewing to either the top or bottom of the core. By limiting the amount of power distribution skewing, core peaking factors are consistent with the assumptions used in the safety analyses. Limiting power distribution skewing over time also minimizes the xenon distribution skewing, which is a significant factor in axial power distribution control.

The AFD limits are selected by considering a range of axial xenon distributions that may occur as a result of large variations of the AFD. Subsequently, power peaking factors and power distributions are examined to ensure that the loss of coolant accident (LOCA), loss of flow accident, and anticipated transient limits are met. Violation of the AFD limits invalidate the conclusions of the accident and transient analyses with regard to fuel cladding integrity.

The AFD is monitored on an automatic basis using the unit process computer, which has an AFD monitor alarm. The computer determines the 1 minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for two or more OPERABLE excore channels is outside its specified limits.

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**APPLICABLE SAFETY ANALYSES** The AFD is a measure of the axial power distribution skewing to either the top or bottom half of the core. The AFD is sensitive to many core related parameters such as control bank positions, core power level, axial burnup, axial xenon distribution, and, to a lesser extent, reactor coolant temperature and boron concentration.

The allowed range of the AFD is used in the nuclear design process to confirm that operation within these limits produces core peaking factors and axial power distributions that meet safety analysis requirements (Ref.1).

The limits on the AFD ensure that the Heat Flux Hot Channel Factor ( $F_Q(X,Y,Z)$ ) is not exceeded during either normal operation or in the event of xenon redistribution following power changes. The limits on the AFD also restrict the range of power distributions that are used as initial conditions in the analyses of Condition 2, 3, or 4 events. A Condition 4 event significantly affected by the initial axial power distribution, as indicated by AFD, is the LOCA. A Condition 3 event significantly affected by AFD is the Complete Loss of RCS Flow event.

BASES

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APPLICABLE SAFETY ANALYSES (continued)

A Condition 2 event significantly affected by AFD is the Uncontrolled RCCA Bank Withdrawal at Power Event (Ref. 2). Condition 2 accidents, simulated to begin from within the AFD limits are used to confirm the adequacy of the Overpower  $\Delta T$  and Overtemperature  $\Delta T$  trip setpoints.

The limits on the AFD satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

---

LCO

The shape of the power profile in the axial (i.e., the vertical) direction is largely under the control of the operator through the manual operation of the control banks or automatic motion of control banks. The automatic motion of the control banks is in response to temperature deviations resulting from manual operation of the Chemical and Volume Control System to change boron concentration or from power level changes.

Signals are available to the operator from the Nuclear Instrumentation System (NIS) excore neutron detectors (Refs. 1 and 3). Separate signals are taken from the top and bottom detectors. The AFD is defined as the difference in normalized flux signals between the top and bottom excore detectors in each detector well. For convenience, this flux difference is converted to provide flux difference units expressed as a percentage and labeled as  $\% \Delta$  flux or  $\% \Delta I$ .

The AFD limits are provided in the COLR. The AFD limits resulting from analysis of core power distributions relative to the initial condition peaking limits comprise a power-dependent envelope of acceptable AFD values. During steady-state operation, the core normally is controlled to a target AFD within a narrow (approximately  $\pm 5\%$  AFD) band. However, the limiting AFD values may be somewhat greater than the extremes of the normal operating band.

Violating this LCO on the AFD could produce unacceptable consequences if a Condition 2, 3, or 4 event occurs while the AFD is outside its specified limits.

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APPLICABILITY

The AFD requirements are applicable in MODE 1 greater than or equal to 50% RTP when the combination of THERMAL POWER and core peaking factors are of primary importance in safety analysis.

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ACTIONS

A.1

As an alternative to restoring the AFD to within its specified limits, Required Action A.1 requires a THERMAL POWER reduction to  $< 50\%$  RTP. This places the core in a condition for which the value of the AFD is not important in the applicable safety analyses. A Completion

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BASES

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ACTIONS (continued)

Time of 30 minutes is reasonable, based on operating experience, to reach 50% RTP without challenging plant systems.

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SURVEILLANCE  
REQUIREMENTS

SR 3.2.3.1

This Surveillance verifies that the AFD, as indicated by the NIS excore channel, is within its specified limits.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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REFERENCES

1. BAW-10163P-A, Revision 0, "Core Operating Limit Methodology for Westinghouse-Designed PWRs," June 1989.
  2. UFSAR, Chapter 15.
  3. UFSAR, Section 4.3.2.
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## B 3.2 POWER DISTRIBUTION LIMITS

### B 3.2.4 QUADRANT POWER TILT RATIO (QPTR)

#### BASES

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**BACKGROUND** The QPTR limit ensures that the gross radial power distribution remains consistent with the design values used in the safety analyses. Precise radial power distribution measurements are made during startup testing, after refueling, and periodically during power operation.

The power density at any point in the core must be limited so that the fuel design criteria are maintained. Together, LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.2.4, and LCO 3.1.6, "Control Bank Insertion Limits," provide limits on process variables that characterize and control the three dimensional power distribution of the reactor core. Control of these variables ensures that the core operates within the fuel design criteria and that the power distribution remains within the bounds used in the safety analyses.

**APPLICABLE SAFETY ANALYSES** This LCO precludes core power distributions that violate the following fuel design criteria:

- a. During a large break loss of coolant accident, the peak cladding temperature must not exceed 2200°F (Ref. 1),
- b. During a loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 departure from nucleate boiling (DNB) criterion) that the hot fuel rod in the core does not experience a DNB condition,
- c. During an ejected rod accident, the energy deposition to the fuel must not exceed 280 cal/gm (Ref. 2), and
- d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 3).

The LCO limits on the AFD, the QPTR, the Heat Flux Hot Channel Factor ( $F_Q(X,Y,Z)$ ), the Nuclear Enthalpy Rise Hot Channel Factor ( $F_{\Delta H}(X,Y)$ ) and control bank insertion are established to preclude core power distributions that exceed the safety analyses limits.

The QPTR limits ensure that  $F_{\Delta H}(X,Y)$  and  $F_Q(X,Y,Z)$  remain below their limiting values by preventing an undetected change in the gross radial power distribution.

BASES

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APPLICABLE SAFETY ANALYSES (continued)

In MODE 1, the  $F_{\Delta H}(X,Y)$  and  $F_Q(X,Y,Z)$  limits must be maintained to preclude core power distributions from exceeding design limits assumed in the safety analyses.

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

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LCO The QPTR limit of 1.02, at which corrective action is required, provides a margin of protection for both the DNB ratio and linear heat generation rate contributing to excessive power peaks resulting from X-Y plane power tilts. A limiting QPTR of 1.02 can be tolerated before the margin for uncertainty in  $F_Q(X,Y,Z)$  and  $F_{\Delta H}(X,Y)$  is possibly challenged.

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APPLICABILITY The QPTR limit must be maintained in MODE 1 with THERMAL POWER > 50% RTP to prevent core power distributions from exceeding the design limits.

Applicability in MODE 1  $\leq$  50% RTP and in other MODES is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the reactor coolant to require the implementation of a QPTR limit on the distribution of core power. The QPTR limit in these conditions is, therefore, not important. Note that the  $F_{\Delta H}(X,Y)$  and  $F_Q(X,Y,Z)$  LCOs still apply, but allow progressively higher peaking factors at 50% RTP or lower.

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ACTIONS

A.1

With the QPTR exceeding its limit, a power level reduction of 3% RTP for each 1% by which the QPTR exceeds 1.02 is a conservative tradeoff of total core power with peak linear power. The Completion Time of 2 hours allows sufficient time to identify the cause and correct the tilt. Note that the power reduction itself may cause a change in the tilted condition.

The maximum allowable power level initially determined by Required Action A.1 may be affected by subsequent determinations of QPTR. Increases in QPTR would require power reduction within 2 hours of QPTR determination, if necessary to comply with the decreased maximum allowable power level. Decreases in QPTR would allow increasing the maximum allowable power level and increasing power up to this revised limit.

BASES

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ACTIONS (continued)

A.2

After completion of Required Action A.1, the QPTR alarm may still be in its alarmed state. As such, any additional changes in the QPTR are detected by requiring a check of the QPTR once per 12 hours thereafter. A 12 hour Completion Time is sufficient because any additional change in QPTR would be relatively slow.

A.3

The peaking factors  $F_Q(X,Y,Z)$  and  $F_{\Delta H}(X,Y)$  are of primary importance in ensuring that the power distribution remains consistent with the initial conditions used in the safety analyses. Performing SRs on  $F_{\Delta H}(X,Y)$  and  $F_Q(X,Y,Z)$  within the Completion Time of 24 hours after achieving equilibrium conditions from a Thermal Power reduction per Required Action A.1 ensures that these primary indicators of power distribution are within their respective limits. Equilibrium conditions are achieved when the core is sufficiently stable at intended operating conditions to support flux mapping. A Completion Time of 24 hours after achieving equilibrium conditions from Thermal Power reduction per Required Action A.1 takes into consideration the rate at which peaking factors are likely to change, and the time required to stabilize the plant and perform a flux map. If these peaking factors are not within their limits, the Required Actions of the applicable LCOs of these Surveillances provide an appropriate response for the abnormal condition. If the QPTR remains above its specified limit, the peaking factor surveillances are required each 7 days thereafter to evaluate  $F_{\Delta H}(X,Y)$  and  $F_Q(X,Y,Z)$  with changes in power distribution. Relatively small changes are expected due to either burnup and xenon redistribution or correction of the cause for exceeding the QPTR limit.

A.4

Although  $F_{\Delta H}(X,Y)$  and  $F_Q(X,Y,Z)$  are of primary importance as initial conditions in the safety analyses, other changes in the power distribution may occur as the QPTR limit is exceeded and may have an impact on the validity of the safety analysis. A change in the power distribution can affect such reactor parameters as bank worths and peaking factors for rod malfunction accidents. When the QPTR exceeds its limit, it does not

BASES

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## ACTIONS (continued)

necessarily mean a safety concern exists. It does mean that there is an indication of a change in the gross radial power distribution that requires an investigation and evaluation that is accomplished by examining the incore power distribution. Specifically, the core peaking factors and the quadrant tilt must be evaluated because they are the factors that best characterize the core power distribution. This re-evaluation is required to ensure that, before increasing THERMAL POWER to above the limit of Required Action A.1, the reactor core conditions are consistent with the assumptions in the safety analyses.

A.5

If the QPTR is still exceeding the 1.02 limit and a re-evaluation of the safety analysis is completed and shows that safety requirements are met, the excore detectors shall be normalized to restore QPTR to within limits prior to increasing THERMAL POWER to above the limit of Required Action A.1. Normalization is accomplished in such a manner that the indicated QPTR following normalization is near 1.02. This is done to detect any subsequent significant changes in QPTR.

Required Action A.5 is modified by two Notes. Note 1 states that the QPTR shall not be restored to within limits by excore detector normalization until after the re-evaluation of the safety analysis has determined that core conditions at RTP are within the safety analysis assumptions (i.e., Required Action A.4). Note 2 states that if Required Action A.5 is performed, then Required Action A.6 shall be performed. Required Action A.5 normalizes the excore detectors to restore QPTR to within limits, which restores compliance with LCO 3.2.4. Thus, Note 2 prevents exiting the Actions prior to completing flux mapping to verify peaking factors, per Required Action A.6. These Notes are intended to prevent any ambiguity about the required sequence of actions.

A.6

Once the flux tilt is restored to within limits (i.e., Required Action A.5 is performed), it is acceptable to return to full power operation. However, as an added check that the core power distribution is consistent with the safety analysis assumptions, Required Action A.6 requires verification

BASES

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ACTIONS (continued)

that  $F_Q(X,Y,Z)$  and  $F_{\Delta H}(X,Y)$  are within their specified limits within 24 hours of achieving equilibrium conditions at RTP. As an added precaution, if the core power does not reach equilibrium conditions at RTP within 24 hours, but is increased slowly, then the peaking factor surveillances must be performed within 48 hours after increasing THERMAL POWER above the limit of Required Action A.1. These Completion Times are intended to allow adequate time to increase THERMAL POWER to above the limit of Required Action A.1, while not permitting the core to remain with unconfirmed power distributions for extended periods of time.

Required Action A.6 is modified by a Note that states that the peaking factor surveillances may only be done after the excore detectors have been normalized to restore QPTR to within limits (i.e., Required Action A.5). The intent of this Note is to have the peaking factor surveillances performed at operating power levels, which can only be accomplished after the excore detectors are normalized to restore QPTR to within limits and the core returned to power.

B.1

If Required Actions A.1 through A.6 are not completed within their associated Completion Times, the unit must be brought to a MODE or condition in which the requirements do not apply. To achieve this status, THERMAL POWER must be reduced to  $\leq 50\%$  RTP within 4 hours. The allowed Completion Time of 4 hours is reasonable, based on operating experience regarding the amount of time required to reach the reduced power level without challenging plant systems.

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SURVEILLANCE  
REQUIREMENTS

SR 3.2.4.1

SR 3.2.4.1 is modified by two Notes. Note 1 allows QPTR to be calculated with three power range channels if THERMAL POWER is  $\leq 75\%$  RTP and the input from one Power Range Neutron Flux channel is inoperable. Note 2 allows performance of SR 3.2.4.2 in lieu of SR 3.2.4.1.

This Surveillance verifies that the QPTR, as indicated by the Nuclear Instrumentation System (NIS) excore channels, is within its limits.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

## BASES

## SURVEILLANCE REQUIREMENTS (continued)

For those causes of QPTR that occur quickly (e.g., a dropped rod), there typically are other indications of abnormality that prompt a verification of core power tilt.

SR 3.2.4.2

This Surveillance is modified by a Note, which states that the surveillance is only required to be performed if input to QPTR from one or more Power Range Neutron Flux channels are inoperable with THERMAL POWER > 75% RTP.

With an NIS power range channel inoperable, tilt monitoring for a portion of the reactor core becomes degraded. Large tilts are likely detected with the remaining channels, but the capability for detection of small power tilts in some quadrants is decreased.

With input to QPTR from one or more Power Range Neutron Flux channels inoperable and with THERMAL POWER > 75% RTP, the surveillance is initially performed within 12 hours. Thereafter, the Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

For purposes of monitoring the QPTR when one power range channel is inoperable, the moveable incore detectors are used to confirm that the normalized symmetric power distribution is consistent with the indicated QPTR and any previous data indicating a tilt. The incore detector monitoring is performed with a full incore flux map or two sets of four thimble locations with quarter core symmetry. The two sets of four symmetric thimbles is a set of eight unique detector locations. These locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11, and N-8.

The symmetric thimble flux map can be used to generate symmetric thimble "tilt." This can be compared to a reference symmetric thimble tilt, from the most recent full core flux map, to generate an incore QPTR. Therefore, incore monitoring of QPTR can be used to confirm that QPTR is within limits.

With one NIS channel inoperable, the indicated tilt may be changed from the value indicated with all four channels OPERABLE. To confirm that no change in tilt has actually occurred, which might cause the QPTR limit to be exceeded, the incore result may be compared against previous flux maps either using the symmetric thimbles as described above or a complete flux map. Nominally, quadrant tilt from the Surveillance should be within 2% of the tilt shown by the most recent flux map data.

BASES

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- REFERENCES
1. 10 CFR 50.46.
  2. Regulatory Guide 1.77, Rev. 0, May 1974.
  3. 10 CFR 50, Appendix A, GDC 26.
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## B 3.3 INSTRUMENTATION

### B 3.3.1 Reactor Trip System (RTS) Instrumentation

#### BASES

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**BACKGROUND** The RTS initiates a unit shutdown, based on the values of selected unit parameters, to protect against violating the core fuel design limits and Reactor Coolant System (RCS) pressure boundary during Anticipated Operational Occurrences (AOOs) and to assist the Engineered Safety Features (ESF) Systems in mitigating accidents.

The protection and monitoring systems have been designed to assure safe operation of the reactor. This is achieved by specifying Limiting Safety System Settings (LSSS) in terms of parameters directly monitored by the RTS, as well as specifying LCOs on other reactor system parameters and equipment performance.

Technical Specifications are required by 10 CFR 50.36 to include LSSS. LSSS are defined by the regulation as settings for automatic protective devices related to those variables having significant safety functions. The regulation also states, "Where a LSSS is specified for a variable on which a safety limit has been placed, the setting must be chosen so that automatic protective actions will correct the abnormal situation before a Safety Limit (SL) is exceeded." The Analytical Limit is the limit of the process variable at which a protective action is initiated, as established by the safety analysis, to ensure that a SL is not exceeded. Any automatic protection action that occurs on reaching the Analytical Limit therefore ensures that the SL is not exceeded. However, in practice, the actual settings for automatic protection channels must be chosen to be more conservative than the Analytical Limit to account for instrument loop uncertainties related to the setting at which the automatic protective action would actually occur.

The Nominal Trip Setpoint (NTSP) specified in Table 3.3.1-1 is a predetermined setting for a protection channel chosen to ensure automatic actuation prior to the process variable reaching the Analytical Limit and thus ensuring that the SL would not be exceeded. As such, the NTSP accounts for uncertainties in setting the channel (e.g., calibration), uncertainties in how the channel might actually perform (e.g., repeatability), changes in the point of action of the channel over time (e.g., drift during surveillance intervals), and any other factors which may influence its actual performance (e.g., harsh accident environments). In this manner, the NTSP ensures that SLs are not exceeded. Therefore, the NTSP meets the definition of an LSSS (Ref. 1).

## BASES

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### BACKGROUND (continued)

Technical Specifications contain values related to the OPERABILITY of equipment required for safe operation of the facility. OPERABLE is defined in Technical Specifications as "...being capable of performing its safety functions(s)." Relying solely on the NTSP to define OPERABILITY in Technical Specifications would be an overly restrictive requirement if it were applied as an OPERABILITY limit for the "as-found" value of a protection channel setting during a surveillance. This would result in Technical Specification compliance problems, as well as reports and corrective actions required by the rule which are not necessary to ensure safety. For example, an automatic protection channel with a setting that has been found to be different from the NTSP due to some drift of the setting may still be OPERABLE since drift is to be expected. This expected drift would have been specifically accounted for in the setpoint methodology for calculating the NTSP and thus the automatic protective action would still have ensured that the SL would not be exceeded with the "as-found" setting of the protection channel.

Therefore, the channel would still be OPERABLE since it would have performed its safety function and the only corrective action required would be to reset the channel within the established as-left tolerance around the NTSP to account for further drift during the next surveillance interval.

During AOOs, which are those events expected to occur one or more times during the unit life, the acceptable limits are:

1. The Departure from Nucleate Boiling Ratio (DNBR) shall be maintained above the Safety Limit (SL) value to prevent Departure from Nucleate Boiling (DNB),
2. Fuel centerline melt shall not occur, and
3. The RCS pressure SL of 2735 psig shall not be exceeded.

Operation within the SLs of Specification 2.0, "Safety Limits (SLs)," also maintains the above values and assures that offsite dose will be within the 10 CFR 50 and 10 CFR 100 criteria during AOOs.

Accidents are events that are analyzed even though they are not expected to occur during the unit life. The acceptable limit during accidents is that offsite dose shall be maintained within an acceptable fraction of 10 CFR 100 limits. Different accident categories are allowed a different fraction of these limits, based on probability of occurrence. Meeting the acceptable dose limit for an accident category is considered having acceptable consequences for that event.

## BASES

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### BACKGROUND (continued)

The RTS instrumentation is segmented into four distinct but interconnected modules as illustrated in Figure 7.2.2-2, UFSAR, Chapter 7 (Ref. 2), and as identified below:

1. Field transmitters or process sensors: provide a measurable electronic signal based upon the physical characteristics of the parameter being measured,
2. Signal Process Control and Protection System, including Process Protection System, Nuclear Instrumentation System (NIS), field contacts, and protection channel sets: provides analog to digital conversion (Digital Protection System), signal conditioning, setpoint comparison, process algorithm actuation, compatible electrical signal output to protection system channels, and control board/control room/miscellaneous indications,
3. Solid State Protection System (SSPS), including input, logic, and output bays: initiates proper unit shutdown and/or ESF actuation in accordance with the defined logic, which is based on the bistable, setpoint comparator, or contact outputs from the signal process control and protection system, and
4. Reactor trip switchgear, including reactor trip breakers and bypass breakers: provides the means to interrupt power to the Control Rod Drive Mechanisms (CRDMs) and allows the Rod Cluster Control Assemblies (RCCAs), or "rods," to fall into the core and shut down the reactor. The bypass breakers allow testing of the reactor trip breakers at power.

#### Field Transmitters or Sensors

To meet the design demands for redundancy and reliability, more than one, and often as many as four, field transmitters or sensors are used to measure unit parameters. To account for the calibration tolerances and instrument drift, which are assumed to occur between calibrations, statistical allowances are provided in the NTSP and Allowable Value. The OPERABILITY of each transmitter or sensor is determined by either "as-found" calibration data evaluated during the CHANNEL CALIBRATION or by qualitative assessment of field transmitter or sensor as related to the channel behavior observed during performance of the CHANNEL CHECK.

## BASES

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### BACKGROUND (continued)

#### Signal Process Control and Protection System

Generally, three or four channels of process control equipment are used for the signal processing of unit parameters measured by the field instruments. The process control equipment provides analog to digital conversion (Digital Protection System), signal conditioning, comparable output signals for instruments located on the main control board, and comparison of measured input signals with NTSPs derived from Analytical Limits established by the safety analyses. Analytical Limits are defined in UFSAR, Chapter 7 (Ref. 2), Chapter 6 (Ref. 3), and Chapter 15 (Ref. 4). If the measured value of a unit parameter exceeds the predetermined setpoint, an output from a bistable, setpoint comparator, or contact is forwarded to the SSPS for decision evaluation. Channel separation is maintained up to and through the input bays. However, not all unit parameters require four channels of sensor measurement and signal processing. Some unit parameters provide input only to the SSPS, while others provide input to the SSPS, the main control board, the unit computer, and one or more control systems.

Generally, if a parameter is used only for input to the protection circuits, three channels with a two-out-of-three logic are sufficient to provide the required reliability and redundancy. If one channel fails in a direction that would not result in a partial Function trip, the Function is still OPERABLE with a two-out-of-two logic. If one channel fails, such that a partial Function trip occurs, a trip will not occur and the Function is still OPERABLE with a one-out-of-two logic.

Generally, if a parameter is used for input to the SSPS and a control function, four channels with a two-out-of-four logic are sufficient to provide the required reliability and redundancy. The circuit must be able to withstand both an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation. Again, a single failure will neither cause nor prevent the protection function actuation. These requirements are described in IEEE-279-1971 (Ref. 5). The actual number of channels required for each unit parameter is specified in Reference 2.

Two logic channels are required to ensure no single random failure of a logic channel will disable the RTS. The logic channels are designed such that testing required while the reactor is at power may be accomplished without causing trip. Provisions to allow removing logic channels from service during maintenance are unnecessary because of the logic system's designed reliability.

BASES

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## BACKGROUND (continued)

Allowable Values and Nominal Trip Setpoints

The trip setpoints used in the bistables, setpoint comparators, or contacts are based on the Analytical Limits stated in Reference 4. The calculation of the NTSP specified in Table 3.3.1-1 is such that adequate protection is provided when all sensor and processing time delays are taken into account. To allow for calibration tolerances, instrumentation uncertainties, instrument drift, and severe environment errors for those RTS channels that must function in harsh environments as defined by 10 CFR 50.49 (Ref. 6), the Allowable Values specified in Table 3.3.1-1 in the accompanying LCO are conservative with respect to the Analytical Limits. A detailed description of the methodology used to calculate the Allowable Values and NTSP, including their explicit uncertainties, is provided in the plant specific setpoint methodology study (Ref. 7) which incorporates all of the known uncertainties applicable to each channel. The as-left tolerance and as-found tolerance band methodology is provided in UFSAR, Section 7.1.2. The magnitudes of these uncertainties are factored into the determination of each NTSP and corresponding Allowable Value. The trip setpoint entered into the bistable or setpoint comparator is more conservative than that specified by the Allowable Value to account for measurement errors detectable by the CHANNEL OPERATIONAL TEST (COT). The Allowable Value serves as the as-found Technical Specification OPERABILITY limit for the purpose of the COT.

The NTSP is the value at which the bistable or setpoint comparator is set and is the expected value to be achieved during calibration. The NTSP value is the LSSS and ensures the safety analysis limits are met for the surveillance interval selected when a channel is adjusted based on stated channel uncertainties. Any bistable or setpoint comparator is considered to be properly adjusted when the "as-left" NTSP value is within the as-left tolerance band for CHANNEL CALIBRATION uncertainty allowance (i.e.,  $\pm$  rack calibration and comparator setting uncertainties). The NTSP value is therefore considered a "nominal" value (i.e., expressed as a value without inequalities) for the purposes of COT and CHANNEL CALIBRATION.

NTSPs, in conjunction with the use of as-found and as-left tolerances, together with the requirements of the Allowable Value ensure that SLs are not violated during AOOs (and that the consequences of DBAs will be acceptable, providing the unit is operated from within the LCOs at the onset of the AOO or DBA and the equipment functions as designed).

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### BACKGROUND (continued)

Note that the Allowable Values listed in Table 3.3.1-1 are the least conservative value of the as-found setpoint that a channel can have during a periodic CHANNEL CALIBRATION, COTs, or a TRIP ACTUATING DEVICE OPERATIONAL TEST that requires trip setpoint verification.

Each channel of the process control equipment can be tested online to verify that the signal or setpoint accuracy is within the specified allowance requirements of Reference 3. Once a designated channel is taken out of service for testing, a simulated signal is injected in place of the field instrument signal. The process equipment for the channel in test is then tested, verified, and calibrated. SRs for the channels are specified in the SRs section.

#### Solid State Protection System

The SSPS equipment is used for the decision logic processing of outputs from the signal processing equipment bistables, setpoint comparators, or contacts. To meet the redundancy requirements, two trains of SSPS, each performing the same functions, are provided. If one train is taken out of service for maintenance or test purposes, the second train will provide reactor trip and/or ESF actuation for the unit. If both trains are taken out of service or placed in test, a reactor trip will result. Each train is packaged in its own cabinet for physical and electrical separation to satisfy separation and independence requirements. The system has been designed to trip in the event of a loss of power, directing the unit to a safe shutdown condition.

The SSPS performs the decision logic for actuating a reactor trip or ESF actuation, generates the electrical output signal that will initiate the required trip or actuation, and provides the status, permissive, and annunciator output signals to the main control room of the unit.

The bistable, setpoint comparator, or contact outputs from the signal processing equipment are sensed by the SSPS equipment and combined into logic matrices that represent combinations indicative of various unit upset and accident transients. If a required logic matrix combination is completed, the system will initiate a reactor trip or send actuation signals via master and slave relays to those components whose aggregate Function best serves to alleviate the condition and restore the unit to a safe condition. Examples are given in the Applicable Safety Analyses, LCO, and Applicability sections of this Bases.

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### BACKGROUND (continued)

#### Reactor Trip Switchgear

The reactor trip breakers are in the electrical power supply line from the control rod drive motor generator set power supply to the CRDMs. Opening of the reactor trip breakers interrupts power to the CRDMs, which allows the shutdown rods and control rods to fall into the core by gravity. Each reactor trip breaker is equipped with a bypass breaker to allow testing of the reactor trip breaker while the unit is at power.

During normal operation the output from the SSPS is a voltage signal that energizes the undervoltage coils in the reactor trip breakers and bypass breakers, if in use. When the required logic matrix combination is completed, the SSPS output voltage signal is removed, the undervoltage coils are de-energized, the breaker trip lever is actuated by the de-energized undervoltage coil, and the reactor trip breakers and bypass breakers are tripped open. This allows the shutdown rods and control rods to fall into the core. In addition to the de-energization of the undervoltage coils, each breaker is also equipped with a shunt trip device that is energized to trip the breaker open upon receipt of a reactor trip signal from the SSPS. Either the undervoltage coil or the shunt trip mechanism is sufficient by itself, thus providing a diverse trip mechanism.

The decision logic matrix Functions are described in the functional diagrams included in Reference 2. In addition to the reactor trip or ESF, these diagrams also describe the various "permissive interlocks" that are associated with unit conditions. Each train has a built in testing device that can automatically test the decision logic matrix Functions and the actuation channels while the unit is at power. When any one train is taken out of service for testing, the other train is capable of providing unit monitoring and protection until the testing has been completed. The testing device is semiautomatic to minimize testing time.

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The RTS functions to preserve the SLs during all AOOs and mitigates the consequences of DBAs in all MODES in which the Rod Control System is capable of rod withdrawal or one or more rods are not fully inserted.

Each of the analyzed accidents and transients can be detected by one or more RTS Functions. The accident analysis described in Reference 4 takes credit for most RTS trip Functions. RTS trip Functions that are retained yet not specifically credited in the accident analysis are implicitly credited in the safety analysis and the NRC staff approved licensing basis for the unit. These RTS trip Functions may provide protection for conditions that do not require dynamic transient analysis to demonstrate Function performance. They may also serve as backups to RTS trip Functions that were credited in the accident analysis.

Permissive and interlock setpoints allow the blocking of trips during plant startups, and restoration of trips when the permissive conditions are not satisfied, but they are not explicitly modeled in the Safety Analyses. These permissives and interlocks ensure that the starting conditions are consistent with the safety analysis, before preventive or mitigating actions occur. Because these permissives or interlocks are only one of multiple conservative starting assumptions for the accident analysis, they are generally considered as nominal values without regard to measurement accuracy.

The LCO requires all instrumentation performing an RTS Function, listed in Table 3.3.1-1 to be OPERABLE. The Allowable Value specified in Table 3.3.1-1 is the least conservative value of the as-found setpoint that the channel can have when tested, such that a channel is OPERABLE if the as-found setpoint is within the as-found tolerance and is conservative with respect to the Allowable Value during a CHANNEL CALIBRATION or COT. As such, the Allowable Value differs from the NTSP by an amount greater than or equal to the expected instrument channel uncertainties, such as drift, during the surveillance interval. In this manner, the actual setting of the channel NTSP will ensure that a SL is not exceeded at any given point of time as long as the channel has not drifted beyond expected tolerances during the surveillance interval. Note that, although the channel is OPERABLE under these circumstances, the trip setpoint must be left adjusted to a value within the as-left tolerance, in accordance with uncertainty assumptions stated in the referenced setpoint methodology (as-left criteria), and confirmed to be operating within the statistical allowances of the uncertainty terms assigned (as-found criteria).

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### APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

If the actual setting of the channel is found to be conservative with respect to the Allowable Value but is beyond the as-found tolerance band, the channel is OPERABLE but degraded. The degraded condition of the channel will be further evaluated during performance of the SR. This evaluation will consist of resetting the channel setpoint to the NTSP (within the allowed tolerance), and evaluating the channel's response. If the channel is functioning as required and is expected to pass the next surveillance, then the channel is OPERABLE and can be restored to service at the completion of the surveillance. After the surveillance is completed, the channel as-found condition will be entered into the Corrective Action Program for further evaluation.

A trip setpoint may be set more conservative than the NTSP as necessary in response to plant conditions. However, in this case, the OPERABILITY of this instrument must be verified based on the field setting and not the NTSP. Failure of any instrument renders the affected channel(s) inoperable and reduces the reliability of the affected Functions.

The LCO generally requires OPERABILITY of four or three channels in each instrumentation Function, two channels of Manual Reactor Trip in each logic Function, and two trains in each Automatic Trip Logic Function. Four OPERABLE instrumentation channels in a two-out-of-four configuration are required when one RTS channel is also used as a control system input. This configuration accounts for the possibility of the shared channel failing in such a manner that it creates a transient that requires RTS action. In this case, the RTS will still provide protection, even with random failure of one of the other three protection channels. Three OPERABLE instrumentation channels in a two-out-of-three configuration are generally required when there is no potential for control system and protection system interaction that could simultaneously create a need for RTS trip and disable one RTS channel. The two-out-of-three and two-out-of-four configurations allow one channel to be tripped during maintenance or testing without causing a reactor trip. Specific exceptions to the above general philosophy exist and are discussed below.

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### APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

#### Reactor Trip System Functions

The safety analyses and OPERABILITY requirements applicable to each RTS Function are discussed below:

##### 1. Manual Reactor Trip

The Manual Reactor Trip ensures that the control room operator can initiate a reactor trip at any time by using either of two reactor trip switches in the control room. A Manual Reactor Trip accomplishes the same results as any one of the automatic trip Functions. It is used by the reactor operator to shut down the reactor whenever any parameter is rapidly trending toward its Trip Setpoint.

There are two Manual Reactor Trip channels arranged in a one-out-of-two logic. The LCO requires two Manual Reactor Trip channels to be OPERABLE. Each channel is controlled by a manual reactor trip switch. Each channel activates the reactor trip breaker in both trains. Two independent channels are required to be OPERABLE so that no single random failure will disable the Manual Reactor Trip Function.

In MODE 1 or 2, manual initiation of a reactor trip must be OPERABLE. These are the MODES in which the reactor is critical. In MODE 3, 4, or 5, the manual initiation Function must also be OPERABLE if one or more shutdown rods or control rods are withdrawn or the Rod Control System is capable of withdrawing the shutdown rods or the control rods. In this condition, inadvertent control rod withdrawal is possible. In MODE 3, 4, or 5, manual initiation of a reactor trip does not have to be OPERABLE if the Rod Control System is not capable of withdrawing the shutdown rods or control rods and if all rods are fully inserted. If the rods cannot be withdrawn from the core, or all of the rods are inserted, there is no need to be able to trip the reactor. In MODE 6, neither the shutdown rods nor the control rods are permitted to be withdrawn and the CRDMs are disconnected from the control rods and shutdown rods. Therefore, the manual initiation Function is not required.

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

2. Power Range Neutron Flux

The NIS power range detectors are located external to the reactor vessel and measure neutrons leaking from the core. The NIS power range detectors provide input to the Rod Control System. Therefore, the actuation logic must be able to withstand an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation. Note that this Function also provides a signal to prevent automatic and manual rod withdrawal prior to initiating a reactor trip. Limiting further rod withdrawal may terminate the transient and eliminate the need to trip the reactor.

a. Power Range Neutron Flux - High

The Power Range Neutron Flux - High trip Function ensures that protection is provided, from all power levels, against a positive reactivity excursion leading to DNB during power operations. These can be caused by rod withdrawal or reductions in RCS temperature.

There are four Power Range Neutron Flux – High channels arranged in a two-out-of-four logic. The LCO requires all four of the Power Range Neutron Flux - High channels to be OPERABLE.

In MODE 1 or 2, when a positive reactivity excursion could occur, the Power Range Neutron Flux - High trip must be OPERABLE. This Function will terminate the reactivity excursion and shut down the reactor prior to reaching a power level that could damage the fuel. In MODE 3, 4, 5, or 6, the NIS power range detectors cannot detect neutron levels in this range. In these MODES, the Power Range Neutron Flux - High does not have to be OPERABLE because the reactor is shut down and reactivity excursions into the power range are extremely unlikely. Other RTS Functions and administrative controls provide protection against reactivity additions when in MODE 3, 4, 5, or 6.

b. Power Range Neutron Flux - Low

The LCO requirement for the Power Range Neutron Flux - Low trip Function ensures that protection is provided against a positive reactivity excursion from low power or subcritical conditions.

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### APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

There are four Power Range Neutron Flux – Low channels arranged in a two-out-of-four logic. The LCO requires all four of the Power Range Neutron Flux - Low channels to be OPERABLE.

In MODE 1, below the Power Range Neutron Flux (P-10 setpoint), and in MODE 2, the Power Range Neutron Flux - Low trip must be OPERABLE. This Function may be manually blocked by the operator when two out of four power range channels are greater than approximately 10% RTP (P-10 setpoint). This Function is automatically unblocked when three out of four power range channels are below the P-10 setpoint. Above the P-10 setpoint, positive reactivity additions are mitigated by the Power Range Neutron Flux - High trip Function.

In MODE 3, 4, 5, or 6, the Power Range Neutron Flux - Low trip Function does not have to be OPERABLE because the reactor is shut down and the NIS power range detectors cannot detect neutron levels in this range. Other RTS trip Functions and administrative controls provide protection against positive reactivity additions or power excursions in MODE 3, 4, 5, or 6.

#### 3. Power Range Neutron Flux Rate

The Power Range Neutron Flux Rate trips use the same channels as discussed for Function 2 above.

##### a. Power Range Neutron Flux - High Positive Rate

The Power Range Neutron Flux - High Positive Rate trip Function ensures that protection is provided against rapid increases in neutron flux that are characteristic of an RCCA drive rod housing rupture and the accompanying ejection of the RCCA. This Function compliments the Power Range Neutron Flux - High and Low Setpoint trip Functions to ensure that the criteria are met for a rod ejection from the power range.

There are four Power Range Neutron Flux – High Positive Rate channels arranged in a two-out-of-four logic. The LCO requires all four of the Power Range Neutron Flux - High Positive Rate channels to be OPERABLE.

In MODE 1 or 2, when there is a potential to add a large amount of positive reactivity from a Rod Ejection Accident (REA), the Power Range Neutron Flux - High Positive Rate trip must be OPERABLE. In MODE 3, 4, 5, with Rod Control System capable

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

of rod withdrawal or one or more rods not fully inserted, or in MODE 6, the Power Range Neutron Flux - High Positive Rate trip Function does not have to be OPERABLE because other RTS trip Functions and administrative controls will provide protection against positive reactivity additions. Also, in MODE 3, 4, or 5, with Rod Control System incapable of rod withdrawal and all rods fully inserted, there is a sufficient degree of SDM in the event of an REA. In MODE 6, no rods are withdrawn and the SDM is increased during refueling operations. The reactor vessel head is also removed or the closure bolts are detensioned preventing any pressure buildup. In addition, the NIS power range detectors cannot detect neutron levels present in this mode.

b. Power Range Neutron Flux - High Negative Rate

The Power Range Neutron Flux - High Negative Rate trip Function ensures that protection is provided for multiple rod drop accidents. At high power levels, a multiple rod drop accident could cause local flux peaking that would result in a nonconservative local DNBR. DNBR is defined as the ratio of the heat flux required to cause a DNB at a particular location in the core to the local heat flux. The DNBR is indicative of the margin to DNB. No credit is taken for the operation of this Function for those rod drop accidents in which the local DNBRs will be greater than the limit.

There are four Power Range Neutron Flux – High Negative Rate channels arranged in a two-out-of-four logic. The LCO requires all four Power Range Neutron Flux - High Negative Rate channels to be OPERABLE.

In MODE 1 or 2, when there is potential for a multiple rod drop accident to occur, the Power Range Neutron Flux - High Negative Rate trip must be OPERABLE. In MODE 3, 4, 5, or 6, the Power Range Neutron Flux - High Negative Rate trip Function does not have to be OPERABLE because the core is not critical and DNB is not a concern. In MODE 6, no rods are withdrawn and the required SDM is increased during refueling operations. In addition, the NIS power range detectors cannot detect neutron levels present in this MODE.

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### APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

#### 4. Intermediate Range Neutron Flux

The Intermediate Range Neutron Flux trip Function ensures that protection is provided against an uncontrolled RCCA bank rod withdrawal accident from a subcritical condition during startup. This trip Function provides redundant protection to the Power Range Neutron Flux - Low Setpoint trip Function. The NIS intermediate range detectors are located external to the reactor vessel and measure neutrons leaking from the core. The NIS intermediate range detectors do not provide any input to control systems. Note that this Function also provides a signal to prevent automatic and manual rod withdrawal prior to initiating a reactor trip. Limiting further rod withdrawal may terminate the transient and eliminate the need to trip the reactor.

There are two Intermediate Range Neutron Flux channels arranged in a one-out-of-two logic. The LCO requires two channels of Intermediate Range Neutron Flux to be OPERABLE. Two OPERABLE channels are sufficient to ensure no single random failure will disable this trip Function.

Because this trip Function is important only during startup, there is generally no need to disable channels for testing while the Function is required to be OPERABLE. Therefore, a third channel is unnecessary.

In MODE 1 below the P-10 setpoint, and in MODE 2 above the P-6 setpoint, when there is a potential for an uncontrolled RCCA bank rod withdrawal accident during reactor startup, the Intermediate Range Neutron Flux trip must be OPERABLE. Above the P-10 setpoint, the Power Range Neutron Flux - High Setpoint trip and the Power Range Neutron Flux - High Positive Rate trip provide core protection for a rod withdrawal accident. In MODE 2 below the P-6 setpoint, the Source Range Neutron Flux Trip provides the core protection for reactivity accidents. In MODE 3, 4, or 5, the Intermediate Range Neutron Flux trip does not have to be OPERABLE because the Rod Control System is not capable of rod withdrawal or the Source Range Neutron Flux function is required to be OPERABLE, providing protection. In MODE 6, all rods are fully inserted and the core has a required increased SDM. Also, the NIS intermediate range detectors cannot detect neutron levels present in this MODE.

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### APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

#### 5. Source Range Neutron Flux

The LCO requirement for the Source Range Neutron Flux trip Function ensures that protection is provided against an uncontrolled RCCA bank rod withdrawal accident from a subcritical condition during startup. This trip Function provides redundant protection to the Power Range Neutron Flux - Low trip Function. In MODES 3, 4, and 5, administrative controls also prevent the uncontrolled withdrawal of rods. The NIS source range detectors are located external to the reactor vessel and measure neutrons leaking from the core. The NIS source range detectors do not provide any inputs to control systems. The source range trip is the only RTS automatic protection function required in MODES 3, 4, and 5 when rods are capable of withdrawal or one or more rods are not fully inserted. Therefore, the functional capability at the specified Trip Setpoint is assumed to be available.

The Source Range Neutron Flux Function provides protection for control rod withdrawal from subcritical and boron dilution events.

There are two Source Range Neutron Flux channels arranged in a one-out-of-two logic. In MODE 2 when below the P-6 setpoint and in MODES 3, 4, and 5 when there is a potential for an uncontrolled RCCA bank rod withdrawal accident, the Source Range Neutron Flux trip must be OPERABLE. Two OPERABLE channels are sufficient to ensure no single random failure will disable this trip Function. Above the P-6 setpoint, the Intermediate Range Neutron Flux trip and the Power Range Neutron Flux - Low trip will provide core protection for reactivity accidents. Above the P-6 setpoint, the NIS source range drawer input is shorted out, driving the output of the drawer to zero.

In MODES 3, 4, and 5 with all rods fully inserted and the Rod Control System not capable of rod withdrawal, and in MODE 6, the outputs of the Function to RTS logic are not required OPERABLE. The requirements for the NIS source range detectors to monitor core neutron levels and provide indication of reactivity changes that may occur as a result of events like a boron dilution are addressed in LCO 3.3.9 "Boron Dilution Monitoring Instrumentation (BDMI)," for MODE 3, 4, or 5 and LCO 3.9.3, "Nuclear Instrumentation," for MODE 6.

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## APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

6. Overtemperature  $\Delta T$ 

The Overtemperature  $\Delta T$  trip Function is provided to ensure that the design limit DNBR is met. This trip Function also limits the range over which the Overpower  $\Delta T$  trip Function must provide protection. The inputs to the Overtemperature  $\Delta T$  trip include pressurizer pressure, coolant temperature, axial power distribution, and reactor power as indicated by loop  $\Delta T$  assuming full reactor coolant flow. Protection from violating the DNBR limit is assured for those transients that are slow with respect to delays from the core to the measurement system. The Function monitors both variation in power and flow since a decrease in flow has the same effect on  $\Delta T$  as a power increase. The Overtemperature  $\Delta T$  trip Function uses each loop's  $\Delta T$  as a measure of reactor power and is compared with a setpoint that is automatically varied with the following parameters:

- reactor coolant average temperature - the Trip Setpoint is varied to correct for changes in coolant density and specific heat capacity with changes in coolant temperature,
- pressurizer pressure - the Trip Setpoint is varied to correct for changes in system pressure, and
- axial power distribution -  $f(\Delta I)$ , the Trip Setpoint is varied to account for imbalances in the axial power distribution as detected by the NIS upper and lower power range detectors. If axial peaks are greater than the design limit, as indicated by the difference between the upper and lower NIS power range detectors, the Trip Setpoint is reduced in accordance with Note 1 of Table 3.3.1-1.

Dynamic compensation is included for system piping delays from the core to the temperature measurement system.

The Overtemperature  $\Delta T$  trip Function is calculated for each loop as described in Note 1 of Table 3.3.1-1. Trip occurs if Overtemperature  $\Delta T$  is indicated in two loops. The pressure and temperature signals are used for other control functions. Therefore, the actuation logic must be able to withstand an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation. Note that this Function also provides a signal to generate a turbine runback prior to reaching the Trip Setpoint. A turbine runback will reduce turbine power and reactor power. A reduction in power will normally alleviate the Overtemperature  $\Delta T$  condition and may prevent a reactor trip.

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

There are four Overtemperature  $\Delta T$  channels arranged in a two-out-of-four logic. The LCO requires all four channels of the Overtemperature  $\Delta T$  trip Function to be OPERABLE. Note that the Overtemperature  $\Delta T$  Function receives input from channels shared with other RTS Functions. Failures that affect multiple Functions require entry into the Conditions applicable to all affected Functions.

In MODE 1 or 2, the Overtemperature  $\Delta T$  trip must be OPERABLE to prevent DNB. In MODE 3, 4, 5, or 6, this trip Function does not have to be OPERABLE because the reactor is not operating and there is insufficient heat production to be concerned about DNB.

7. Overpower  $\Delta T$

The Overpower  $\Delta T$  trip Function ensures that protection is provided to ensure the integrity of the fuel (i.e., no fuel pellet melting and less than 1% cladding strain) under all possible overpower conditions. This trip Function also limits the required range of the Overtemperature  $\Delta T$  trip Function and provides a backup to the Power Range Neutron Flux - High Setpoint trip. The Overpower  $\Delta T$  trip Function ensures that the allowable heat generation rate (kW/ft) of the fuel is not exceeded. It uses the  $\Delta T$  of each loop as a measure of reactor power with a setpoint that is automatically varied with the following parameters:

- reactor coolant average temperature - the Trip Setpoint is varied to correct for changes in coolant density and specific heat capacity with changes in coolant temperature,
- rate of change of reactor coolant average temperature - including dynamic compensation for the delays between the core and the temperature measurement system, and
- axial power distribution -  $f(\Delta I)$ , the Trip Setpoint is varied to account for imbalances in the axial power distribution as detected by the NIS upper and lower power range detectors. If axial peaks are greater than the design limit, as indicated by the difference between the upper and lower NIS power range detectors, the Trip Setpoint is reduced in accordance with Note 2 of Table 3.3.1-1.

The Overpower  $\Delta T$  trip Function is calculated for each loop as per Note 2 of Table 3.3.1-1. Trip occurs if Overpower  $\Delta T$  is indicated in two loops. The temperature signals are used for other control functions. Therefore, the actuation logic must be able to withstand an input failure to the control system, which may then require the

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### APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

protection function actuation and a single failure in the remaining channels providing the protection function actuation. Note that this Function also provides a signal to generate a turbine runback prior to reaching the Allowable Value. A turbine runback will reduce turbine power and reactor power. A reduction in power will normally alleviate the Overpower  $\Delta T$  condition and may prevent a reactor trip.

There are four Overpower  $\Delta T$  channels arranged in a two-out-of-four logic. The LCO requires four channels of the Overpower  $\Delta T$  trip Function to be OPERABLE. Note that the Overpower  $\Delta T$  trip Function receives input from channels shared with other RTS Functions. Failures that affect multiple Functions require entry into the Conditions applicable to all affected Functions.

In MODE 1 or 2, the Overpower  $\Delta T$  trip Function must be OPERABLE. These are the only times that enough heat is generated in the fuel to be concerned about the heat generation rates and overheating of the fuel. In MODE 3, 4, 5, or 6, this trip Function does not have to be OPERABLE because the reactor is not operating and there is insufficient heat production to be concerned about fuel overheating and fuel damage.

#### 8. Pressurizer Pressure

The same sensors provide input to the Pressurizer Pressure - High and - Low trips and the Overtemperature  $\Delta T$  trip. The Pressurizer Pressure channels are also used to provide input to the Pressurizer Pressure Control System. The actuation logic must be able to withstand an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation.

##### a. Pressurizer Pressure - Low

The Pressurizer Pressure - Low trip Function ensures that protection is provided against violating the DNBR limit due to low pressure.

There are four Pressurizer Pressure - Low channels arranged in a two-out-of-four logic. The LCO requires four channels of Pressurizer Pressure - Low to be OPERABLE.

In MODE 1, when DNB is a major concern, the Pressurizer Pressure - Low trip must be OPERABLE. This trip Function is automatically enabled on increasing power by the P-7 interlock (NIS power range P-10 or turbine impulse pressure greater than

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approximately 10% of full power equivalent (P-13)). On decreasing power, this trip Function is automatically blocked below P-7.

b. Pressurizer Pressure - High

The Pressurizer Pressure - High trip Function ensures that protection is provided against overpressurizing the RCS. This trip Function operates in conjunction with the pressurizer relief and safety valves to prevent RCS overpressure conditions.

There are four Pressurizer Pressure - High channels arranged in a two-out-of-four logic. The LCO requires four channels of the Pressurizer Pressure - High to be OPERABLE.

The Pressurizer Pressure - High LSSS is selected to be below the pressurizer safety valve actuation pressure and above the Power Operated Relief Valve (PORV) setting. This setting minimizes challenges to safety valves while avoiding unnecessary reactor trip for those pressure increases that can be controlled by the PORVs.

In MODE 1 or 2, the Pressurizer Pressure - High trip must be OPERABLE to help prevent RCS overpressurization and minimize challenges to the relief and safety valves. In MODE 3, 4, 5, or 6, the Pressurizer Pressure - High trip Function does not have to be OPERABLE because transients that could cause an overpressure condition will be slow to occur. Therefore, the operator will have sufficient time to evaluate unit conditions and take corrective actions. Additionally, low temperature overpressure protection systems provide overpressure protection when below MODE 4.

9. Pressurizer Water Level - High

The Pressurizer Water Level - High trip Function provides a backup signal for the Pressurizer Pressure - High trip and also provides protection against water relief through the pressurizer safety valves. These valves are designed to pass steam in order to achieve their design energy removal rate. A reactor trip is actuated prior to the pressurizer becoming water solid. The LCO requires three channels of Pressurizer Water Level - High to be OPERABLE. The pressurizer level channels are used as input to the Pressurizer Level Control System. A fourth channel is not required to address control/protection interaction concerns. The level channels do not

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### APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

actuate the safety valves, and the high pressure reactor trip is set below the safety valve setting. Therefore, with the slow rate of charging available, pressure overshoot due to level channel failure cannot cause the safety valve to lift before reactor high pressure trip.

There are three Pressurizer Level - High channels arranged in a two-out-of-three logic. In MODE 1, when there is a potential for overfilling the pressurizer, the Pressurizer Water Level - High trip must be OPERABLE. This trip Function is automatically enabled on increasing power by the P-7 interlock. On decreasing power, this trip Function is automatically blocked below P-7.

#### 10. Reactor Coolant Flow - Low

The Reactor Coolant Flow - Low trip Function ensures that protection is provided against violating the DNBR limit due to low flow in one or more RCS loops, while avoiding reactor trips due to normal variations in loop flow. Above the P-7 setpoint, the reactor trip on low flow in two or more RCS loops is automatically enabled. Above the P-8 setpoint, which is approximately 35% RTP, a loss of flow in any RCS loop will actuate a reactor trip. Each RCS loop has three flow detectors to monitor flow. There are three per loop Reactor Coolant Flow - Low channels using these detectors and are arranged in a two-out-of-three logic for each loop. The flow signals are not used for any control system input.

Design flow is 94,600 (91,400 X 1.035) gpm per loop (Reference 14). UFSAR Table 5.1-1 lists this value as the Full Power Operability Flow, gpm/loop.

The LCO requires three Reactor Coolant Flow - Low channels per loop to be OPERABLE in MODE 1 above P-7.

In MODE 1 above the P-8 setpoint, a loss of flow in one RCS loop could result in DNB conditions in the core because of the higher power level. In MODE 1 below the P-8 setpoint and above the P-7 setpoint, a loss of flow in two or more loops is required to actuate a reactor trip because of the lower power level and the greater margin to the design limit DNBR. Below the P-7 setpoint, all reactor trips on low flow are automatically blocked.

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

11. Undervoltage Reactor Coolant Pumps (RCPs)

The Undervoltage RCPs reactor trip Function ensures that protection is provided against violating the DNBR limit due to a loss of flow in two or more RCS loops. The voltage to each RCP is monitored. Above the P-7 setpoint, a loss of voltage detected on two or more RCP buses will initiate a reactor trip. This trip Function will generate a reactor trip before the Reactor Coolant Flow - Low Trip Setpoint is reached. Time delays are incorporated into the Undervoltage RCPs channels to prevent reactor trips due to momentary electrical power transients.

There are four (one per bus) Undervoltage RCP channels arranged in a two-out-of-four logic. The LCO requires four Undervoltage RCPs channels (one per bus) to be OPERABLE.

In MODE 1 above the P-7 setpoint, the Undervoltage RCP trip must be OPERABLE. Below the P-7 setpoint, all reactor trips on loss of flow are automatically blocked since no conceivable power distributions could occur that would cause a DNB concern at this low power level. Above the P-7 setpoint, the reactor trip on loss of flow in two or more RCS loops is automatically enabled.

12. Underfrequency Reactor Coolant Pumps

The Underfrequency RCPs reactor trip Function ensures that protection is provided against violating the DNBR limit due to a loss of flow in two or more RCS loops from a major network frequency disturbance. An underfrequency condition will slow down the pumps, thereby reducing their coastdown time following a pump trip. The proper coastdown time is required so that reactor heat can be removed immediately after reactor trip. The frequency of each RCP bus is monitored. Above the P-7 setpoint, a loss of frequency detected on two or more RCP buses will initiate a reactor trip. This trip Function will generate a reactor trip before the Reactor Coolant Flow - Low Trip Setpoint is reached. Time delays are incorporated into the Underfrequency RCPs channels to prevent reactor trips due to momentary electrical power transients.

There are four (one per bus) Underfrequency RCP channels arranged in a two-out-of-four logic. The LCO requires four Underfrequency RCPs channels (one per bus) to be OPERABLE.

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### APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

In MODE 1 above the P-7 setpoint, the Underfrequency RCPs trip must be OPERABLE. Below the P-7 setpoint, all reactor trips on loss of flow are automatically blocked since no conceivable power distributions could occur that would cause a DNB concern at this low power level. Above the P-7 setpoint, the reactor trip on loss of flow in two or more RCS loops is automatically enabled. Note that this Function also provides a signal to trip all four reactor coolant pumps.

#### 13. Steam Generator Water Level - Low Low

The Steam Generator Water Level Low-Low trip protects the reactor from loss of heat sink in the event of a sustained steam/feedwater flow mismatch resulting from loss of normal feedwater or a feedwater system pipe break outside of containment. This function also provides input to the steam generator level control system. IEEE-279 requirements are satisfied by 2/3 logic for protection function actuation, thus allowing for a single failure of a channel and still performing the protection function.

Control/protection interaction is addressed by the use of the Median Signal Selector that prevents a single failure of a channel providing input to the control system requiring protection function action. That is, a single failure of a channel providing input to the control system does not result in the control system initiating a condition requiring protection function action. The Median Signal Selector performs this by not selecting the channels indicating the highest or lowest steam generator levels as input to the control system.

With the transmitters located inside containment and thus possibly experiencing adverse environmental conditions (due to a feedline break), the Environmental Allowance Modifier (EAM) was devised. The EAM function (Containment Pressure (EAM) with a setpoint of < 0.5 psig) senses the presence of adverse containment conditions (elevated pressure) and enables the Steam Generator Water Level - Low-Low trip setpoint (Adverse) which reflects the increased transmitter uncertainties due to this environment. The EAM allows the use of a lower Steam Generator Water Level - Low-Low (EAM) trip setpoint when these conditions are not present, thus allowing more margin to trip for normal operating conditions.

The Trip Time Delay (TTD) creates additional operational margin when the plant needs it most, during early escalation to power, by allowing the operator time to recover level when the primary side load is sufficiently small to allow such action. The TTD is based on continuous monitoring of primary side power through the use of RCS loop  $\Delta T$ .

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Two time delays are calculated, based on the number of steam generators indicating less than the Low-Low Level trip setpoint and the primary side power level. The magnitude of the delays decreases with increasing primary side power level, up to 50% RTP. Above 50% RTP there are no time delays for the Low-Low level trips.

In the event of failure of a Steam Generator Water Level channel, it is placed in the trip condition as input to the Solid State Protection System and does not affect either the EAM or TTD setpoint calculations for the remaining operable channels. Failure of the Containment Pressure (EAM) channel to a protection set also does not affect the EAM setpoint calculations. It is then necessary for the operator to force the use of the shorter TTD by adjustment of the single steam generator time delay calculation ( $T_S$ ) to match the multiple steam generator time delay calculation ( $T_M$ ) for the affected protection set, through the Eagle-21 System Man-Machine-Interface (MMI) test cart. Failure of the RCS loop  $\Delta T$  channel input (failure of more than one  $T_H$  RTD or failure of a  $T_C$  RTD) does not affect the TTD calculation for a protection set. Although not affecting the TTD calculation, this results in the requirement that the operator adjust the threshold power level for zero seconds time delay from 50% RTP to 0% RTP, through the MMI, or place the affected protection sets Steam Generator Water Level - Low-Low channel in trip.

This Function also performs the ESFAS function of starting the Auxiliary Feedwater (AFW) pumps on low low SG level.

There are three Steam Generator Water Level Low-Low channels per steam generator arranged in a two-out-of-three logic. These channels are arranged in four protection sets with each channel of the Containment Pressure (EAM) and RCS Loop  $\Delta T$  inputting into its associated protection set. The LCO requires three channels of SG Water Level - Low Low per SG to be OPERABLE.

In MODE 1 or 2, when the reactor requires a heat sink, the SG Water Level - Low Low trip must be OPERABLE. The normal source of water for the SGs is the Main Feedwater (MFW) System (not safety related). The MFW System is only in operation in MODE 1 or 2. The AFW System is the safety related backup source of water to ensure that the SGs remain the heat sink for the reactor. During normal startups and shutdowns, the AFW System provides feedwater to maintain SG level. In MODE 3, 4, 5, or 6, the SG Water Level - Low Low Function does not have to be OPERABLE because the MFW System is not in operation and the reactor is not operating or even critical. Decay heat removal is accomplished by the AFW System

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

in MODE 3 and by the Residual Heat Removal (RHR) System in MODE 4, 5, or 6.

14. Turbine Trip

a. Turbine Trip - Low Fluid Oil Pressure

The Turbine Trip - Low Fluid Oil Pressure trip Function anticipates the loss of heat removal capabilities of the secondary system following a turbine trip. This trip Function acts to minimize the pressure/temperature transient on the reactor. Any turbine trip from a power level below the P-9 setpoint, approximately 50% power, will not actuate a reactor trip. Three pressure switches monitor the auto stop oil pressure in the Turbine Electrohydraulic Control System. A low pressure condition sensed by two-out-of-three pressure switches will actuate a reactor trip. These pressure switches do not provide any input to the control system. The unit is designed to withstand a complete loss of load and not sustain core damage or challenge the RCS pressure limitations. Core protection is provided by the Pressurizer Pressure - High trip Function and RCS integrity is ensured by the pressurizer safety valves.

The LCO requires three channels of Turbine Trip - Low Fluid Oil Pressure to be OPERABLE in MODE 1 above P-9.

Below the P-9 setpoint, a turbine trip does not actuate a reactor trip. In MODE 2, 3, 4, 5, or 6, there is no potential for a turbine trip, and the Turbine Trip - Low Fluid Oil Pressure trip Function does not need to be OPERABLE.

b. Turbine Trip - Turbine Stop Valve Closure

The Turbine Trip - Turbine Stop Valve Closure trip Function anticipates the loss of heat removal capabilities of the secondary system following a turbine trip from a power level above the P-9 setpoint, approximately 50% power. This action will actuate a reactor trip. The trip Function anticipates the loss of secondary heat removal capability that occurs when the stop valves close. Tripping the reactor in anticipation of loss of secondary heat removal acts to minimize the pressure and temperature transient on the reactor. This trip Function will not and is not required to operate in the presence of a single channel failure. The unit is designed to withstand a complete loss of load and not sustain core damage or challenge the RCS pressure limitations.

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### APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Core protection is provided by the Pressurizer Pressure - High trip Function, and RCS integrity is ensured by the pressurizer safety valves. This trip Function is diverse to the Turbine Trip - Low Fluid Oil Pressure trip Function. Each turbine stop valve is equipped with one limit switch that inputs to the RTS. If all four limit switches indicate that the stop valves are all closed, a reactor trip is initiated.

The LSSS for this Function is set to assure channel trip occurs when the associated stop valve is completely closed.

The LCO requires four Turbine Trip - Turbine Stop Valve Closure channels, one per valve, to be OPERABLE in MODE 1 above P-9. All four channels must trip to cause reactor trip.

Below the P-9 setpoint, a load rejection can be accommodated by the Steam Dump System. In MODE 2, 3, 4, 5, or 6, there is no potential for a load rejection, and the Turbine Trip - Stop Valve Closure trip Function does not need to be OPERABLE.

#### 15. Safety Injection Input from Engineered Safety Feature Actuation System

The SI Input from ESFAS ensures that if a reactor trip has not already been generated by the RTS, the ESFAS automatic actuation logic will initiate a reactor trip upon any signal that initiates SI. This is not a condition of acceptability for the LOCA. However, other transients and accidents take credit for varying levels of ESF performance and rely upon rod insertion, except for the most reactive rod that is assumed to be fully withdrawn, to ensure reactor shutdown. Therefore, a reactor trip is initiated every time an SI signal is present.

Trip Setpoint and Allowable Values are not applicable to this Function. The SI Input is provided by solid state logic in the ESFAS. Therefore, there is no measurement signal with which to associate an LSSS.

There are two trains of SI input from ESFAS arranged in a one-out-of-two logic. The LCO requires two trains of SI Input from ESFAS to be OPERABLE in MODE 1 or 2.

A reactor trip is initiated every time an SI signal is present. Therefore, this trip Function must be OPERABLE in MODE 1 or 2, when the reactor is critical, and must be shut down in the event of an accident. In MODE 3, 4, 5, or 6, the reactor is not critical,

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### APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

and this trip Function does not need to be OPERABLE.

#### 16. Reactor Trip System Interlocks

Reactor protection interlocks are provided to ensure reactor trips are in the correct configuration for the current unit status. They back up operator actions to ensure protection system Functions are not bypassed during unit conditions under which the safety analysis assumes the Functions are not bypassed. Therefore, the interlock Functions do not need to be OPERABLE when the associated reactor trip functions are outside the applicable MODES. These are:

##### a. Intermediate Range Neutron Flux, P-6

The Intermediate Range Neutron Flux, P-6 interlock is actuated when any NIS intermediate range channel goes approximately four decades above the minimum channel reading. If both channels drop below the setpoint, the permissive will automatically be defeated. The LCO requirement for the P-6 interlock ensures that the following Functions are performed:

- on increasing power, the P-6 interlock allows the manual block of the NIS Source Range, Neutron Flux reactor trip. This prevents a premature block of the source range trip and allows the operator to ensure that the intermediate range is OPERABLE prior to leaving the source range. When the source range trip is blocked, the input to the SR drawer is shorted out driving the output of drawer to zero, and
- on decreasing power, the P-6 interlock automatically energizes the NIS source range detectors and enables the NIS Source Range Neutron Flux reactor trip.

There are two Intermediate Range Neutron Flux, P-6 channels arranged in a one-out-of-two logic. The LCO requires two channels of Intermediate Range Neutron Flux, P-6 interlock to be OPERABLE in MODE 2 when below the P-6 interlock setpoint.

Above the P-6 interlock setpoint, the NIS Source Range Neutron Flux reactor trip will be blocked, and this Function will no longer be necessary.

In MODE 3, 4, 5, or 6, the P-6 interlock does not have to be OPERABLE because the NIS Source Range is providing core protection.

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

b. Low Power Reactor Trips Block, P-7

The Low Power Reactor Trips Block, P-7 interlock is actuated by input from either the Power Range Neutron Flux, P-10, or the Turbine Impulse Pressure, P-13 interlock. The LCO requirement for the P-7 interlock ensures that the following Functions are performed:

(1) on increasing power, the P-7 interlock automatically enables reactor trips on the following Functions:

- Pressurizer Pressure - Low,
- Pressurizer Water Level - High,
- Reactor Coolant Flow - Low (low flow in two or more RCS loops),
- Undervoltage RCPs, and
- Underfrequency RCPs.

These reactor trips are only required when operating above the P-7 setpoint (approximately 10% power). The reactor trips provide protection against violating the DNBR limit. Below the P-7 setpoint, the RCS is capable of providing sufficient natural circulation without any RCP running.

(2) on decreasing power, the P-7 interlock automatically blocks reactor trips on the following Functions:

- Pressurizer Pressure - Low,
- Pressurizer Water Level - High,
- Reactor Coolant Flow - Low (low flow in two or more RCS loops),
- Undervoltage RCPs, and
- Underfrequency RCPs.

Trip Setpoint and Allowable Value are not applicable to the P-7 interlock because it is a logic Function and thus has no parameter with which to associate an LSSS.

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### APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The P-7 interlock is a logic Function with train and not channel identity. Therefore, the LCO requires one channel per train of Low Power Reactor Trips Block, P-7 interlock to be OPERABLE in MODE 1.

The low power trips are blocked below the P-7 setpoint and unblocked above the P-7 setpoint. In MODE 2, 3, 4, 5, or 6, this Function does not have to be OPERABLE because the interlock performs its Function when power level drops below 10% power, which is in MODE 1.

c. Power Range Neutron Flux, P-8

The Power Range Neutron Flux, P-8 interlock is actuated at approximately 35% power as determined by two-out-of-four NIS power range detectors. The P-8 interlock automatically enables the Reactor Coolant Flow - Low reactor trip on low flow in one or more RCS loops on increasing power. The LCO requirement for this trip Function ensures that protection is provided against a loss of flow in any RCS loop that could result in DNB conditions in the core when greater than approximately 35% power. On decreasing power, the reactor trip on low flow in any loop is automatically blocked.

The LCO requires four channels of Power Range Neutron Flux, P-8 interlock to be OPERABLE in MODE 1.

In MODE 1, a loss of flow in one RCS loop could result in DNB conditions, so the Power Range Neutron Flux, P-8 interlock must be OPERABLE. In MODE 2, 3, 4, 5, or 6, this Function does not have to be OPERABLE because the core is not producing sufficient power to be concerned about DNB conditions.

d. Power Range Neutron Flux, P-9

The Power Range Neutron Flux, P-9 interlock is actuated at approximately 50% power as determined by two-out-of-four NIS power range detectors. The LCO requirement for this Function ensures that the Turbine Trip - Low Fluid Oil Pressure and Turbine Trip - Turbine Stop Valve Closure reactor trips are enabled above the P-9 setpoint. Above the P-9 setpoint, a turbine trip will cause a load rejection beyond the capacity of the Steam Dump System. A reactor trip is automatically initiated on a turbine trip when it is above the P-9 setpoint, to minimize the transient on the reactor.

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### APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The LCO requires four channels of Power Range Neutron Flux, P-9 interlock to be OPERABLE in MODE 1.

In MODE 1, a turbine trip could cause a load rejection beyond the capacity of the Steam Dump System, so the Power Range Neutron Flux interlock must be OPERABLE. In MODE 2, 3, 4, 5, or 6, this Function does not have to be OPERABLE because the reactor is not at a power level sufficient to have a load rejection beyond the capacity of the Steam Dump System.

e. Power Range Neutron Flux, P-10

The Power Range Neutron Flux, P-10 interlock is actuated at approximately 10% power, as determined by two-out-of-four NIS power range detectors. If power level falls below 10% RTP on 3 of 4 channels, the nuclear instrument trips will be automatically unblocked. The LCO requirement for the P-10 interlock ensures that the following Functions are performed:

- on increasing power, the P-10 interlock allows the operator to manually block the Intermediate Range Neutron Flux reactor trip. Note that blocking the reactor trip also blocks the signal to prevent automatic and manual rod withdrawal,
- on increasing power, the P-10 interlock allows the operator to manually block the Power Range Neutron Flux - Low reactor trip,
- on increasing power, the P-10 interlock automatically provides a backup signal to block the Source Range Neutron Flux reactor trip, and also shorts out the input to the SR drawer, driving the output of drawer to zero,
- the P-10 interlock provides one of the two inputs to the P-7 interlock, and
- on decreasing power, the P-10 interlock automatically enables the Power Range Neutron Flux - Low reactor trip and the Intermediate Range Neutron Flux reactor trip (and rod stop).

The LCO requires four channels of Power Range Neutron Flux, P-10 interlock to be OPERABLE in MODE 1 or 2.

## BASES

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### APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

OPERABILITY in MODE 1 ensures the Function is available to perform its decreasing power Functions in the event of a reactor shutdown. This Function must be OPERABLE in MODE 2 to ensure that core protection is provided during a startup or shutdown by the Power Range Neutron Flux - Low and Intermediate Range Neutron Flux reactor trips. In MODE 3, 4, 5, or 6, this Function does not have to be OPERABLE because the reactor is not at power and the Source Range Neutron Flux reactor trip provides core protection.

f. Turbine Impulse Pressure, P-13

The Turbine Impulse Pressure, P-13 interlock is actuated when the pressure in the first stage of the high pressure turbine is greater than approximately 10% of the rated full power pressure. This is determined by one-out-of-two pressure detectors. The LCO requirement for this Function ensures that one of the inputs to the P-7 interlock is available.

The LCO requires two channels of Turbine Impulse Pressure, P-13 interlock to be OPERABLE in MODE 1.

The Turbine Impulse Chamber Pressure, P-13 interlock must be OPERABLE when the turbine generator is operating. The interlock Function is not required OPERABLE in MODE 2, 3, 4, 5, or 6 because the turbine generator is not operating.

17. Reactor Trip Breakers

This trip Function applies to the reactor trip breakers exclusive of individual trip mechanisms. There are two Reactor Trip Breakers arranged in a one-out-of-two logic. The LCO requires two OPERABLE trains of trip breakers. A trip breaker train consists of all trip breakers associated with a single RTS logic train that are racked in, closed, and capable of supplying power to the Rod Control System. Thus, the train may consist of the main breaker, bypass breaker, or main breaker and bypass breaker, depending upon the system configuration. Two OPERABLE trains ensure no single random failure can disable the RTS trip capability.

These trip Functions must be OPERABLE in MODE 1 or 2 when the reactor is critical. In MODE 3, 4, or 5, these RTS trip Functions must be OPERABLE when the Rod Control System is capable of rod withdrawal or one or more rods are not fully inserted.

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

18. Reactor Trip Breaker Undervoltage and Shunt Trip Mechanisms

The LCO requires both the Undervoltage and Shunt Trip Mechanisms to be OPERABLE for each reactor trip breaker that is in service. The trip mechanisms are not required to be OPERABLE for trip breakers that are open, racked out, incapable of supplying power to the Rod Control System, or declared inoperable under Function 17 above. OPERABILITY of both trip mechanisms on each breaker ensures that no single trip mechanism failure will prevent opening any breaker on a valid signal.

These trip Functions must be OPERABLE in MODE 1 or 2 when the reactor is critical. In MODE 3, 4, or 5, these RTS trip Functions must be OPERABLE when the Rod Control System is capable of rod withdrawal or one or more rods are not fully inserted.

19. Automatic Trip Logic

The LCO requirement for the reactor trip breakers (Functions 17 and 18) and Automatic Trip Logic (Function 19) ensures that means are provided to interrupt the power to allow the rods to fall into the reactor core. Each reactor trip breaker is equipped with an undervoltage coil and a shunt trip coil to trip the breaker open when needed. Each reactor trip breaker is equipped with a bypass breaker to allow testing of the trip breaker while the unit is at power. The reactor trip signals generated by the RTS Automatic Trip Logic cause the reactor trip breakers and associated bypass breakers to open and shut down the reactor.

There are two RTS Automatic Trip Logic trains arranged in a one-out-of-two logic. The LCO requires two trains of RTS Automatic Trip Logic to be OPERABLE. Having two OPERABLE channels ensures that random failure of a single logic channel will not prevent reactor trip.

These trip Functions must be OPERABLE in MODE 1 or 2 when the reactor is critical. In MODE 3, 4, or 5, these RTS trip Functions must be OPERABLE when the Rod Control System is capable of rod withdrawal or one or more rods are not fully inserted.

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

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### ACTIONS

A Note has been added to the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed in Table 3.3.1-1.

In the event a channel's trip setting is found nonconservative with respect to the Allowable Value, or the channel is not functioning as required, or the transmitter, instrument loop, signal processing electronics, setpoint comparator trip output, contact output, or bistable is found inoperable, then all affected Functions provided by that channel must be declared inoperable and the LCO Condition(s) entered for the protection Function(s) affected. When the Required Channels in Table 3.3.1-1 are specified on a "per" basis (e.g., on a per steam line loop, per SG, etc., basis), then the Condition may be entered separately for each steam line, loop, SG, etc., as appropriate.

When the number of inoperable channels in a trip Function exceed those specified in one or other related Conditions associated with a trip Function, then the unit is outside the safety analysis. Therefore, LCO 3.0.3 must be immediately entered if applicable in the current MODE of operation.

#### A.1

Condition A applies to all RTS protection Functions. Condition A addresses the situation where one or more required channels or trains for one or more Functions are inoperable at the same time. The Required Action is to refer to Table 3.3.1-1 and to take the Required Actions for the protection functions affected. The Completion Times are those from the referenced Conditions and Required Actions.

#### B.1 and B.2

Condition B applies to the Manual Reactor Trip in MODE 1 or 2. This action addresses the train orientation of the SSPS for this Function. With one channel inoperable, the inoperable channel must be restored to OPERABLE status within 48 hours. In this Condition, the remaining OPERABLE channel is adequate to perform the safety function.

The Completion Time of 48 hours is reasonable considering that there are two automatic actuation trains and another manual initiation channel OPERABLE, and the low probability of an event occurring during this interval.

If the Manual Reactor Trip Function cannot be restored to OPERABLE status within the allowed 48 hour Completion Time, the unit must be brought to a MODE in which the requirement does not apply. To achieve

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### ACTIONS (continued)

this status, the unit must be brought to at least MODE 3 within 6 additional hours (54 hours total time). The 6 additional hours to reach MODE 3 is reasonable, based on operating experience, to reach MODE 3 from full power operation in an orderly manner and without challenging unit systems. With the unit in MODE 3, ACTION C would apply to any inoperable Manual Reactor Trip Function if the Rod Control System is capable of rod withdrawal or one or more rods are not fully inserted.

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

Condition C applies to the following reactor trip Functions in MODE 3, 4, or 5 with the Rod Control System capable of rod withdrawal or one or more rods not fully inserted:

- Manual Reactor Trip,
- reactor trip breakers,
- reactor trip breaker Undervoltage and Shunt Trip Mechanisms, and
- Automatic Trip Logic.

This action addresses the train orientation of the SSPS for these Functions. With one channel or train inoperable, the inoperable channel or train must be restored to OPERABLE status within 48 hours. If the affected Function(s) cannot be restored to OPERABLE status within the allowed 48 hour Completion Time, the unit must be placed in a MODE in which the requirement does not apply. To achieve this status, action must be initiated within the same 48 hours to ensure that all rods are fully inserted, and the Rod Control System must be placed in a condition incapable of rod withdrawal within the next hour. The additional hour provides sufficient time to accomplish the action in an orderly manner. With rods fully inserted and the Rod Control System incapable of rod withdrawal, these Functions are no longer required.

The Completion Time is reasonable considering that in this Condition, the remaining OPERABLE train is adequate to perform the safety function, and given the low probability of an event occurring during this interval.

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### ACTIONS (continued)

#### D.1 and D.2

Condition D applies to the Power Range Neutron Flux - High Function.

The NIS power range detectors provide input to the Rod Control System and therefore, have a two-out-of-four trip logic. A known inoperable channel must be placed in the tripped condition. This results in a partial trip condition requiring only one-out-of-three logic for actuation. The 72 hours allowed by Required Action D.1 to place the inoperable channel in the tripped condition is justified in WCAP-14333-P-A (Ref. 8).

If Required Action D.1 cannot be met within the specified Completion Time, the plant must be placed in a MODE where this Function is no longer required OPERABLE. Seventy-eight hours are allowed to place the plant in MODE 3. The 78 hour Completion Time includes 6 hours for the MODE reduction as required by Required Action D.2. This is a reasonable time, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging plant systems.

The Required Actions have been modified by two Notes. Note 1 allows the inoperable channel to be placed in the bypassed condition for up to 12 hours while performing routine surveillance testing of other channels. With one channel inoperable, the Note also allows routine surveillance testing of another channel with the inoperable channel in bypass. The Note also allows placing the inoperable channel in the bypass condition to allow setpoint adjustments of other channels when required to reduce the Power Range Neutron Flux-High setpoint in accordance with other Technical Specifications. The 12 hour time limit is justified in Reference 8.

Note 2 states to perform SR 3.2.4.2 if input to QPTR from one or more Power Range Neutron Flux channels are inoperable with THERMAL POWER > 75% RTP.

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### ACTIONS (continued)

#### E.1 and E.2

Condition E applies to the following reactor trip Functions:

- Power Range Neutron Flux - Low,
- Overtemperature  $\Delta T$ ,
- Overpower  $\Delta T$ ,
- Power Range Neutron Flux - High Positive Rate,
- Power Range Neutron Flux - High Negative Rate, and
- Pressurizer Pressure - High.

A known inoperable channel must be placed in the tripped condition within 72 hours. Placing the channel in the tripped condition results in a partial trip condition requiring only one-out-of-two logic for actuation of the two-out-of-three trips and one-out-of-three logic for actuation of the two-out-of-four trips. The 72 hours allowed to place the inoperable channel in the tripped condition is justified in Reference 8.

If the inoperable channel cannot be placed in the trip condition within the specified Completion Time, the unit must be placed in a MODE where these Functions are not required OPERABLE. An additional 6 hours is allowed to place the unit in MODE 3. Six hours is a reasonable time, based on operating experience, to place the unit in MODE 3 from full power in an orderly manner and without challenging unit systems.

The Required Actions have been modified by a Note that allows placing the inoperable channel in the bypassed condition for up to 12 hours while performing routine surveillance testing of the other channels. The 12 hour time limit is justified in Reference 8.

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## ACTIONS (continued)

F.1 and F.2

Condition F applies to the Intermediate Range Neutron Flux trip when THERMAL POWER is above the P-6 setpoint and below the P-10 setpoint and one channel is inoperable. Above the P-6 setpoint and below the P-10 setpoint, the NIS intermediate range detector performs the monitoring Functions. If THERMAL POWER is greater than the P-6 setpoint but less than the P-10 setpoint, 24 hours is allowed to reduce THERMAL POWER below the P-6 setpoint or to increase THERMAL POWER above the P-10 setpoint. The NIS Intermediate Range Neutron Flux channels must be OPERABLE when the power level is above the capability of the source range, P-6, and below the capability of the power range, P-10. If THERMAL POWER is greater than the P-10 setpoint, the NIS power range detectors perform the monitoring and protection functions and the intermediate range is not required. The Completion Times allow for a slow and controlled power adjustment above P-10 or below P-6 and take into account the redundant capability afforded by the redundant OPERABLE channel, and the low probability of its failure during this period. This action does not require the inoperable channel to be tripped because the Function uses one-out-of-two logic. Tripping one channel would trip the reactor. Thus, the Required Actions specified in this Condition are only applicable when channel failure does not result in reactor trip.

G.1 and G.2

Condition G applies to two inoperable Intermediate Range Neutron Flux trip channels in MODE 2 when THERMAL POWER is above the P-6 setpoint and below the P-10 setpoint. Required Actions specified in this Condition are only applicable when channel failures do not result in reactor trip. Above the P-6 setpoint and below the P-10 setpoint, the NIS intermediate range detector performs the monitoring Functions. With no intermediate range channels OPERABLE, the Required Actions are to suspend operations involving positive reactivity additions immediately. This will preclude any power level increase since there are no OPERABLE Intermediate Range Neutron Flux channels. The operator must also reduce THERMAL POWER below the P-6 setpoint within two hours. Below P-6, the Source Range Neutron Flux channels will be able to monitor the core power level. The Completion Time of 2 hours will allow a slow and controlled power reduction to less than the P-6 setpoint and takes into account the low probability of occurrence of an event during this period that may require the protection afforded by the NIS Intermediate Range Neutron Flux trip.

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### ACTIONS (continued)

Required Action G.1 is modified by a Note to indicate that normal plant control operations that individually add limited positive reactivity (e.g., temperature or boron fluctuations associated with RCS inventory management or temperature control) are not precluded by this Action, provided they are accounted for in the calculated SDM.

#### H.1

Condition H applies to one inoperable Source Range Neutron Flux trip channel when in MODE 2, below the P-6 setpoint, and performing a reactor startup. With the unit in this Condition, below P-6, the NIS source range performs the monitoring and protection functions. With one of the two channels inoperable, operations involving positive reactivity additions shall be suspended immediately.

This will preclude any power escalation. With only one source range channel OPERABLE, core protection is severely reduced and any actions that add positive reactivity to the core must be suspended immediately.

Required Action H.1 is modified by a Note to indicate that normal plant control operations that individually add limited positive reactivity (e.g., temperature or boron fluctuations associated with RCS inventory management or temperature control) are not precluded by this Action, provided they are accounted for in the calculated SDM.

#### I.1

Condition I applies to two inoperable Source Range Neutron Flux trip channels when in MODE 2, below the P-6 setpoint, and in MODE 3, 4, or 5 with the Rod Control System capable of rod withdrawal or one or more rods not fully inserted. With the unit in this Condition, below P-6, the NIS source range performs the monitoring and protection functions. With both source range channels inoperable, the reactor trip breakers must be opened immediately. With the reactor trip breakers open, the core is in a more stable condition.

#### J.1, J.2.1, and J.2.2

Condition J applies to one inoperable source range channel in MODE 3, 4, or 5 with the Rod Control System capable of rod withdrawal or one or more rods not fully inserted. With the unit in this Condition, below P-6, the NIS source range performs the monitoring and protection functions. With one of the source range channels inoperable, 48 hours is allowed to

## BASES

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### ACTIONS (continued)

restore it to an OPERABLE status. If the channel cannot be returned to an OPERABLE status, action must be initiated within the same 48 hours to ensure that all rods are fully inserted, and the Rod Control System must be placed in a condition incapable of rod withdrawal within the next hour.

#### K.1 and K.2

Condition K applies to the following reactor trip Functions:

- Pressurizer Pressure - Low,
- Pressurizer Water Level - High,
- Reactor Coolant Flow – Low,
- Undervoltage RCPs, and
- Underfrequency RCPs.

With one channel inoperable, the inoperable channel must be placed in the tripped condition within 72 hours (Ref. 8). For the Pressurizer Pressure - Low, Pressurizer Water Level - High, Undervoltage RCPs, and Underfrequency RCPs trip Functions, placing the channel in the tripped condition when above the P-7 setpoint results in a partial trip condition requiring only one additional channel to initiate a reactor trip. For the Reactor Coolant Flow - Low trip Function, placing the channel in the tripped condition when above the P-8 setpoint results in a partial trip condition requiring only one additional channel in the same loop to initiate a reactor trip. For the latter trip Function, two tripped channels in two RCS loops are required to initiate a reactor trip when below the P-8 setpoint and above the P-7 setpoint. These Functions do not have to be OPERABLE below the P-7 setpoint because there are no loss of flow trips below the P-7 setpoint. There is insufficient heat production to generate DNB conditions below the P-7 setpoint. The 72 hours allowed to place the channel in the tripped condition is justified in Reference 8. An additional 6 hours is allowed to reduce THERMAL POWER to below P-7 if the inoperable channel cannot be restored to OPERABLE status or placed in trip within the specified Completion Time.

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### ACTIONS (continued)

Allowance of this time interval takes into consideration the redundant capability provided by the remaining redundant OPERABLE channel, and the low probability of occurrence of an event during this period that may require the protection afforded by the Functions associated with Condition K.

The Required Actions have been modified by a Note that allows placing the inoperable channel in the bypassed condition for up to 12 hours while performing routine surveillance testing of the other channels. The 12 hour time limit is justified in Reference 8.

#### L.1 and L.2

Condition L applies to Turbine Trip on Low Fluid Oil Pressure or on Turbine Stop Valve Closure. With one channel inoperable, the inoperable channel must be placed in the trip condition within 72 hours. If placed in the tripped condition, this results in a partial trip condition requiring only one additional Low Fluid Oil Pressure channel or three additional Turbine Stop Valve Closure channels to initiate a reactor trip. If the channel cannot be restored to OPERABLE status or placed in the trip condition, then power must be reduced below the P-9 setpoint within the next 4 hours. The 72 hours allowed to place the inoperable channel in the tripped condition is justified in Reference 8. Four hours is allowed for reducing power.

The Required Actions have been modified by a Note that allows placing the inoperable channel in the bypassed condition for up to 12 hours while performing routine surveillance testing of the other channels. The 12 hour time limit is justified in Reference 8.

#### M.1 and M.2

Condition M applies to the SI Input from ESFAS reactor trip and the RTS Automatic Trip Logic in MODES 1 and 2. These actions address the train orientation of the RTS for these Functions. With one train inoperable, 24 hours are allowed to restore the train to OPERABLE status (Required Action M.1) or the unit must be placed in MODE 3 within the next 6 hours. The Completion Time of 24 hours (Required Action M.1) is reasonable considering that in this Condition, the remaining OPERABLE train is adequate to perform the safety function and given the low probability of an event during this interval. The 24 hours allowed to restore the inoperable RTS Automatic Trip Logic train to OPERABLE status is justified in Reference 8. An additional 6 hours is allowed to place the unit in MODE 3. Six hours is reasonable, based on operating experience,

## BASES

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### ACTIONS (continued)

to reach MODE 3 from full power in an orderly manner and without challenging unit systems.

The Required Actions have been modified by a Note that allows bypassing one train up to 4 hours for surveillance testing, provided the other train is OPERABLE. The 4 hour time limit for testing the RTS Automatic Trip logic train may include testing the reactor trip breaker also, if both the Logic test and reactor trip breaker test are conducted within the 4 hour time limit. The 4 hour time limit is justified in Reference 8.

#### N.1 and N.2

Condition N applies to the reactor trip breakers in MODES 1 and 2. These actions address the train orientation of the RTS for the reactor trip breakers. With one train inoperable, 24 hours is allowed for train corrective maintenance to restore the train to OPERABLE status or the unit must be placed in MODE 3 within the next 6 hours. The 24 hour Completion Time is justified in Reference 12. An additional 6 hours is allowed to place the unit in MODE 3. Six hours is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging unit systems.

Placing the unit in MODE 3 results in Condition C entry while a reactor trip breaker is inoperable.

The Required Actions have been modified by a Note. The Note allows one train to be bypassed for up to 4 hours for surveillance testing, provided the other train is OPERABLE. The 4 hour time limit is justified in Reference 12.

#### O.1 and O.2

Condition O applies to the P-6 and P-10 interlocks. With one or more channels inoperable for one-out-of-two or two-out-of-four coincidence logic, the associated interlock must be verified to be in its required state for the existing unit condition within 1 hour or the unit must be placed in MODE 3 within the next 6 hours. Verifying the interlock status manually accomplishes the interlock's Function. The Completion Time of 1 hour is based on operating experience and the minimum amount of time allowed for manual operator actions. An additional 6 hours is allowed to place the unit in MODE 3. Six hours is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging unit systems. The 1 hour and 6 hour Completion

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### ACTIONS (continued)

Times are equal to the time allowed by LCO 3.0.3 for shutdown actions in the event of a complete loss of RTS Function.

#### P.1 and P.2

Condition P applies to the P-7, P-8, P-9, and P-13 interlocks. With one or more channels inoperable for one-out-of-two or two-out-of-four coincidence logic, the associated interlock must be verified to be in its required state for the existing unit condition within 1 hour or the unit must be placed in MODE 2 within the next 6 hours. These actions are conservative for the case where power level is being raised. Verifying the interlock status manually accomplishes the interlock's Function. The Completion Time of 1 hour is based on operating experience and the minimum amount of time allowed for manual operator actions. An additional 6 hours is allowed to place the unit in MODE 2. Six hours is reasonable, based on operating experience, to reach MODE 2 from full power in an orderly manner and without challenging unit systems.

#### Q.1 and Q.2

Condition Q applies to the reactor trip breaker Undervoltage and Shunt Trip Mechanisms, or diverse trip features, in MODES 1 and 2. With one of the diverse trip features inoperable, it must be restored to an OPERABLE status within 48 hours or the unit must be placed in a MODE where the requirement does not apply. This is accomplished by placing the unit in MODE 3 within the next 6 hours (54 hours total time). Six hours is a reasonable time, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging unit systems. With the unit in MODE 3, ACTION C would apply to any inoperable reactor trip breaker trip mechanism. The Required Actions have been modified by a Note. The Note states that the affected reactor trip breaker shall not be bypassed while one of the diverse features is inoperable except for up to 4 hours to perform maintenance to one of the diverse features. The allowable time for performing maintenance of the diverse features is 4 hours for the reasons stated under Condition N.

The Completion Time of 48 hours for Required Action Q.1 is reasonable considering that in this Condition there is one remaining diverse feature for the affected reactor trip breaker, and one OPERABLE reactor trip breaker capable of performing the safety function and given the low probability of an event occurring during this interval.

## BASES

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### ACTIONS (continued)

#### R.1 and R.2

Condition R applies to the following reactor trip Functions:

- Steam Generator Water Level--Low-Low (Adverse), and
- Steam Generator Water Level--Low-Low (EAM)

A known inoperable channel must be placed in the tripped condition within 6 hours. Placing the channel in the tripped condition results in a partial trip condition requiring only one-out-of-two logic for actuation of the two-out-of-three trips.

In addition to placing the channel in the tripped condition it is also necessary to force the use of the shorter TTD by adjustment of the single steam generator time delay calculation ( $T_S$ ) to match the multiple steam generator time delay calculation ( $T_M$ ) for the affected protection set within 4 hours.

The Required Actions have been modified by a Note that allows placing the inoperable channel in the bypassed condition for up to 4 hours while performing routine surveillance testing of the other channels.

#### S.1, S.2, and S.3

Condition S applies to the Containment Pressure (EAM) coincident with Steam Generator Water Level--Low-Low (Adverse) reactor trip.

Failure of the Containment Pressure (EAM) channel to a protection set does not affect the EAM setpoint calculations. A known inoperable Containment Pressure channel results in the requirement to adjust the Steam Generator Water Level - Low-Low (EAM) channels trip setpoints for the affected protection set to the same value as Steam Generator Water Level - Low-Low (Adverse) within 6 hours.

An alternative to adjusting the affected Steam Generator Water Level - Low-Low (EAM) trip setpoints to the same value as the Steam Generator Water Level - Low-Low (Adverse) trip setpoints is to place the associated protection set's SG Water Level Low-Low channels in the tripped condition within 6 hours

If neither of the above Required Actions are completed within their associated Completion Time, then the unit must be placed in a MODE where these Functions are not required OPERABLE. This requires the

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### ACTIONS (continued)

unit be placed in MODE 3 within 12 hours. The allowed Completion Times are reasonable to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. In MODE 3, these Functions are no longer required OPERABLE.

#### T.1, T.2, and T.3

Condition T applies to the RCS Loop  $\Delta T$  coincident with SG Water Level -  
- Low Low reactor trips.

Failure of the RCS loop  $\Delta T$  channel input (failure of more than one  $T_H$  RTD or failure of a  $T_C$  RTD) does not affect the TTD calculation for a protection set. This results in the requirement that the operator adjust the threshold power level for zero seconds time delay from 50% RTP to 0% RTP within 6 hours. With the trip time delay adjusted to zero seconds the additional operational margin that allows the operator time to recover SG level is removed.

An alternative to adjusting the threshold power level for zero seconds time delay is to place the affected protection set's SG Water Level Low-Low level channels in the tripped condition within 6 hours.

If neither of the above Required Actions can be completed within their associated Completion Times then the unit must be placed in a MODE where these Functions are not required OPERABLE. This requires the unit be placed in MODE 3 within 12 hours. The allowed Completion Times are reasonable to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. In MODE 3, these Functions are no longer required OPERABLE.

#### U.1

If the Required Action is not met within the specified Completion Time of Condition R, the unit must be placed in a MODE where this Function is not required OPERABLE. Six hours is allowed to place the unit in MODE 3. Six hours is a reasonable time, based on operating experience, to place the unit in MODE 3 from full power in an orderly manner and without challenging unit systems.

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### SURVEILLANCE REQUIREMENTS

The SRs for each RTS Function are identified by the SRs column of Table 3.3.1-1 for that Function.

A Note has been added to the SR Table stating that Table 3.3.1-1 determines which SRs apply to which RTS Functions.

Note that each channel of process protection supplies both trains of the RTS. When testing Channel I, Train A and Train B must be examined. Similarly, Train A and Train B must be examined when testing Channel II, Channel III, and Channel IV (if applicable). The CHANNEL CALIBRATION and COTs are performed in a manner that is consistent with the assumptions used in analytically calculating the required channel accuracies.

#### SR 3.3.1.1

Performance of the CHANNEL CHECK ensures that gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the unit staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.2

SR 3.3.1.2 compares the calorimetric heat balance calculation to the power range channel output. If the absolute difference is greater than 2 percent, the power range channel is not declared inoperable, but must be adjusted. The power range channel output shall be adjusted consistent with the calorimetric heat balance calculation results if the absolute difference is greater than 2 percent. If the power range channel output cannot be properly adjusted, the channel is declared inoperable.

The Note clarifies that this Surveillance is required only if reactor power is  $\geq 15\%$  RTP and that 12 hours are allowed for performing the first Surveillance after reaching 15% RTP. A power level of 15% RTP is chosen based on plant stability, i.e., automatic rod control capability and turbine generator synchronized to the grid.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.3.1.3

SR 3.3.1.3 compares the incore system to the NIS channel output. If the absolute difference is  $\geq 3\%$ , the NIS channel is still OPERABLE, but must be readjusted. The excore NIS channel shall be adjusted if the absolute difference between the incore and excore AFD is  $\geq 3\%$ .

If the NIS channel cannot be properly readjusted, the channel is declared inoperable. This Surveillance is performed to verify the  $f(\Delta I)$  input to the overtemperature  $\Delta T$  and overpower  $\Delta T$  Functions.

A Note clarifies that the Surveillance is required only if reactor power is  $\geq 15\%$  RTP and that 96 hours is allowed for performing the first Surveillance after reaching 15% RTP.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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### SURVEILLANCE REQUIREMENTS (continued)

#### SR 3.3.1.4

SR 3.3.1.4 is the performance of a TADOT. This test shall verify OPERABILITY by actuation of the end devices. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable TADOT of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions.

The reactor trip breaker test shall include separate verification of the undervoltage and shunt trip mechanisms. Independent verification of reactor trip breaker undervoltage and shunt trip Function is not required for the bypass breakers. No capability is provided for performing such a test at power. The independent test for bypass breakers is included in SR 3.3.1.12. The bypass breaker test shall include a local shunt trip. A Note has been added to indicate that this test must be performed on the bypass breaker prior to placing it in service.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

#### SR 3.3.1.5

SR 3.3.1.5 is the performance of an ACTUATION LOGIC TEST. The SSPS is tested using the semiautomatic tester. The train being tested is placed in the bypass condition, thus preventing inadvertent actuation. Through the semiautomatic tester, all possible logic combinations, with and without applicable permissives, are tested for each protection function, including operation of the P-7 permissive which is a logic function only.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

#### SR 3.3.1.6

SR 3.3.1.6 is a calibration of the excore channels to the incore channels. If the measurements do not agree, the excore channels are not declared inoperable but must be calibrated to agree with the incore detector measurements. If the excore channels cannot be adjusted, the channels are declared inoperable. This Surveillance is performed to verify the  $f(\Delta I)$  input to the overtemperature  $\Delta T$  and overpower  $\Delta T$  Functions.

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### SURVEILLANCE REQUIREMENTS (continued)

A Note modifies SR 3.3.1.6. The Note states that this Surveillance is required only if reactor power is > 50% RTP and that 24 hours is allowed for performing the first surveillance after reaching 50% RTP.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

#### SR 3.3.1.7

SR 3.3.1.7 is the performance of a COT.

A COT is performed on each required channel to ensure the entire channel will perform the intended Function. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable COT of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions.

Setpoints must be conservative with respect to the Allowable Values specified in Table 3.3.1-1.

The difference between the current "as found" values and the previous test "as left" values must be consistent with the drift allowance used in the setpoint methodology. The setpoint shall be left set consistent with the assumptions of the current unit specific setpoint methodology.

The "as-found" and "as-left" values must also be recorded and reviewed for consistency with the assumptions of Reference 9.

SR 3.3.1.7 is modified by a Note that provides a 4 hour delay in the requirement to perform this Surveillance for source range instrumentation when entering MODE 3 from MODE 2. This Note allows a normal shutdown to proceed without a delay for testing in MODE 2 and for a short time in MODE 3 until the reactor trip breakers are open and SR 3.3.1.7 is no longer required to be performed. If the unit is to be in MODE 3 with the reactor trip breakers closed for > 4 hours this Surveillance must be performed prior to 4 hours after entry into MODE 3.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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## SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.7 is modified by two Notes as identified in Table 3.3.1-1. The first Note requires evaluation of channel performance for the condition where the as-found setting for the channel setpoint is outside its as-found tolerance but conservative with respect to the Allowable Value. Evaluation of channel performance will verify that the channel will continue to behave in accordance with safety analysis assumptions and the channel performance assumptions in the setpoint methodology. The purpose of the assessment is to ensure confidence in the channel performance prior to returning the channel to service. For channels determined to be OPERABLE but degraded, after returning the channel to service the channels will be evaluated under the plant Corrective Action Program. Entry into the Corrective Action Program will ensure required review and documentation of the condition. The second Note requires that the as-left setting for the channel be returned to within the as-left tolerance of the NTSP. Where a setpoint more conservative than the NTSP is used in the plant surveillance procedures (field setting), the as-left and as-found tolerances, as applicable, will be applied to the surveillance procedure setpoint. This will ensure that sufficient margin to the Safety Limit and/or Analytical Limit is maintained. If the as-left channel setting cannot be returned to a setting within the as-left tolerance of the NTSP, then the channel shall be declared inoperable.

The second Note also requires that the methodologies for calculating the as-left and the as-found tolerances be in UFSAR, Section 7.1.2.

SR 3.3.1.8

SR 3.3.1.8 is the performance of a COT as described in SR 3.3.1.7, except it is modified by a Note stating that this test shall include verification that the P-6 and P-10 interlocks are in their required state for the existing unit condition. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable COT of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. The Frequency is modified by a Note that allows this surveillance to be satisfied if it has been performed within the Frequency specified in the Surveillance Frequency Control Program or of the Frequencies prior to reactor startup and after reducing power below P-10 and P-6. The Frequency of "prior to startup" ensures this surveillance is performed prior to critical operations and applies to the source, intermediate and power range low instrument channels. The Frequency of 12 hours after reducing power below P-10 (applicable to intermediate and power range low channels) and 4 hours after reducing

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SURVEILLANCE REQUIREMENTS (continued)

power below P-6 (applicable to source range channels) allows a normal shutdown to be completed and the unit removed from the MODE of Applicability for this surveillance without a delay to perform the testing required by this surveillance. The Frequency thereafter applies if the plant remains in the MODE of Applicability after the initial performances of prior to reactor startup and 12 and four hours after reducing power below P-10 or P-6, respectively. The MODE of Applicability for this surveillance is < P-10 for the power range low and intermediate range channels and < P-6 for the source range channels. Once the unit is in MODE 3, this surveillance is no longer required. If power is to be maintained < P-10 for more than 12 hours or < P-6 for more than 4 hours, then the testing required by this surveillance must be performed prior to the expiration of the time limit. Twelve hours and four hours are reasonable times to complete the required testing or place the unit in a MODE where this surveillance is no longer required. This test ensures that the NIS source, intermediate, and power range low channels are OPERABLE prior to taking the reactor critical and after reducing power into the applicable MODE (< P-10 or < P-6) for periods > 12 and 4 hours, respectively.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.3.1.8 is modified by two Notes as identified in Table 3.3.1-1. The first Note requires evaluation of channel performance for the condition where the as-found setting for the channel setpoint is outside its as-found tolerance but conservative with respect to the Allowable Value. Evaluation of channel performance will verify that the channel will continue to behave in accordance with safety analysis assumptions and the channel performance assumptions in the setpoint methodology. The purpose of the assessment is to ensure confidence in the channel performance prior to returning the channel to service. For channels determined to be OPERABLE but degraded, after returning the channel to service the performance of these channels will be evaluated under the plant Corrective Action Program. Entry into the Corrective Action Program will ensure required review and documentation of the condition. The second Note requires that the as-left setting for the channel be returned to within the as-left tolerance of the NTSP. Where a setpoint more conservative than the NTSP is used in the plant surveillance procedures (field setting), the as-left and as-found tolerances, as applicable, will be applied to the surveillance procedure setpoint. This will ensure that sufficient margin to the Safety Limit and/or Analytical Limit is maintained. If the as-left channel setting cannot be returned to a setting within the as-left tolerance of the NTSP, then the channel shall be declared inoperable.

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### SURVEILLANCE REQUIREMENTS (continued)

The second Note also requires that the methodologies for calculating the as-left and the as-found tolerances be in UFSAR, Section 7.1.2.

#### SR 3.3.1.9

SR 3.3.1.9 is the performance of a TADOT.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable TADOT of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions.

The SR is modified by a Note that excludes verification of setpoints from the TADOT. Since this SR applies to RCP undervoltage and underfrequency relays, setpoint verification requires elaborate bench calibration and is accomplished during the CHANNEL CALIBRATION.

#### SR 3.3.1.10

CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy.

CHANNEL CALIBRATIONS must be performed consistent with the assumptions of the unit specific setpoint methodology. The difference between the current "as-found" values and NTSP must be consistent with the drift allowance used in the setpoint methodology.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.3.1.10 is modified by a Note stating that this test shall include verification that the time constants are adjusted to the prescribed values where applicable.

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## SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.10 is modified by two Notes as identified in Table 3.3.1-1. The first Note requires evaluation of channel performance for the condition where the as-found setting for the channel setpoint is outside its as-found tolerance but conservative with respect to the Allowable Value. Evaluation of channel performance will verify that the channel will continue to behave in accordance with safety analysis assumptions and the channel performance assumptions in the setpoint methodology. The purpose of the assessment is to ensure confidence in the channel performance prior to returning the channel to service. For channels determined to be OPERABLE but degraded, after returning the channel to service the performance of these channels will be evaluated under the plant Corrective Action Program. Entry into the Corrective Action Program will ensure required review and documentation of the condition. The second Note requires that the as-left setting for the channel be returned to within the as-left tolerance of the NTSP. Where a setpoint more conservative than the NTSP is used in the plant surveillance procedures (field setting), the as-left and as-found tolerances, as applicable, will be applied to the surveillance procedure setpoint. This will ensure that sufficient margin to the Safety Limit and/or Analytical Limit is maintained. If the as-left channel setting cannot be returned to a setting within the as-left tolerance of the NTSP, then the channel shall be declared inoperable.

The second Note also requires that the methodologies for calculating the as-left and the as-found tolerances be in UFSAR, Section 7.1.2.

SR 3.3.1.11

SR 3.3.1.11 is the performance of a CHANNEL CALIBRATION, as described in SR 3.3.1.10. This SR is modified by a Note stating that neutron detectors are excluded from the CHANNEL CALIBRATION. The CHANNEL CALIBRATION for the power range neutron detectors consists of a normalization of the detectors based on a power calorimetric and flux map performed above 15% RTP. The CHANNEL CALIBRATION for the source range consists of checking the discriminator voltage and adjusting if necessary. The CHANNEL CALIBRATION for the intermediate range neutron detectors consists of comparing the output of the intermediate range drawer to the secondary side calorimetric and adjusting if necessary. This Surveillance is not required for the NIS power range detectors for entry into MODE 2 or 1, and is not required for the NIS intermediate range detectors for entry into MODE 2, because the unit must be in at least MODE 2 to perform the test for the intermediate range detectors and MODE 1 for the power range detectors.

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### SURVEILLANCE REQUIREMENTS (continued)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.3.1.11 is modified by two Notes as identified in Table 3.3.1-1. The first Note requires evaluation of channel performance for the condition where the as-found setting for the channel setpoint is outside its as-found tolerance but conservative with respect to the Allowable Value. Evaluation of channel performance will verify that the channel will continue to behave in accordance with safety analysis assumptions and the channel performance assumptions in the setpoint methodology. The purpose of the assessment is to ensure confidence in the channel performance prior to returning the channel to service. For channels determined to be OPERABLE but degraded, after returning the channel to service the performance of these channels will be evaluated under the plant Corrective Action Program. Entry into the Corrective Action Program will ensure required review and documentation of the condition. The second Note requires that the as-left setting for the channel be returned to within the as-left tolerance of the NTSP. Where a setpoint more conservative than the NTSP is used in the plant surveillance procedures (field setting), the as-left and as-found tolerances, as applicable, will be applied to the surveillance procedure setpoint. This will ensure that sufficient margin to the Safety Limit and/or Analytical Limit is maintained. If the as-left channel setting cannot be returned to a setting within the as-left tolerance of the NTSP, then the channel shall be declared inoperable.

The second Note also requires that the methodologies for calculating the as-left and the as-found tolerances be in UFSAR, Section 7.1.2.

#### SR 3.3.1.12

SR 3.3.1.12 is the performance of a TADOT of the Manual Reactor Trip and the SI Input from ESFAS. The test shall independently verify the OPERABILITY of the undervoltage and shunt trip mechanisms for the Manual Reactor Trip Function for the Reactor Trip Breakers and Reactor Trip Bypass Breakers. The Reactor Trip Bypass Breaker test shall include testing of the automatic and manual undervoltage trip.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

The SR is modified by a Note that excludes verification of setpoints from the TADOT. The Functions affected have no setpoints associated with them.

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## SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.13

SR 3.3.1.13 is the performance of a TADOT of Turbine Trip Functions. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable TADOT of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. This TADOT is as described in SR 3.3.1.4, except that this test is performed prior to exceeding the P-9 interlock whenever the unit has been in MODE 3. This Surveillance is not required if it has been performed within the previous 31 days. Verification of the Trip Setpoint does not have to be performed for this Surveillance. Performance of this test will ensure that the turbine trip Function is OPERABLE prior to exceeding the P-9 interlock.

SR 3.3.1.14

SR 3.3.1.14 verifies that the individual channel/train actuation response times are less than or equal to the maximum values assumed in the accident analysis. Response time testing acceptance criteria are included in UFSAR Table 7.2.1-5. Individual component response times are not modeled in the analyses.

The analyses model the overall or total elapsed time, from the point at which the parameter exceeds the trip setpoint value at the sensor to the point at which the equipment reaches the required functional state (i.e., control and shutdown rods fully inserted in the reactor core).

For channels that include dynamic transfer Functions (e.g., lag, lead/lag, rate/lag, etc.), the response time test may be performed with the transfer Function set to one, with the resulting measured response time compared to the appropriate UFSAR response time. Alternately, the response time test can be performed with the time constants set to their nominal value, provided the required response time is analytically calculated assuming the time constants are set at their nominal values. The response time may be measured by a series of overlapping tests such that the entire response time is measured.

Response time may be verified by actual response time tests in any series of sequential, overlapping or total channel measurements, or by the summation of allocated sensor, signal processing and actuation logic response times with actual response time tests on the remainder of the

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### SURVEILLANCE REQUIREMENTS (continued)

channel. Allocations for sensor response times may be obtained from: (1) historical records based on acceptable response time tests (hydraulic, noise, or power interrupt tests), (2) in place, onsite, or offsite (e.g., vendor) test measurements, or (3) utilizing vendor engineering specifications. WCAP-13632-P-A, Revision 2, "Elimination of Pressure Sensor Response Time Testing Requirements," (Ref. 10) provides the basis and methodology for using allocated sensor response times in the overall verification of the channel response time for specific sensors identified in the WCAP. Response time verification for other sensor types must be demonstrated by test.

WCAP-14036-P, Revision 1, "Elimination of Periodic Protection Channel Response Time Tests," (Ref. 13) provides the basis and methodology for using allocated signal processing and actuation logic response times in the overall verification of the protection system channel response time. The allocations for sensor, signal conditioning, and actuation logic response times must be verified prior to placing the component in operational service and re-verified following maintenance that may adversely affect response time. In general, electrical repair work does not impact response time provided the parts used for repair are of the same type and value. Specific components identified in the WCAP may be replaced without verification testing. One example where response time could be affected is replacing the sensing assembly of a transmitter.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.3.1.14 is modified by a Note stating that neutron detectors are excluded from RTS RESPONSE TIME testing. This Note is necessary because of the difficulty in generating an appropriate detector input signal. Excluding the detectors is acceptable because the principles of detector operation ensure a virtually instantaneous response.

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- REFERENCES
1. Regulatory Guide 1.105, Revision 3, "Setpoints for Safety Related Instrumentation."
  2. UFSAR, Chapter 7.
  3. UFSAR, Chapter 6.
  4. UFSAR, Chapter 15.
  5. IEEE-279-1971.
  6. 10 CFR 50.49.
  7. Calculation SQN-EEB-PL&S, Precautions, Limitations, and Setpoints for NSSS.
  8. WCAP-14333-P-A, Rev. 1, October 1998.
  9. WCAP-10271-P-A, Supplement 1, May 1986.
  10. WCAP-13632-P-A, Revision 2, "Elimination of Pressure Sensor Response Time Testing Requirements," January 1996.
  11. WCAP-10271-P-A, Supplement 2, June 1990.
  12. WCAP-15376, Rev. 0, October 2000.
  13. WCAP-14036-P, Revision 1, "Elimination of Periodic Protection Channel Response Time Tests," December 1995.
  14. Letter from Siva P. Lingam (NRC) to Joseph W. Shea (TVA), "Sequoyah Nuclear Plant, Units 1 and 2 - Issuance of Amendments to Revise the Technical Specification to allow use of Areva Advanced W17 High Performance Fuel (TS-SQN-2011-07) (TAC NOS. ME6538 and ME6539)," dated September 26, 2012.
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## B 3.3 INSTRUMENTATION

### B 3.3.2 Engineered Safety Feature Actuation System (ESFAS) Instrumentation

#### BASES

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**BACKGROUND** The ESFAS initiates necessary safety systems, based on the values of selected unit parameters, to protect against violating core design limits and the Reactor Coolant System (RCS) pressure boundary, and to mitigate accidents. This is achieved by specifying limiting safety system settings (LSSS) in terms of parameters directly monitored by the ESFAS, as well as specifying LCOs on other reactor system parameters and equipment performance.

Technical Specifications are required by 10 CFR 50.36 to include LSSS. LSSS are defined by the regulation as settings for automatic protective devices related to those variables having significant safety functions. The regulation also states, "Where a LSSS is specified for a variable on which a safety limit has been placed, the setting must be chosen so that automatic protective actions will correct the abnormal situation before a Safety Limit (SL) is exceeded." The Analytical Limit is the limit of the process variable at which a protective action is initiated, as established by the safety analysis, to ensure that a SL is not exceeded. Any automatic protection action that occurs on reaching the Analytical Limit therefore ensures that the SL is not exceeded. However, in practice, the actual settings for automatic protection channels must be chosen to be more conservative than the Analytical Limit to account for instrument loop uncertainties related to the setting at which the automatic protective action would actually occur.

The Nominal Trip Setpoint (NTSP) specified in Table 3.3.2-1 is a predetermined setting for a protection channel chosen to ensure automatic actuation prior to the process variable reaching the Analytical Limit and thus ensuring that the SL would not be exceeded. As such, the NTSP accounts for uncertainties in setting the channel (e.g., calibration), uncertainties in how the channel might actually perform (e.g., repeatability), changes in the point of action of the channel over time (e.g., drift during surveillance intervals), and any other factors which may influence its actual performance (e.g., harsh accident environments). In this manner, the NTSP ensures that SLs are not exceeded. Therefore, the NTSP meets the definition of an LSSS (Ref. 1).

Technical Specifications contain values related to the OPERABILITY of equipment required for safe operation of the facility. OPERABLE is defined in Technical Specifications as "...being capable of performing its safety functions(s)." Relying solely on the NTSP to define OPERABILITY in Technical Specifications would be an overly restrictive requirement if it were applied as an OPERABILITY limit for the "as-found" value of a protection channel setting during a surveillance. This would result in

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### BACKGROUND (continued)

Technical Specification compliance problems, as well as reports and corrective actions required by the rule which are not necessary to ensure safety. For example, an automatic protection channel with a setting that has been found to be different from the NTSP due to some drift of the setting may still be OPERABLE since drift is to be expected. This expected drift would have been specifically accounted for in the setpoint methodology for calculating the NTSP and thus the automatic protective action would still have ensured that the SL would not be exceeded with the "as-found" setting of the protection channel. Therefore, the channel would still be OPERABLE since it would have performed its safety function and the only corrective action required would be to reset the channel within the established as-left tolerance around the NTSP to account for further drift during the next surveillance interval.

During Anticipated Operational Occurrences (AOOs), which are those events expected to occur one or more times during the unit life, the acceptable limits are:

1. The Departure from Nucleate Boiling Ratio (DNBR) shall be maintained above the SL value to prevent departure from nucleate boiling (DNB),
2. Fuel centerline melt shall not occur, and
3. The RCS pressure SL of 2735 psig shall not be exceeded.

Operation within the SLs of Specification 2.0, "Safety Limits (SLs)," also maintains the above values and assures that offsite dose will be within the 10 CFR 50 and 10 CFR 100 criteria during AOOs.

Accidents are events that are analyzed even though they are not expected to occur during the unit life. The acceptable limit during accidents is that offsite dose shall be maintained within an acceptable fraction of 10 CFR 100 limits. Different accident categories are allowed a different fraction of these limits, based on probability of occurrence. Meeting the acceptable dose limit for an accident category is considered having acceptable consequences for that event.

The ESFAS instrumentation is segmented into three distinct but interconnected modules as identified below:

- Field transmitters or process sensors and instrumentation: provide a measurable electronic signal based on the physical characteristics of the parameter being measured,

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### BACKGROUND (continued)

- Signal processing equipment including Process protection system, field contacts, and protection channel sets: provide signal conditioning, setpoint comparison, process algorithm actuation, compatible electrical signal output to protection system channels, and control board/control room/miscellaneous indications, and
- Solid State Protection System (SSPS) including input, logic, and output bays: initiates the proper unit shutdown or engineered safety feature (ESF) actuation in accordance with the defined logic and based on the bistable, setpoint comparator, or contact outputs from the signal process control and protection system.

#### Field Transmitters or Sensors

To meet the design demands for redundancy and reliability, more than one, and often as many as four, field transmitters or sensors are used to measure unit parameters. In many cases, field transmitters or sensors that input to the ESFAS are shared with the Reactor Trip System (RTS). In some cases, the same channels also provide control system inputs. To account for calibration tolerances and instrument drift, which are assumed to occur between calibrations, statistical allowances are provided in the NTSP and Allowable Value. The OPERABILITY of each transmitter or sensor is determined by either "as-found" calibration data evaluated during the CHANNEL CALIBRATION or by qualitative assessment of field transmitter or sensor, as related to the channel behavior observed during performance of the CHANNEL CHECK.

#### Signal Processing Equipment

Generally, three or four channels of process control equipment are used for the signal processing of unit parameters measured by the field instruments. The process control equipment provides analog to digital conversion (Digital Protection System), signal conditioning, comparable output signals for instruments located on the main control board, and comparison of measured input signals with NTSPs derived from Analytical Limits established by the safety analyses. Analytical Limits are defined in UFSAR, Chapter 6 (Ref. 2), Chapter 7 (Ref. 3), and Chapter 15 (Ref. 4). If the measured value of a unit parameter exceeds the predetermined setpoint, an output from a bistable, setpoint comparator, or contact is forwarded to the SSPS for decision evaluation. Channel separation is maintained up to and through the input bays. However, not all unit parameters require four channels of sensor measurement and signal processing. Some unit parameters provide input only to the SSPS, while others provide input to the SSPS, and one or more control systems.

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### BACKGROUND (continued)

Generally, if a parameter is used only for input to the protection circuits, three channels with a two-out-of-three logic are sufficient to provide the required reliability and redundancy. If one channel fails in a direction that would not result in a partial Function trip, the Function is still OPERABLE with a two-out-of-two logic. If one channel fails such that a partial Function trip occurs, a trip will not occur and the Function is still OPERABLE with a one-out-of-two logic.

Generally, if a parameter is used for input to the SSPS and a control function, four channels with a two-out-of-four logic are sufficient to provide the required reliability and redundancy. The circuit must be able to withstand both an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation. Again, a single failure will neither cause nor prevent the protection function actuation.

These requirements are described in IEEE-279-1971 (Ref. 5). The actual number of channels required for each unit parameter is specified in Reference 3.

#### NTSPs and ESFAS Setpoints Allowable Values

The trip setpoints used in the bistables, setpoint comparators, or contacts are based on the analytical limits stated in Reference 3. The calculation of the NTSPs specified in Table 3.3.2-1 is such that adequate protection is provided when all sensor and processing time delays are taken into account. To allow for calibration tolerances, instrumentation uncertainties, instrument drift, and severe environment errors for those ESFAS channels that must function in harsh environments as defined by 10 CFR 50.49 (Ref. 6), the Allowable Values specified in Table 3.3.2-1 in the accompanying LCO are conservative with respect to the analytical limits. A detailed description of the methodology used to calculate the Allowable Values and ESFAS NTSPs including their explicit uncertainties, is provided in the plant specific setpoint methodology study (Ref. 7) which incorporates all of the known uncertainties applicable to each channel. The as-left tolerance and as-found tolerance band methodology is provided in UFSAR, Section 7.1.2. The magnitudes of these uncertainties are factored into the determination of each ESFAS NTSP and corresponding Allowable Value. The nominal ESFAS setpoint entered is more conservative than that specified by the Allowable Value to account for measurement errors detectable by the CHANNEL OPERATIONAL TEST (COT). The Allowable Value serves as the as-found Technical Specification OPERABILITY limit for the purpose of the COT.

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### BACKGROUND (continued)

The NTSP is the value at which the bistables or setpoint comparators are set and is the expected value to be achieved during calibration. The NTSP value is the LSSS and ensures the safety analysis limits are met for the surveillance interval selected when a channel is adjusted based on stated channel uncertainties. Any bistable or setpoint comparator is considered to be properly adjusted when the "as-left" NTSP value is within the as-left tolerance for CHANNEL CALIBRATION uncertainty allowance (i.e., + rack calibration and comparator setting uncertainties). The NTSP value is therefore considered a "nominal value" (i.e., expressed as a value without inequalities) for the purposes of the COT and CHANNEL CALIBRATION.

Nominal Trip Setpoints, in conjunction with the use of as-found and as-left tolerances together with the requirements of the Allowable Value ensure that the consequences of Design Basis Accidents (DBAs) will be acceptable, providing the unit is operated from within the LCOs at the onset of the DBA and the equipment functions as designed.

Note that the Allowable Values listed in Table 3.3.2-1 are the least conservative value of the as-found setpoint that a channel can have during a periodic CHANNEL CALIBRATION, COT, or a TADOT.

Each channel can be tested on line to verify that the signal processing equipment and setpoint accuracy is within the specified allowance requirements of Reference 3. Once a designated channel is taken out of service for testing, a simulated signal is injected in place of the field instrument signal. The process equipment for the channel in test is then tested, verified, and calibrated. SRs for the channels are specified in the SR section.

#### Solid State Protection System

The SSPS equipment is used for the decision logic processing of outputs from the signal processing equipment bistables, setpoint comparators, or contacts. To meet the redundancy requirements, two trains of SSPS, each performing the same functions, are provided. If one train is taken out of service for maintenance or test purposes, the second train will provide ESF actuation for the unit. If both trains are taken out of service or placed in test, a reactor trip will result. Each train is packaged in its own cabinet for physical and electrical separation to satisfy separation and independence requirements.

The SSPS performs the decision logic for most ESF equipment actuation; generates the electrical output signals that initiate the required actuation; and provides the status, permissive, and annunciator output signals to the main control room of the unit.

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### BACKGROUND (continued)

The bistable, setpoint comparator, or contact outputs from the signal processing equipment are sensed by the SSPS equipment and combined into logic matrices that represent combinations indicative of various transients. If a required logic matrix combination is completed, the system will send actuation signals via master and slave relays to those components whose aggregate Function best serves to alleviate the condition and restore the unit to a safe condition. Examples are given in the Applicable Safety Analyses, LCO, and Applicability sections of this Bases.

Each SSPS train has a built in testing device that can automatically test the decision logic matrix functions and the actuation channels while the unit is at power. When any one train is taken out of service for testing, the other train is capable of providing unit monitoring and protection until the testing has been completed. The testing device is semiautomatic to minimize testing time.

The actuation of ESF components is accomplished through master and slave relays. The SSPS energizes the master relays appropriate for the condition of the unit. Each master relay then energizes one or more slave relays, which then cause actuation of the end devices. The master and slave relays are routinely tested to ensure operation. The test of the master relays energizes the relay, which then operates the contacts and applies a low voltage to the associated slave relays. The low voltage is not sufficient to actuate the slave relays but only demonstrates signal path continuity. The SLAVE RELAY TEST actuates the devices if their operation will not interfere with continued unit operation. For the latter case, actual component operation is prevented by the SLAVE RELAY TEST circuit, and slave relay contact operation is verified by a continuity check of the circuit containing the slave relay.

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### APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

Each of the analyzed accidents can be detected by one or more ESFAS Functions. One of the ESFAS Functions is the primary actuation signal for that accident. An ESFAS Function may be the primary actuation signal for more than one type of accident. An ESFAS Function may also be a secondary, or backup, actuation signal for one or more other accidents. For example, Pressurizer Pressure - Low is a primary actuation signal for small loss of coolant accidents (LOCAs) and a backup actuation signal for steam line breaks (SLBs) outside containment.

Functions such as manual initiation, not specifically credited in the accident safety analysis, are implicitly credited in the safety analysis and the NRC staff approved licensing basis for the unit. These Functions may provide protection for conditions that do not require dynamic transient analysis to demonstrate Function performance. These Functions may

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### APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

also serve as backups to Functions that were credited in the accident analysis (Ref. 4).

Permissive and interlock setpoints allow the blocking of trips during plant startups, and restoration of trips when the permissive conditions are not satisfied, but they are not explicitly modeled in the Safety Analyses. These permissives and interlocks ensure that the starting conditions are consistent with the safety analysis, before preventive or mitigating actions occur. Because these permissives or interlocks are only one of multiple conservative starting assumptions for the accident analysis, they are generally considered as nominal values without regard to measurement accuracy.

The LCO requires all instrumentation performing an ESFAS Function, listed in Table 3.3.2-1 in the accompanying LCO, to be OPERABLE. The Allowable Value specified in Table 3.3.2-1 is the least conservative value of the as-found setpoint that the channel can have when tested, such that a channel is OPERABLE if the as-found setpoint is within the as-found tolerance and is conservative with respect to the Allowable Value during the CHANNEL CALIBRATION or COT. As such, the Allowable Value differs from the NTSP by an amount greater than or equal to the expected instrument channel uncertainties, such as drift, during the surveillance interval. In this manner, the actual setting of the channel NTSP will ensure that a SL is not exceeded at any given point of time as long as the channel has not drifted beyond expected tolerances during the surveillance interval. Note that, although the channel is OPERABLE under these circumstances, the trip setpoint must be left adjusted to a value within the as-left tolerance, in accordance with uncertainty assumptions stated in the referenced setpoint methodology (as-left criteria), and confirmed to be operating within the statistical allowances of the uncertainty terms assigned (as-found criteria).

If the actual setting of the channel is found to be conservative with respect to the Allowable Value but is beyond the as-found tolerance band, the channel is OPERABLE, but degraded. The degraded condition of the channel will be evaluated during performance of the SR. This evaluation will consist of resetting the channel setpoint to the NTSP (within the allowed tolerance) and evaluating the channel response. If the channel is functioning as required and expected to pass the next surveillance, then the channel can be restored to service at the completion of the surveillance.

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### APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

A trip setpoint may be set more conservative than the NTSP as necessary in response to plant conditions. However, in this case, the OPERABILITY of this instrument must be verified based on the field setting and not the NTSP. Failure of any instrument renders the affected channel(s) inoperable and reduces the reliability of the affected Functions.

The LCO generally requires OPERABILITY of four or three channels in each instrumentation function and two channels in each logic and manual initiation function. The two-out-of-three and the two-out-of-four configurations allow one channel to be tripped during maintenance or testing without causing an ESFAS initiation. Two logic or manual initiation channels are required to ensure no single random failure disables the ESFAS.

The required channels of ESFAS instrumentation provide unit protection in the event of any of the analyzed accidents. ESFAS protection functions are as follows:

#### 1. Safety Injection

Safety Injection (SI) provides two primary functions:

1. Primary side water addition to ensure maintenance or recovery of reactor vessel water level (coverage of the active fuel for heat removal, clad integrity, and for limiting peak clad temperature to < 2200°F), and
2. Boration to ensure recovery and maintenance of SDM ( $k_{\text{eff}} < 1.0$ ).

These functions are necessary to mitigate the effects of high energy line breaks (HELBs) both inside and outside of containment. The SI signal is also used to initiate other Functions such as:

- Phase A Isolation,
- Containment Ventilation Isolation,
- Reactor Trip,
- ERCW and CCS Pump Start and System Isolation,
- Turbine Trip,
- Feedwater Isolation,

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### APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

- Start of motor driven auxiliary feedwater (AFW) pumps,
- Control room ventilation isolation, and
- Enabling automatic switchover of Emergency Core Cooling Systems (ECCS) suction to containment sump.

These other functions ensure:

- Isolation of nonessential systems through containment penetrations,
- Trip of the turbine and reactor to limit power generation,
- Isolation of main feedwater (MFW) to limit secondary side mass losses,
- Start of AFW to ensure secondary side cooling capability,
- Isolation of the control room to ensure habitability, and
- Enabling ECCS suction from the refueling water storage tank (RWST) switchover on low low RWST level to ensure continued cooling via use of the containment sump.

a. Safety Injection - Manual Initiation

The LCO requires one channel per train to be OPERABLE. The operator can initiate SI at any time by using either of two switches in the control room. This action will cause actuation of all components in the same manner as any of the automatic actuation signals.

The LCO for the Manual Initiation Function ensures the proper amount of redundancy is maintained in the manual ESFAS actuation circuitry to ensure the operator has manual ESFAS initiation capability.

Each channel consists of one hand switch and the interconnecting wiring to the actuation logic cabinet. Each hand switch actuates both trains. This configuration does not allow testing at power.

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

b. Safety Injection - Automatic Actuation Logic and Actuation Relays

This LCO requires two trains to be OPERABLE. Actuation logic consists of all circuitry housed within the actuation subsystems, including the initiating relay contacts responsible for actuating the ESF equipment. The two trains are redundant such that only one is necessary to perform the ESFAS Function.

Manual and automatic initiation of SI must be OPERABLE in MODES 1, 2, and 3. In these MODES, there is sufficient energy in the primary and secondary systems to warrant automatic initiation of ESF systems. Manual Initiation is also required in MODE 4 even though automatic actuation is not required. In this MODE, adequate time is available to manually actuate required components in the event of an abnormal condition or accident, but because of the large number of components actuated on a SI, actuation is simplified by the use of the manual actuation hand switches. Automatic actuation logic and actuation relays must be OPERABLE in MODE 4 to support system level manual initiation.

These Functions are not required to be OPERABLE in MODES 5 and 6 because there is adequate time for the operator to evaluate unit conditions and respond by manually starting individual systems, pumps, and other equipment to mitigate the consequences of an abnormal condition or accident. Unit pressure and temperature are very low and many ESF components are administratively locked out or otherwise prevented from actuating to prevent inadvertent overpressurization of unit systems.

c. Safety Injection - Containment Pressure - High

This signal provides protection against the following accidents:

- SLB inside containment,
- LOCA, and
- Feed line break inside containment.

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### APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Containment Pressure - High provides no input to any control functions. Thus, three OPERABLE channels are sufficient to satisfy protective requirements with a two-out-of-three logic. The transmitters and electronics are located inside the containment annulus, but outside containment, and experience more adverse environmental conditions than if they were located outside containment altogether. However, the environmental effects are less severe than if the transmitters were located inside containment. The NTSP reflects the inclusion of both steady state instrument uncertainties and slightly more adverse environmental instrument uncertainties.

Containment Pressure - High must be OPERABLE in MODES 1, 2, and 3 when there is sufficient energy in the primary and secondary systems to pressurize the containment following a pipe break. In MODES 4, 5, and 6, there is insufficient energy in the primary or secondary systems to pressurize the containment.

d. Safety Injection - Pressurizer Pressure – Low

This signal provides protection against the following accidents:

- Inadvertent opening of a steam generator (SG) relief or safety valve,
- SLB,
- A spectrum of rod cluster control assembly ejection accidents (rod ejection),
- Inadvertent opening of a pressurizer relief or safety valve,
- LOCAs, and
- SG Tube Rupture.

Three protection channels are necessary to satisfy the protective requirements with a two-out-of-three logic.

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### APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The transmitters are located inside containment, with the taps in the vapor space region of the pressurizer, and thus possibly experiencing adverse environmental conditions (LOCA, SLB inside containment, rod ejection). Therefore, the NTSP reflects the inclusion of both steady state and adverse environmental instrument uncertainties.

This Function must be OPERABLE in MODES 1, 2, and 3 (above P-11) to mitigate the consequences of an HELB inside containment. This signal may be manually blocked by the operator below the P-11 setpoint. Automatic SI actuation below this pressure setpoint is then performed by the Containment Pressure - High signal.

This Function is not required to be OPERABLE in MODE 3 below the P-11 setpoint. Other ESF functions are used to detect accident conditions and actuate the ESF systems in this MODE. In MODES 4, 5, and 6, this Function is not needed for accident detection and mitigation.

e. Safety Injection - Steam Line Pressure

(1) Steam Line Pressure – Low

Steam Line Pressure - Low provides protection against the following accidents:

- SLB,
- Feed line break, and
- Inadvertent opening of an SG relief or an SG safety valve.

Steam Line Pressure - Low provides no input to any control functions. Thus, three OPERABLE channels on each steam line are sufficient to satisfy the protective requirements with a two-out-of-three logic on each steam line.

With the transmitters typically located inside the steam valve vaults, it is possible for them to experience adverse environmental conditions during a secondary side break. Therefore, the NTSP reflects both steady state and adverse environmental instrument uncertainties.

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### APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

This Function is anticipatory in nature and has a lead/lag ratio of approximately 50/5.

Steam Line Pressure - Low must be OPERABLE in MODES 1, 2, and 3 (above P-11) when a secondary side break or stuck open valve could result in the rapid depressurization of the steam lines. This signal may be manually blocked by the operator below the P-11 setpoint. Below P-11, feed line break is not a concern. Inside containment SLB will be terminated by automatic SI actuation via Containment Pressure - High, and outside containment SLB will be terminated by the Steam Line Pressure - Negative Rate - High signal for steam line isolation. This Function is not required to be OPERABLE in MODE 4, 5, or 6 because there is insufficient energy in the secondary side of the unit to cause an accident.

#### 2. Containment Spray

Containment Spray functions to lower containment pressure and temperature after an HELB in containment.

This function is necessary to ensure the pressure boundary integrity of the containment structure.

The containment spray actuation signal starts the containment spray pumps and aligns the discharge of the pumps to the containment spray nozzle headers in the upper levels of containment. Water is initially drawn from the RWST by the containment spray pumps. When the RWST reaches the low low level setpoint, the spray pump suction is shifted to the containment sump if continued containment spray is required. Containment spray is actuated manually or automatically by Containment Pressure – High - High.

##### a. Containment Spray - Manual Initiation

The operator can initiate containment spray at any time from the control room by simultaneously turning two Phase B & Containment Ventilation Isolation switches in the same train. Because an inadvertent actuation of containment spray could have such serious consequences, two switches must be turned simultaneously to initiate containment spray. There are two sets of two switches each in the control room. Simultaneously turning the two switches in either set will actuate containment spray in both trains in the same manner as the automatic actuation

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### APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

signal. Two Manual Initiation switches in each train are required to be OPERABLE to ensure no single failure disables the Manual Initiation Function. Note that Manual Initiation of containment spray also actuates Phase B containment isolation but does not close the Main Steam Isolation Valves.

b. Containment Spray - Automatic Actuation Logic and Actuation Relays

Automatic actuation logic and actuation relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b.

Manual and automatic initiation of containment spray must be OPERABLE in MODES 1, 2, and 3 when there is a potential for an accident to occur, and sufficient energy in the primary or secondary systems to pose a threat to containment integrity due to overpressure conditions. Manual initiation is also required in MODE 4, even though automatic actuation is not required. In this MODE, adequate time is available to manually actuate required components in the event of a DBA. However, because of the large number of components actuated on a containment spray, actuation is simplified by the use of the manual actuation switches. Automatic actuation logic and actuation relays must be OPERABLE in MODE 4 to support system level manual initiation. In MODES 5 and 6, there is insufficient energy in the primary and secondary systems to result in containment overpressure. In MODES 5 and 6, there is also adequate time for the operators to evaluate unit conditions and respond, to mitigate the consequences of abnormal conditions by manually starting individual components.

c. Containment Spray - Containment Pressure

This signal provides protection against a LOCA or a SLB inside containment. The transmitters (d/p cells) are located outside of containment with the sensing line (high pressure side of the transmitter) located inside containment. The transmitters and electronics are located inside the containment annulus, but outside containment, and experience more adverse environmental conditions than if they were located outside containment altogether. However, the environmental effects are less severe than if the transmitters were located inside containment. The NTSP reflects the inclusion of both steady

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### APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

state instrument uncertainties and slightly more adverse environmental instrument uncertainties.

This is one of the only Functions that requires the output to energize to perform its required action. It is not desirable to have a loss of power actuate containment spray, since the consequences of an inadvertent actuation of containment spray could be serious. Note that this Function also has the inoperable channel placed in bypass rather than trip to decrease the probability of an inadvertent actuation.

This function uses four channels in a two-out-of-four logic configuration. This arrangement exceeds the minimum redundancy requirements. Additional redundancy is warranted because this Function is energized to trip. Containment Pressure – High - High must be OPERABLE in MODES 1, 2, and 3 when there is sufficient energy in the primary and secondary sides to pressurize the containment following a pipe break. In MODES 4, 5, and 6, there is insufficient energy in the primary and secondary sides to pressurize the containment and reach the Containment Pressure - High - High setpoint.

#### 3. Containment Isolation

Containment Isolation provides isolation of the containment atmosphere, and all process systems that penetrate containment, from the environment. This Function is necessary to prevent or limit the release of radioactivity to the environment in the event of a large break LOCA.

There are two separate Containment Isolation signals, Phase A and Phase B. Phase A isolation isolates all automatically isolable process lines, except component cooling water, essential raw cooling water, and control air, at a relatively low containment pressure indicative of primary or secondary system leaks. For these types of events, forced circulation cooling using the reactor coolant pumps (RCPs) and SGs is the preferred (but not required) method of decay heat removal. Since component cooling water is required to support RCP operation, not isolating component cooling water on the Phase A signal enhances unit safety by allowing operators to use forced RCS circulation to cool the unit. Isolating component cooling water on the Phase A signal may force the use of feed and bleed cooling, which could prove more difficult to control.

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### APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Phase A containment isolation is actuated automatically by SI, or manually via the automatic actuation logic. All process lines penetrating containment, with the exception of component cooling water, essential raw cooling water, and control air, are isolated.

Component cooling water is not isolated at this time to permit continued operation of the RCPs with cooling water flow to the thermal barrier heat exchangers and air or oil coolers. All process lines not equipped with remote operated isolation valves are manually closed, or otherwise isolated, prior to reaching MODE 4.

Manual Phase A Containment Isolation is accomplished by either of two switches in the control room. Either switch actuates both trains. Note that manual actuation of Phase A Containment Isolation also actuates Containment Ventilation Isolation.

The Phase B signal isolates component cooling water, essential raw cooling water, and control air. This occurs at a relatively high containment pressure that is indicative of a large break LOCA or a SLB. For these events, forced circulation using the RCPs is no longer desirable. Isolating the component cooling water at the higher pressure does not pose a challenge to the containment boundary because the Component Cooling Water System is a closed loop inside containment. Although some system components do not meet all of the ASME Code requirements applied to the containment itself, the system is continuously pressurized to a pressure greater than the Phase B setpoint. Thus, routine operation demonstrates the integrity of the system pressure boundary for pressures exceeding the Phase B setpoint. Furthermore, because system pressure exceeds the Phase B setpoint, any system leakage prior to initiation of Phase B isolation would be into containment. Therefore, the combination of Component Cooling Water System design and Phase B isolation ensures the Component Cooling Water System is not a potential path for radioactive release from containment.

Phase B containment isolation is actuated by Containment Pressure – High - High, or manually, via the automatic actuation logic, as previously discussed. For containment pressure to reach a value high enough to actuate Containment Pressure – High - High, a large break LOCA or SLB must have occurred. RCP operation will no longer be required and component cooling water to the RCPs is, therefore, no longer necessary. The RCPs can be operated with seal injection flow alone and without component cooling water flow to the thermal barrier heat exchanger.

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### APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Manual Phase B Containment Isolation is accomplished by the same switches that actuate Containment Spray. When the two switches in either set are turned simultaneously, Phase B Containment Isolation and Containment Spray will be actuated in both trains.

a. Containment Isolation - Phase A Isolation

(1) Phase A Isolation - Manual Initiation

Manual Phase A Containment Isolation is actuated by either of two switches in the control room. Either switch actuates both trains. Note that manual initiation of Phase A Containment Isolation also actuates Containment Ventilation Isolation.

(2) Phase A Isolation - Automatic Actuation Logic and Actuation Relays

Automatic Actuation Logic and Actuation Relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b.

Manual and automatic initiation of Phase A Containment Isolation must be OPERABLE in MODES 1, 2, and 3, when there is a potential for an accident to occur. Manual initiation is also required in MODE 4 even though automatic actuation is not required. In this MODE, adequate time is available to manually actuate required components in the event of an accident, but because of the large number of components actuated on a Phase A Containment Isolation, actuation is simplified by the use of the manual actuation switches. Automatic actuation logic and actuation relays must be OPERABLE in MODE 4 to support system level manual initiation. In MODES 5 and 6, there is insufficient energy in the primary or secondary systems to pressurize the containment to require Phase A Containment Isolation. There also is adequate time for the operator to evaluate unit conditions and manually actuate individual isolation valves in response to abnormal or accident conditions.

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

(3) Phase A Isolation - Safety Injection

Phase A Containment Isolation is also initiated by all Functions that initiate SI. The Phase A Containment Isolation requirements for these Functions are the same as the requirements for their SI function. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead, Function 1, SI, is referenced for all initiating Functions and requirements.

b. Containment Isolation - Phase B Isolation

Phase B Containment Isolation is accomplished by Manual Initiation, Automatic Actuation Logic and Actuation Relays, and by Containment Pressure channels (the same channels that actuate Containment Spray, Function 2). The Containment Pressure trip of Phase B Containment Isolation is energized to trip in order to minimize the potential of spurious trips that may damage the RCPs.

(1) Phase B Isolation - Manual Initiation

The operator can initiate Phase B containment isolation at any time from the control room by simultaneously turning two Phase B & Containment Ventilation Isolation switches in the same train. There are two sets of two switches each in the control room. Simultaneously turning the two switches in either set will actuate Phase B containment isolation in both trains in the same manner as the automatic actuation signal. Two Manual Initiation switches in each train are required to be OPERABLE to ensure no single failure disables the Manual Initiation Function. Note that Manual Initiation of Phase B containment isolation also actuates containment spray and containment vent isolation.

(2) Phase B Isolation - Automatic Actuation Logic and Actuation Relays

Manual and automatic initiation of Phase B containment isolation must be OPERABLE in MODES 1, 2, and 3, when there is a potential for an accident to occur. Manual initiation is also required in MODE 4 even though automatic actuation is not required. In this MODE, adequate time is available to manually actuate required components in the event of an accident. However, because of the large number of components actuated on a Phase B containment

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### APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

isolation, actuation is simplified by the use of the manual actuation hand switches. Automatic actuation logic and actuation relays must be OPERABLE in MODE 4 to support system level manual initiation. In MODES 5 and 6, there is insufficient energy in the primary or secondary systems to pressurize the containment to require Phase B containment isolation. There also is adequate time for the operator to evaluate unit conditions and manually actuate individual isolation valves in response to abnormal or accident conditions.

#### (3) Phase B Isolation - Containment Pressure

The basis for containment pressure MODE applicability is as discussed for ESFAS Function 2.c above.

#### 4. Steam Line Isolation

Isolation of the main steam lines provides protection in the event of a SLB inside or outside containment. Rapid isolation of the steam lines will limit the steam break accident to the blowdown from one SG, at most. For a SLB upstream of the main steam isolation valves (MSIVs), inside or outside of containment, closure of the MSIVs limits the accident to the blowdown from only the affected SG. For a SLB downstream of the MSIVs, closure of the MSIVs terminates the accident as soon as the steam lines depressurize. Steam Line Isolation also mitigates the effects of a feed line break and ensures a source of steam for the turbine driven AFW pump during a feed line break.

##### a. Steam Line Isolation - Manual Initiation

Manual initiation of Steam Line Isolation can be accomplished from the control room. There are four switches in the control room and each switch initiates action to immediately close its associated MSIV. The LCO requires one channel per steam line to be OPERABLE.

##### b. Steam Line Isolation - Automatic Actuation Logic and Actuation Relays

Automatic actuation logic and actuation relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b.

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Manual and automatic initiation of steam line isolation must be OPERABLE in MODES 1, 2, and 3 when there is sufficient energy in the RCS and SGs to have a SLB or other accident. This could result in the release of significant quantities of energy and cause a cooldown of the primary system. The Steam Line Isolation Function is required in MODES 2 and 3 unless all MSIVs are closed. In MODES 4, 5, and 6, there is insufficient energy in the RCS and SGs to experience a SLB or other accident releasing significant quantities of energy.

c. Steam Line Isolation - Containment Pressure - High - High

This Function actuates closure of the MSIVs in the event of a LOCA or a SLB inside containment to maintain at least one unfaulted SG as a heat sink for the reactor, and to limit the mass and energy release to containment. The transmitters (d/p cells) are located outside containment with the sensing line (high pressure side of the transmitter) located inside containment. Containment Pressure – High - High provides no input to any control functions. Thus, three OPERABLE channels are sufficient to satisfy protective requirements with two-out-of-three logic. However, for enhanced reliability, this Function was designed with four channels and a two-out-of-four logic.

The transmitters and electronics are located inside the containment annulus, but outside containment, and experience more adverse environmental conditions than if they were located outside containment altogether. However, the environmental effects are less severe than if the transmitters were located inside containment. The NTSP reflects the inclusion of both steady state instrument uncertainties and slightly more adverse environmental instrument uncertainties.

Containment Pressure – High - High must be OPERABLE in MODES 1, 2, and 3, when there is sufficient energy in the primary and secondary side to pressurize the containment following a pipe break. This would cause a significant increase in the containment pressure, thus allowing detection and closure of the MSIVs. The Steam Line Isolation Function remains OPERABLE in MODES 2 and 3 unless all MSIVs are closed. In MODES 4, 5, and 6, there is not enough energy in the primary and secondary sides to pressurize the containment to the Containment Pressure – High - High setpoint.

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

d. Steam Line Isolation - Steam Line Pressure

(1) Steam Line Pressure – Low

Steam Line Pressure - Low provides closure of the MSIVs in the event of a SLB to maintain at least one unfaulted SG as a heat sink for the reactor, and to limit the mass and energy release to containment. This Function provides closure of the MSIVs in the event of a feed line break to ensure a supply of steam for the turbine driven AFW pump. Steam Line Pressure - Low was discussed previously under SI Function 1.e.1.

Steam Line Pressure - Low Function must be OPERABLE in MODES 1, 2, and 3 (when the Steam Line Isolation on Steam Line Pressure, Negative Rate-High is blocked), with any main steam valve open, when a secondary side break or stuck open valve could result in the rapid depressurization of the steam lines. This signal may be manually blocked by the operator below the P-11 setpoint. Below P-11, an inside containment SLB will be terminated by automatic actuation via Containment Pressure - High - High. Stuck valve transients and outside containment SLBs will be terminated by the Steam Line Pressure - Negative Rate - High signal for Steam Line Isolation below P-11 when SI has been manually blocked. The Steam Line Isolation Function is required in MODES 2 and 3 unless all MSIVs are closed. This Function is not required to be OPERABLE in MODES 4, 5, and 6 because there is insufficient energy in the secondary side of the unit to have an accident.

(2) Steam Line Pressure - Negative Rate – High

Steam Line Pressure - Negative Rate - High provides closure of the MSIVs for a SLB when less than the P-11 setpoint, to maintain at least one unfaulted SG as a heat sink for the reactor, and to limit the mass and energy release to containment. When the operator manually blocks the Steam Line Pressure - Low main steam isolation signal when less than the P-11 setpoint, the Steam Line Pressure - Negative Rate - High signal is automatically enabled. Steam Line Pressure - Negative Rate - High provides no input to any control functions. Thus, three OPERABLE channels are sufficient to satisfy requirements with a two-out-of-three logic on each steam line.

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### APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Steam Line Pressure - Negative Rate - High must be OPERABLE in MODE 3 when less than the P-11 setpoint, and the Steam Line Isolation on Steam Line Pressure, Low is blocked, when a secondary side break or stuck open valve could result in the rapid depressurization of the steam line(s). In MODES 1 and 2, and in MODE 3, when above the P-11 setpoint, this signal is automatically disabled and the Steam Line Pressure - Low signal is automatically enabled. The Steam Line Isolation Function is required to be OPERABLE in MODES 2 and 3 unless all MSIVs are closed. In MODES 4, 5, and 6, there is insufficient energy in the primary and secondary sides to have a SLB or other accident that would result in a release of significant enough quantities of energy to cause a cooldown of the RCS.

While the transmitters may experience elevated ambient temperatures due to a SLB, the trip function is based on rate of change, not the absolute accuracy of the indicated steam pressure. Therefore, the NTSP reflects only steady state instrument uncertainties.

#### 5. Turbine Trip and Feedwater Isolation

The primary functions of the Turbine Trip and Feedwater Isolation signals are to prevent damage to the turbine due to water in the steam lines, and to stop the excessive flow of feedwater into the SGs. These Functions are necessary to mitigate the effects of a high water level in the SGs, which could result in carryover of water into the steam lines and excessive cooldown of the primary system. The SG high water level is due to excessive feedwater flows.

The Function is actuated when the level in any SG exceeds the high high setpoint, and performs the following functions:

- Trips the main turbine,
- Trips the MFW pumps,
- Initiates feedwater isolation, and
- Shuts the MFW regulating valves and the bypass feedwater regulating valves.

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

This Function is actuated by SG Water Level - High High, or by a SI signal. The nominal trip setpoint and allowable value limits are a percentage of the narrow range instrument span for each steam generator. The RTS also initiates a turbine trip signal whenever a reactor trip (P-4) is generated. In the event of SI, the unit is taken off line and the turbine generator must be tripped. The MFW System is also taken out of operation and the AFW System is automatically started. The SI signal was discussed previously.

a. Turbine Trip and Feedwater Isolation - Automatic Actuation Logic and Actuation Relays

Automatic Actuation Logic and Actuation Relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b.

b. Turbine Trip and Feedwater Isolation - Steam Generator Water Level - High High (P-14)

This signal provides protection against excessive feedwater flow. The ESFAS SG water level instruments provide input to the SG Water Level Control System. Therefore, the actuation logic must be able to withstand both an input failure to the control system (which may then require the protection function actuation) and a single failure in the other channels providing the protection function actuation. Only three protection channels are necessary, with a two-out-of-three logic, to satisfy the protective requirements because a median signal selector is provided.

The transmitters (d/p cells) are located inside containment. However, the events that this Function protects against cannot cause a severe environment in containment. Therefore, the NTSP reflects only steady state instrument uncertainties.

c. Turbine Trip and Feedwater Isolation - Safety Injection

Turbine Trip and Feedwater Isolation is also initiated by all Functions that initiate SI. The Feedwater Isolation Function requirements for these Functions are the same as the requirements for their SI function. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead Function 1, SI, is referenced for all initiating functions and requirements.

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Turbine Trip and Feedwater Isolation Functions must be OPERABLE in MODES 1 and 2 and 3 except when all MFIVs, MFRVs, and associated MFRV bypass valves are closed or isolated by a closed manual valve when the MFW System is in operation and the turbine generator may be in operation. In MODES 4, 5, and 6, the MFW System and the turbine generator are not in service and this Function is not required to be OPERABLE.

6. Auxiliary Feedwater

The AFW System is designed to provide a secondary side heat sink for the reactor in the event that the MFW System is not available. The system has two motor driven pumps and a turbine driven pump, making it available during normal unit operation, during a loss of AC power, a loss of MFW, and during a Feedwater System pipe break. The normal source of water for the AFW System is the condensate storage tank (CST) (not safety related). A low pressure in the AFW suction line will automatically realign the pump suctions to the Essential Raw Cooling Water (ERCW) System (safety related).

a. Auxiliary Feedwater - Automatic Actuation Logic and Actuation Relays (Solid State Protection System)

Automatic actuation logic and actuation relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b.

b. Auxiliary Feedwater - Steam Generator Water Level - Low Low

SG Water Level - Low Low provides protection against a loss of heat sink due to a feed line break outside of containment, or a loss of MFW, which results in a loss of SG water level. SG Water Level - Low Low provides input to the SG Level Control System. Therefore, the actuation logic must be able to withstand both an input failure to the control system which may then require a protection function actuation and a single failure in the other channels providing the protection function actuation. Only three protection channels, with a two-out-of-three logic, are necessary to satisfy the protective requirements because a median signal selector is provided.

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### APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

With the transmitters located inside containment and thus possibly experiencing adverse environmental conditions (due to a feedline break), the Environmental Allowance Modifier (EAM) was devised. The EAM function (Containment Pressure (EAM) with a setpoint of < 0.5 psig) senses the presence of adverse containment conditions (elevated pressure) and enables the Steam Generator Water Level - Low-Low setpoint (Adverse) which reflects the increased transmitter uncertainties due to this environment. The EAM allows the use of a lower Steam Generator Water Level - Low-Low (EAM) setpoint when these conditions are not present, thus allowing more margin for normal operating conditions. Additionally, the NTSP reflects the inclusion of both steady state and adverse environmental instrument uncertainties.

The Trip Time Delay (TTD) creates additional operational margin when the plant needs it most, during early escalation to power, by allowing the operator time to recover level when the primary side load is sufficiently small to allow such action. The TTD is based on continuous monitoring of primary side power through the use of RCS loop  $\Delta T$ . Two time delays are calculated, based on the number of steam generators indicating less than the Low-Low Level setpoint and the primary side power level. The magnitude of the delays decreases with increasing primary side power level, up to 50% RTP. Above 50% RTP there are no time delays for the Low-Low level trips.

In the event of failure of a Steam Generator Water Level channel, it is placed in the trip condition as input to the Solid State Protection System and does not affect either the EAM or TTD setpoint calculations for the remaining OPERABLE channels. Failure of the Containment Pressure (EAM) channel to a protection set also does not affect the EAM setpoint calculations. This results in the requirement that the operator adjust the affected Steam Generator Water Level - Low-Low (EAM) trip setpoints to the same value as the Steam Generator Water Level - Low-Low (Adverse) trip setpoints or actuate the SG Water Level Low-Low setpoint. Failure of the RCS loop  $\Delta T$  channel input (failure of more than one  $T_H$  resistance temperature detectors (RTD) or failure of a  $T_C$  RTD) does not affect the TTD calculation for a protection set. This results in the requirement that the operator adjust the threshold power level for zero seconds time delay from 50% RTP to 0% RTP, through the man-machine-interface (MMI) test cart.

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

There are three Steam Generator Water Level Low-Low channels per steam generator arranged in a two-out-of-three logic. These channels are arranged in four protection sets with each channel of the Containment Pressure (EAM) and RCS Loop  $\Delta T$  inputting into its associated protection set.

With the transmitters (d/p cells) located inside containment and the accidents the channel provides protection for occurring outside containment, the NTSP reflects only steady state instrument uncertainties. Because the transmitters (d/p cells) are located inside containment, thus possibly experiencing adverse environmental conditions during a feed line break inside containment, the SG Water Level-Low Low Trip Setpoint may not have sufficient margin to account for adverse environmental instrument uncertainties; in this case, AFW pump start will be provided by a Containment Pressure-High SI signal.

c. Auxiliary Feedwater - Safety Injection

A SI signal starts the motor driven and turbine driven AFW pumps. The AFW initiation functions are the same as the requirements for their SI function. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead, Function 1, SI, is referenced for all initiating functions and requirements.

d. Auxiliary Feedwater - Loss of Offsite Power

A loss of offsite power to the 6.9 kV Unit-boards (RCP buses) will be accompanied by a loss of reactor coolant pumping power and the subsequent need for some method of decay heat removal. The AFW loss of offsite power is detected by a voltage drop on each 6.9 kV shutdown board. Loss of power to either 6.9 kV shutdown board will start the turbine driven AFW pump to ensure that at least one SG contains enough water to serve as the heat sink for reactor decay heat and sensible heat removal following the reactor trip.

The loss-of-voltage relaying on the 6.9 kV shutdown board uses three solid-state voltage sensors in a two-out-of-three voltage sensor logic (27T-S1A, S1B, & S1C) for loss-of-power detection. A two-out-of-three logic from the voltage sensor channels energizes two parallel separate timing relays with a one-out-of-two logic scheme (LV1 and LV2). These voltage sensors and timing relays provide emergency diesel generator start, load-shed initiation, and subsequent turbine driven auxiliary feedwater

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### APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

(TDAFW) pump start through separate blackout relays (BOX and BOY).

A footnote has been added to clarify that this requirement only applies to shutdown board instrumentation on the same unit. This clarification removes the potential to declare the AFW loss-of-power start instrumentation inoperable for a given unit when only the opposite unit's instrumentation is inoperable.

The AFW turbine-driven pump is considered OPERABLE when one train of the AFW loss of power start function is declared inoperable, in accordance with technical specifications, because both 6.9 kilovolt shutdown board logic trains supply this function.

Functions 6.a through 6.d must be OPERABLE in MODES 1, 2, and 3 to ensure that the SGs remain the heat sink for the reactor. SG Water Level - Low Low in any operating SG will cause the motor driven AFW pumps to start. SG Water Level - Low Low in any two operating SGs will cause the turbine driven pump to start. These Functions do not have to be OPERABLE in MODES 5 and 6 because there is not enough heat being generated in the reactor to require the SGs as a heat sink. In MODE 4, AFW actuation does not need to be OPERABLE because either AFW or residual heat removal (RHR) will already be in operation to remove decay heat or sufficient time is available to manually place either system in operation.

e. Auxiliary Feedwater - Trip of All Main Feedwater Pumps

A Trip of all MFW pumps is an indication of a loss of MFW and the subsequent need for some method of decay heat and sensible heat removal to bring the reactor back to no load temperature and pressure. A turbine driven MFW pump is equipped with one pressure switch on the control oil line for the speed control system. A low pressure signal from this pressure switch indicates a trip of that pump. A trip of all MFW pumps starts the motor driven and turbine driven AFW pumps to ensure that at least one SG is available with water to act as the heat sink for the reactor.

This Function includes a footnote indicating that MODE 2 applicability is limited to operation when one or more MFW pumps are supplying feedwater to the steam generators (SGs).

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### APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The footnote limits the Applicability to require the auto-start logic to be operable in MODE 2 only when at least one MFW pump is in service supplying feedwater to the SGs. Because plant conditions at the time of entry into Mode 2 do not allow the MFW pumps to operate, without this footnote the channels would need to be tripped resulting in an AFW start signal, starting the turbine-driven pump in addition to the motor-driven AFW pumps, which is an undesirable situation. This resolves a conflict between the MODE applicability and plant design, which does not support MFW pump operation at the time of entry into MODE 2. Also, modifying the requirement for auto-start of the AFW pumps to be only required when the MFW pumps are in service limits the potential for inadvertent AFW actuations during normal plant startups and shutdowns that could lead to reactivity control issues due to over cooling transients.

Function 6.e must be OPERABLE in MODES 1 and 2. This ensures that at least one SG is provided with water to serve as the heat sink to remove reactor decay heat and sensible heat in the event of an accident. In MODES 3, 4, and 5, the MFW pumps may be normally shut down, and thus neither pump trip is indicative of a condition requiring automatic AFW initiation.

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

f. Auxiliary Feedwater - Pump Suction Transfer on Suction Pressure – Low

A low pressure signal in the AFW pump suction line protects the AFW pumps against a loss of the normal supply of water for the pumps, the CST. Three pressure switches are located on the AFW pump suction line from the CST. A low pressure signal sensed by two of three switches will cause the emergency supply of water for the pump to be aligned to the emergency source of water. ERCW (safety grade) is then lined up to supply the AFW pumps to ensure an adequate supply of water for the AFW System to maintain at least one of the SGs as the heat sink for reactor decay heat and sensible heat removal.

Since the detectors are located in an area not affected by HELBs or high radiation, they will not experience any adverse environmental conditions however the NTSP reflects both steady state and adverse environmental instrument uncertainties.

This Function must be OPERABLE in MODES 1, 2, and 3 to ensure a safety grade supply of water for the AFW System to maintain the SGs as the heat sink for the reactor. This Function does not have to be OPERABLE in MODES 5 and 6 because there is not enough heat being generated in the reactor to require the SGs as a heat sink. In MODE 4, AFW automatic suction transfer does not need to be OPERABLE because RHR will already be in operation, or sufficient time is available to place RHR in operation, to remove decay heat.

g. Auxiliary Feedwater Suction Transfer Time Delays

A low pressure signal in the AFW pump suction line protects the AFW pumps against a loss of the normal supply of water for the pumps, the CST. The pressure switch setpoints and the logic time delays for the AFW pump suction switchover were determined to ensure that adequate net positive suction head (NPSH) for the AFW pumps is maintained during the pump suction transfer sequence.

The available NPSH for the pumps is calculated assuming a water level in the supply header that would not be reached until after the time delays are exceeded, even when accounting for the two TDAFW timers in series. The TDAFW pump has two timers because this pump can be switched to either of the two trains in the ECRW system: one timer is for the transfer to one of

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

the two trains. The timers operate in sequence to assure that the TDAFW pump is transferred to one of the ERCW trains.

This Function must be OPERABLE in MODES 1, 2, and 3 to ensure a safety grade supply of water for the AFW System to maintain the SGs as the heat sink for the reactor. This Function does not have to be OPERABLE in MODES 5 and 6 because there is not enough heat being generated in the reactor to require the SGs as a heat sink. In MODE 4, AFW automatic suction transfer does not need to be OPERABLE because RHR will already be in operation, or sufficient time is available to place RHR in operation, to remove decay heat.

7. Automatic Switchover to Containment Sump

At the end of the injection phase of a LOCA, the RWST will be nearly empty. Continued cooling must be provided by the ECCS to remove decay heat. The source of water for the ECCS pumps is automatically switched to the containment recirculation sump. The low head residual heat removal (RHR) pumps and containment spray pumps draw the water from the containment recirculation sump, the RHR pumps pump the water through the RHR heat exchanger, inject the water back into the RCS, and supply the cooled water to the other ECCS pumps. Switchover from the RWST to the containment sump must occur before the RWST empties to prevent damage to the RHR pumps and a loss of core cooling capability. For similar reasons, switchover must not occur before there is sufficient water in the containment sump to support ESF pump suction. Furthermore, early switchover must not occur to ensure that sufficient borated water is injected from the RWST. This ensures the reactor remains shut down in the recirculation mode.

a. Automatic Switchover to Containment Sump - Automatic Actuation Logic and Actuation Relays

Automatic actuation logic and actuation relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b.

b. Automatic Switchover to Containment Sump - Refueling Water Storage Tank (RWST) Level - Low Coincident With Safety Injection and Coincident With Containment Sump Level – High

During the injection phase of a LOCA, the RWST is the source of water for all ECCS pumps. A low level in the RWST coincident with a SI signal provides protection against a loss of water for the

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### APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

ECCS pumps and indicates the end of the injection phase of the LOCA. The RWST is equipped with four level transmitters. These transmitters provide no control functions. Therefore, a two-out-of-four logic is adequate to initiate the protection function actuation. Although only three channels would be sufficient, a fourth channel has been added for increased reliability.

The RWST - Low Allowable Value has both upper and lower limits. The lower limit is selected to ensure switchover occurs before the RWST empties, to prevent ECCS pump damage. The upper limit is selected to ensure containment sump tall strainer submergence.

The RSWT level transmitters are located in an area not affected by HELBs or post accident high radiation. Thus, they will not experience any adverse environmental conditions and the NTSP reflects only steady state instrument uncertainties.

Automatic switchover occurs only if the RWST low level signal is coincident with SI. This prevents accidental switchover during normal operation. Accidental switchover could damage ECCS pumps if they are attempting to take suction from an empty sump. The automatic switchover Function requirements for the SI Functions are the same as the requirements for their SI function. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead, Function 1, SI, is referenced for all initiating Functions and requirements.

Additional protection from spurious switchover is provided by requiring a Containment Sump Level - High signal as well as RWST Level - Low and SI. This ensures sufficient water is available in containment to support the recirculation phase of the accident. A Containment Sump Level - High signal must be present, in addition to the SI signal and the RWST Level - Low signal, to transfer the suctions of the RHR pumps to the containment sump. The containment sump is equipped with four level transmitters. These transmitters provide no control functions. Therefore, a two-out-of-four logic is adequate to initiate the protection function actuation. Although only three channels would be sufficient, a fourth channel has been added for increased reliability. The containment sump level Trip Setpoint/Allowable Value is selected to ensure that automatic switchover is permitted before RWST level decreases below the RWST Level - Low setpoint. This ensures an adequate suction supply to the ECCS pumps by allowing sufficient time for completion of the switchover before vortexing occurs in the

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### APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

RWST. The transmitters are located inside containment and thus possibly experience adverse environmental conditions. Therefore, the NTSP reflects the inclusion of both steady state and environmental instrument uncertainties.

These Functions must be OPERABLE in MODES 1, 2, 3, and 4 when there is a potential for a LOCA to occur, to ensure a continued supply of water for the ECCS pumps. These Functions are not required to be OPERABLE in MODES 5 and 6 because there is adequate time for the operator to evaluate unit conditions and respond by manually starting systems, pumps, and other equipment to mitigate the consequences of an abnormal condition or accident. System pressure and temperature are very low and many ESF components are administratively locked out or otherwise prevented from actuating to prevent inadvertent overpressurization of unit systems.

#### 8. Engineered Safety Feature Actuation System Interlocks

To allow some flexibility in unit operations, several interlocks are included as part of the ESFAS. These interlocks permit the operator to block some signals, automatically enable other signals, prevent some actions from occurring, and cause other actions to occur. The interlock Functions back up manual actions to ensure bypassable functions are in operation under the conditions assumed in the safety analyses.

##### a. Engineered Safety Feature Actuation System Interlocks - Reactor Trip, P-4

The P-4 interlock is enabled when a reactor trip breaker and its associated bypass breaker is open. Once the P-4 interlock is enabled, automatic SI initiation may be blocked after a 60 second time delay. This Function allows operators to take manual control of SI systems after the initial phase of injection is complete. Once SI is blocked, automatic actuation of SI cannot occur until the reactor trip breakers have been manually closed. The functions of the P-4 interlock are:

- Trip the main turbine,
- Isolate MFW with coincident low  $T_{avg}$ ,
- Prevent automatic reactivation of SI after a manual reset of SI, and

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

- Prevent opening of the MFW isolation valves if they were closed on SI or SG Water Level - High High.

Each of the above Functions is interlocked with P-4 to avert or reduce the continued cooldown of the RCS following a reactor trip. An excessive cooldown of the RCS following a reactor trip could cause an insertion of positive reactivity with a subsequent increase in generated power. To avoid such a situation, the noted Functions have been interlocked with P-4 as part of the design of the unit control and protection system. There are two P-4 channels arranged in a one-out-of-one logic per channel.

None of the noted Functions serves a mitigation function in the unit licensing basis safety analyses. Only the turbine trip Function is explicitly assumed since it is an immediate consequence of the reactor trip Function. Neither turbine trip, nor any of the other three Functions associated with the reactor trip signal, is required to show that the unit licensing basis safety analysis acceptance criteria are not exceeded.

The reactor trip breaker position switches provide input to the P-4 interlock to indicate open or close. Therefore, this Function has no adjustable trip setpoint with which to associate a NTSP and Allowable Value.

This Function must be OPERABLE in MODES 1, 2, and 3 when the reactor may be critical or approaching criticality. This Function does not have to be OPERABLE in MODE 4, 5, or 6 because the main turbine, and the MFW System are not in operation.

b. Engineered Safety Feature Actuation System Interlocks - Pressurizer Pressure, P-11

The P-11 interlock permits a normal unit cooldown and depressurization without actuation of SI or main steam line isolation. With two-out-of-three pressurizer pressure channels (discussed previously) less than the P-11 setpoint, the operator can manually block the Pressurizer Pressure - Low and Steam Line Pressure - Low SI signals and the Steam Line Pressure - Low steam line isolation signal (previously discussed). When the Steam Line Pressure - Low steam line isolation signal is manually blocked, a main steam isolation signal on Steam Line Pressure - Negative Rate - High is enabled. This provides protection for a SLB by closure of the MSIVs. With two-out-of-three pressurizer pressure channels above the P-11 setpoint, the

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### APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Pressurizer Pressure - Low and Steam Line Pressure - Low SI signals and the Steam Line Pressure - Low steam line isolation signal are automatically enabled, and the main steam isolation on Steam Line Pressure - Negative Rate - High is disabled. The NTSP reflects only steady state instrument uncertainties.

This Function must be OPERABLE in MODES 1, 2, and 3 to allow an orderly cooldown and depressurization of the unit without the actuation of SI or main steam isolation. This Function does not have to be OPERABLE in MODE 4, 5, or 6 because system pressure must already be below the P-11 setpoint for the requirements of the heatup and cooldown curves to be met.

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

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## ACTIONS

A Note has been added in the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed on Table 3.3.2-1.

In the event a channel's NTSP is found nonconservative with respect to the Allowable Value, or the channel is not functioning as required, or the transmitter, instrument Loop, signal processing electronics, setpoint comparator output, contact output, or bistable is found inoperable, then all affected Functions provided by that channel must be declared inoperable and the LCO Condition(s) entered for the protection Function(s) affected. When the Required Channels in Table 3.3.2-1 are specified on a "per" basis (e.g., on a per steam line, per loop, per SG, etc., basis), then the Condition may be entered separately for each steam line, loop, SG, etc., as appropriate.

When the number of inoperable channels in a trip function exceed those specified in one or other related Conditions associated with a trip function, then the unit is outside the safety analysis. Therefore, LCO 3.0.3 should be immediately entered if applicable in the current MODE of operation.

### A.1

Condition A applies to all ESFAS protection functions.

Condition A addresses the situation where one or more channels or trains for one or more Functions are inoperable at the same time. The Required Action is to refer to Table 3.3.2-1 and to take the Required Actions for the protection functions affected. The Completion Times are those from the referenced Conditions and Required Actions.

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### ACTIONS (continued)

#### B.1, B.2.1, and B.2.2

Condition B applies to manual initiation of:

- SI,
- Containment Spray,
- Phase A Isolation, and
- Phase B Isolation.

This action addresses the train orientation of the SSPS for the functions listed above. If a channel or train is inoperable, 48 hours is allowed to return it to an OPERABLE status. Note that for containment spray and Phase B isolation, failure of one or both channels in one train renders the train inoperable. Condition B, therefore, encompasses both situations. The specified Completion Time is reasonable considering that there are two automatic actuation trains and another manual initiation train OPERABLE for each Function, and the low probability of an event occurring during this interval. If the train cannot be restored to OPERABLE status, the unit must be placed in a MODE in which the LCO does not apply. This is done by placing the unit in at least MODE 3 within an additional 6 hours (54 hours total time) and in MODE 5 within an additional 30 hours (84 hours total time). The allowable 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

Condition C applies to the automatic actuation logic and actuation relays for the following functions:

- SI,
- Containment Spray,
- Phase A Isolation, and
- Phase B Isolation.

## BASES

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### ACTIONS (continued)

This action addresses the train orientation of the SSPS and the master and slave relays. If one train is inoperable, 24 hours are allowed to restore the train to OPERABLE status. The 24 hours allowed for restoring the inoperable train to OPERABLE status is justified in Reference 9. The specified Completion Time is reasonable considering that there is another train OPERABLE, and the low probability of an event occurring during this interval. If the train cannot be restored to OPERABLE status, the unit must be placed in a MODE in which the LCO does not apply. This is done by placing the unit in at least MODE 3 within an additional 6 hours (30 hours total time) and in MODE 5 within an additional 30 hours (60 hours total time). The Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

The Required Actions are modified by a Note that allows one train to be bypassed for up to 4 hours for surveillance testing, provided the other train is OPERABLE. This allowance is based on the reliability analysis assumption of WCAP-10271-P-A (Ref. 10) that 4 hours is the average time required to perform train surveillance.

#### D.1, D.2.1, and D.2.2

Condition D applies to:

- Containment Pressure - High,
- Pressurizer Pressure – Low,
- Steam Line Pressure - Low,
- Steam Line Pressure - Negative Rate - High, and
- SG Water level - High High (P-14).

If one channel is inoperable, 72 hours are allowed to restore the channel to OPERABLE status or to place it in the tripped condition. Generally this Condition applies to functions that operate on two-out-of-three logic. Therefore, failure of one channel places the Function in a two-out-of-two configuration. One channel must be tripped to place the Function in a one-out-of-two configuration that satisfies redundancy requirements. The 72 hours allowed to restore the channel to OPERABLE status or to place it in the tripped condition is justified in Reference 9.

## BASES

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### ACTIONS (continued)

Failure to restore the inoperable channel to OPERABLE status or place it in the tripped condition within 72 hours requires the unit be placed in MODE 3 within the following 6 hours and MODE 4 within the next 6 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. In MODE 4, these Functions are no longer required OPERABLE.

The Required Actions are modified by a Note that allows the inoperable channel to be bypassed for up to 12 hours for surveillance testing of other channels. The 12 hours allowed for testing, are justified in Reference 9.

#### E.1, E.2.1, and E.2.2

Condition E applies to:

- Containment Spray Containment Pressure - High - High,
- Containment Phase B Isolation Containment Pressure – High - High, and
- Steam Line Isolation Containment Pressure – High – High.

None of these signals has input to a control function. Thus, two-out-of-three logic is necessary to meet acceptable protective requirements. However, a two-out-of-three design would require tripping a failed channel. This is undesirable because a single failure would then cause spurious containment spray initiation. Spurious spray actuation is undesirable because of the cleanup problems presented. Therefore, these channels are designed with two-out-of-four logic so that a failed channel may be bypassed rather than tripped. Note that one channel may be bypassed and still satisfy the single failure criterion. Furthermore, with one channel bypassed, a single instrumentation channel failure will not spuriously initiate containment spray.

To avoid the inadvertent actuation of containment spray and Phase B containment isolation, the inoperable channel should not be placed in the tripped condition. Instead it is bypassed. Restoring the channel to OPERABLE status, or placing the inoperable channel in the bypass condition within 72 hours, is sufficient to assure that the Function remains OPERABLE and minimizes the time that the Function may be in a partial trip condition (assuming the inoperable channel has failed high). The Completion Time is further justified based on the low probability of an

## BASES

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### ACTIONS (continued)

event occurring during this interval. Failure to restore the inoperable channel to OPERABLE status, or place it in the bypassed condition within 72 hours, requires the unit be placed in MODE 3 within the following 6 hours and MODE 4 within the next 6 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. In MODE 4, these Functions are no longer required OPERABLE.

The Required Actions are modified by a Note that allows one additional channel to be bypassed for up to 12 hours for surveillance testing. Placing a second channel in the bypass condition for up to 12 hours for testing purposes is acceptable based on the results of Reference 9.

#### F.1 and F.2

Condition F applies to the Steam Line Isolation, Manual Initiation ESFAS Function.

If a train or channel is inoperable, 48 hours is allowed to return it to OPERABLE status. The specified Completion Time is reasonable considering the nature of this Function, the available redundancy, and the low probability of an event occurring during this interval. If the Function cannot be returned to OPERABLE status, the associate MSIV is declared inoperable and the associated Required Actions followed for an inoperable MSIV.

#### G.1, G.2.1, and G.2.2

Condition G applies to the P-4 interlock.

For the P-4 Interlock Function, this action addresses the train orientation of the SSPS. If a train or channel is inoperable, 48 hours is allowed to return it to OPERABLE status. The specified Completion Time is reasonable considering the nature of the Function, the available redundancy, and the low probability of an event occurring during this interval. If the Function cannot be returned to OPERABLE status, the unit must be placed in MODE 3 within the next 6 hours and MODE 4 within the following 6 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. In MODE 4, the unit does not have any analyzed transients or conditions that require the explicit use of the protection function noted above.

## BASES

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### ACTIONS (continued)

#### H.1, H.2.1, and H.2.2

Condition H applies to the automatic actuation logic and actuation relays for the Steam Line Isolation Turbine Trip and Feedwater Isolation, and AFW actuation Functions.

The action addresses the train orientation of the SSPS and the master and slave relays for these functions. If one train is inoperable, 24 hours are allowed to restore the train to OPERABLE status. The 24 hours allowed for restoring the inoperable train to OPERABLE status is justified in Reference 9. The Completion Time for restoring a train to OPERABLE status is reasonable considering that there is another train OPERABLE, and the low probability of an event occurring during this interval. If the train cannot be returned to OPERABLE status, the unit must be brought to MODE 3 within the next 6 hours and MODE 4 within the following 6 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. Placing the unit in MODE 4 removes all requirements for OPERABILITY of the protection channels and actuation functions. In this MODE, the unit does not have analyzed transients or conditions that require the explicit use of the protection functions noted above.

The Required Actions are modified by a Note that allows one train to be bypassed for up to 4 hours for surveillance testing provided the other train is OPERABLE. This allowance is based on the reliability analysis (Ref. 10) assumption that 4 hours is the average time required to perform channel surveillance.

#### I.1 and I.2

Condition I applies to the following ESFAS Functions:

- Steam Generator Water Level--Low-Low (Adverse), and
- Steam Generator Water Level--Low-Low (EAM)

A known inoperable channel must be placed in the tripped condition within 6 hours. Placing the channel in the tripped condition results in a partial trip condition requiring only one-out-of-two logic for actuation of the two-out-of-three trips.

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### ACTIONS (continued)

In addition to placing the channel in the tripped condition, it is necessary to force the use of the shorter TTD by adjustment of the single steam generator time delay calculation ( $T_S$ ) to match the multiple steam generator time delay calculation ( $T_M$ ) for the affected protection set within 4 hours.

The Required Actions have been modified by a Note that allows placing the inoperable channel in the bypassed condition for up to 4 hours while performing routine surveillance testing of the other channels.

#### J.1, J.2, J.3.1, and J.3.2

Condition J applies to the Containment Pressure (EAM) coincidence with Steam Generator Water Level – Low-Low (Adverse) ESFAS Function.

Failure of the Containment Pressure (EAM) channel to a protection set does not affect the EAM setpoint calculations. A known inoperable Containment Pressure channel results in the requirement to adjust the affected Steam Generator Water Level – Low-Low (EAM) trip setpoints for the affected protection set to the same value as the Steam Generator Water Level – Low-Low (Adverse) trip setpoint within 6 hours.

An alternative to adjusting the affected Steam Generator Water Level - Low-Low (EAM) trip setpoints to the same value as the Steam Generator Water Level – Low-Low (Adverse) trip setpoints is to place the associated protection set's SG Water Level – Low-Low channels in the tripped condition within 6 hours.

If neither of the above Required Actions are completed within their associated Completion Time, then the unit must be placed in a MODE where these Functions are not required OPERABLE. This requires the unit be placed in MODE 3 within 12 hours and MODE 4 within 18 hours. The allowed Completion Times are reasonable to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. In MODE 4, these Functions are no longer required OPERABLE.

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### ACTIONS (continued)

#### K.1, K.2, K.3.1, and K.3.2

Condition K applies to the RCS Loop  $\Delta T$  coincidence with SG Water Level – Low-Low.

Failure of the RCS loop  $\Delta T$  channel input (failure of more than one  $T_H$  RTD or failure of a  $T_C$  RTD) does not affect the TTD calculation for a protection set. This results in the requirement that the operator adjust the threshold power level for zero seconds time delay from 50% RTP to 0% RTP within 6 hours. With the trip time delay adjusted to zero seconds the additional operational margin that allows the operator time to recover SG level is removed.

An alternative to adjusting the threshold power level for zero seconds time delay is to place the affected protection set's SG Water Level Low-Low level channels in the tripped condition within 6 hours.

If neither of the above Required Actions can be completed within their associated Completion Times then the unit must be placed in a MODE where these Functions are not required OPERABLE. This requires the unit be placed in MODE 3 within 12 hours and MODE 4 within 18 hours. The allowed Completion Times are reasonable to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. In MODE 4, these Functions are no longer required OPERABLE.

#### L.1 and L.2

Condition L applies to the Loss of Voltage sensors associated with the Loss of Power AFW pump start ESFAS Function. These are the same sensors for the DG loss of Voltage start.

This function is provided by voltage sensors for each train arranged in a two-out-of-three logic scheme. If a sensor is inoperable, 6 hours is allowed to return it to OPERABLE status.

If the inoperable sensor cannot be restored to OPERABLE status within the specified Completion Time, the associated AFW pump must be declared inoperable. The TDAFW pump is considered OPERABLE when at least one train of the AFW loss of power start function is OPERABLE because both 6.9 kV shutdown board logic trains supply this function.

## BASES

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### ACTIONS (continued)

#### M.1.1, M.1.2, and M.2

Condition M applies to the Loss of Voltage sensors and load shed timers associated with the Loss of Power AFW pump start ESFAS Function. These are the same sensors and timers for the DG loss of Voltage start.

This function is provided by voltage sensors for each train arranged in a two-out-of-three logic scheme with associated load shed timers arranged in a one-out-of-two logic. If two or more voltage sensors or one required load shed timer are inoperable, 1 hour is allowed to return the inoperable channel(s) to OPERABLE status.

If the inoperable sensors cannot be made OPERABLE such that only one sensor is inoperable or one required load shed timer cannot be made OPERABLE within the specified Completion Time, the associated auxiliary feedwater pump must be declared inoperable. The AFW turbine-driven pump is considered OPERABLE when at least one train of the AFW loss of power start function is OPERABLE because both 6.9 kV shutdown board logic trains supply this function.

#### N.1 and N.2

Condition N applies to the AFW pump start on trip of all MFW pumps.

This action addresses the train orientation of the SSPS for the auto start function of the AFW System on loss of all MFW pumps. The OPERABILITY of the AFW System must be assured by allowing automatic start of the AFW System pumps. If a channel is inoperable, 48 hours are allowed to return it to an OPERABLE status. If the function cannot be returned to an OPERABLE status, 6 hours are allowed to place the unit in MODE 3. The allowed Completion Time is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging unit systems. In MODE 3, the unit does not have any analyzed transients or conditions that require the explicit use of the protection function noted above. The allowance of 48 hours to return the train to an OPERABLE status is justified in Reference 10.

The Required Actions are modified by a note delaying the entry into the Required Action statement when starting or stopping MFW pumps during MODE 1. Starting and stopping MFW pumps during plant startup and shutdown is a normal evolution, which will normally be accomplished within a short time. It was not intended to result in unnecessary entries into the Required Actions, which provides a timeframe to correct unplanned equipment failures.

## BASES

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### ACTIONS (continued)

The 4 hours is consistent with similar allowances in other SQN TSs.

#### O.1

Condition O applies to the following ESFAS Functions:

- Auxiliary Feedwater Pump Suction Transfer on Suction Pressure - Low,
- Auxiliary Feedwater Suction Transfer Time Delays, Motor-Driven Pump, and
- Auxiliary Feedwater Suction Transfer Time Delays, Turbine-Driven Pump.

These functions are provided by three pressure sensors located on the suction of each AFW pump arranged in a two-out-of-three logic scheme. The motor driven AFW pumps have one time delay, while the TDAFW pump has two. The motor driven and the first TDAFW pump time delays prevent spurious transfer. The TDAFW Pump second time delay ensures ERCW Train A valves stroke open sufficiently.

If a pressure sensor channel or a time delay channel is inoperable, the associated AFW pump must be declared inoperable immediately.

#### P.1, P.2.1, and P.2.2

Condition P applies to the RWST Level - Low Coincident with Safety Injection and Coincident with Containment Sump Level - High.

RWST Level - Low Coincident With SI and Coincident With Containment Sump Level - High provides actuation of switchover to the containment sump. Note that this Function requires the comparators to energize to perform their required action. The failure of up to two channels will not prevent the operation of this Function. However, placing a failed channel in the tripped condition could result in a premature switchover to the sump, prior to the injection of the minimum volume from the RWST. Placing the inoperable channel in bypass results in a two-out-of-three logic configuration, which satisfies the requirement to allow another failure

## BASES

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### ACTIONS (continued)

without disabling actuation of the switchover when required. Restoring the channel to OPERABLE status or placing the inoperable channel in the bypass condition within 6 hours is sufficient to ensure that the Function remains OPERABLE, and minimizes the time that the Function may be in a partial trip condition (assuming the inoperable channel has failed high). The 6 hour Completion Time is justified in Reference 11. If the channel cannot be returned to OPERABLE status or placed in the bypass condition within 6 hours, the unit must be brought to MODE 3 within the following 6 hours and MODE 5 within the next 30 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. In MODE 5, the unit does not have any analyzed transients or conditions that require the explicit use of the protection functions noted above.

The Required Actions are modified by a Note that allows placing a second channel in the bypass condition for up to 4 hours for surveillance testing. The total of 12 hours to reach MODE 3 and 4 hours for a second channel to be bypassed is acceptable based on the results of Reference 11.

#### Q.1, Q.2.1, and Q.2.2

Condition Q applies to the P-11 interlock.

With one or more channels inoperable, the operator must verify that the interlock is in the required state for the existing unit condition. This action manually accomplishes the function of the interlock. Determination must be made within 1 hour. The 1 hour Completion Time is equal to the time allowed by LCO 3.0.3 to initiate shutdown actions in the event of a complete loss of ESFAS function. If the interlock is not in the required state (or placed in the required state) for the existing unit condition, the unit must be placed in MODE 3 within the next 6 hours and MODE 4 within the following 6 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. Placing the unit in MODE 4 removes all requirements for OPERABILITY of these interlocks.

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ACTIONS (continued)

R.1 and R.2

If the inoperable channel cannot be placed in the tripped condition or the TTD of the single steam generator time delay calculation ( $T_S$ ) adjusted to match the multiple steam generator time delay calculation ( $T_M$ ) for the affected protection set within the specified Completion Time, the unit must be placed in a MODE where these Functions are not required OPERABLE. This requires the unit placed in MODE 3 within 6 hours and MODE 4 within 12 hours. The allowed Completion Times are reasonable to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. In MODE 4, these Functions are no longer required OPERABLE.

S.1 and S.2

Condition S applies to the automatic actuation logic and actuation relays for the Automatic Switchover to Containment Sump.

This action addresses the train orientation of the SSPS and the master and slave relays. If one train is inoperable the unit must be placed in a MODE in which the LCO does not apply. This is done by placing the unit in at least MODE 3 within 12 hours and in MODE 5 within an additional 30 hours (42 hours total time). The Completion Times are reasonable to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

The Required Actions are modified by a Note that allows one train to be bypassed for up to 4 hours for surveillance testing, provided the other train is OPERABLE.

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SURVEILLANCE  
REQUIREMENTS

The SRs for each ESFAS Function are identified by the SRs column of Table 3.3.2-1.

A Note has been added to the SR Table to clarify that Table 3.3.2-1 determines which SRs apply to which ESFAS Functions.

Note that each channel of process protection supplies both trains of the ESFAS. When testing channel I, train A and train B must be examined. Similarly, train A and train B must be examined when testing channel II, channel III, and channel IV (if applicable). The CHANNEL CALIBRATION and COTs are performed in a manner that is consistent with the assumptions used in analytically calculating the required channel accuracies.

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.2.1

Performance of the CHANNEL CHECK ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the unit staff, based on a combination of the channel instrument uncertainties, including indication and reliability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.3.2.2

SR 3.3.2.2 is the performance of an ACTUATION LOGIC TEST using the semiautomatic tester. The train being tested is placed in the bypass condition, thus preventing inadvertent actuation. Through the semiautomatic tester, all possible logic combinations, with and without applicable permissives, are tested for each protection function. In addition, the master relay coil is pulse tested for continuity. This verifies that the logic modules are OPERABLE and that there is an intact voltage signal path to the master relay coils

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

## BASES

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### SURVEILLANCE REQUIREMENTS (continued)

#### SR 3.3.2.3

SR 3.3.2.3 is the performance of a MASTER RELAY TEST. The MASTER RELAY TEST is the energizing of the master relay, verifying contact operation and a low voltage continuity check of the slave relay coil. Upon master relay contact operation, a low voltage is injected to the slave relay coil. This voltage is insufficient to pick up the slave relay, but large enough to demonstrate signal path continuity. The time allowed for the testing on a STAGGERED TEST BASIS (4 hours) is justified in Reference 12.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

#### SR 3.3.2.4

SR 3.3.2.4 is the performance of a COT.

A COT is performed on each required channel to ensure the entire channel will perform the intended Function. Setpoints must be found conservative with respect to the Allowable Values specified in Table 3.3.2-1. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable COT of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions.

The difference between the current "as-found" values and the previous test "as-left" values must be consistent with the drift allowance used in the setpoint methodology. The setpoint shall be left set consistent with the assumptions of the current unit specific setpoint methodology.

The "as-found" and "as-left" values must also be recorded and reviewed for consistency with the assumptions of Reference 7.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.3.2.4 is modified by two Notes as identified in Table 3.3.2-1. The first Note requires evaluation of channel performance for the condition where the as-found setting for the channel setpoint is outside its as-found tolerance but conservative with respect to the Allowable Value. Evaluation of channel performance will verify that the channel will

## BASES

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### SURVEILLANCE REQUIREMENTS (continued)

continue to behave in accordance with safety analysis assumptions and the channel performance assumptions in the setpoint methodology. The purpose of the assessment is to ensure confidence in the channel performance prior to returning the channel to service. For channels determined to be OPERABLE but degraded, after returning the channel to service the performance of these channels will be evaluated under the plant Corrective Action Program. Entry into the Corrective Action Program will ensure required review and documentation of the condition. The second Note requires that the as-left setting for the channel be returned to within the as-left tolerance of the NTSP. Where a setpoint more conservative than the NTSP is used in the plant surveillance procedures (field setting), the as-left and as-found tolerances, as applicable, will be applied to the surveillance procedure setpoint. This will ensure that sufficient margin to the Safety Limit and/or Analytical Limit is maintained. If the as-left channel setting cannot be returned to a setting within the as-left tolerance of the NTSP, then the channel shall be declared inoperable.

The second Note also requires that the methodologies for calculating the as-left and the as-found tolerances be in UFSAR, Section 7.1.2.

#### SR 3.3.2.5

SR 3.3.2.5 is the performance of a SLAVE RELAY TEST. The SLAVE RELAY TEST is the energizing of the slave relays. Contact operation is verified in one of two ways. Actuation equipment that may be operated in the design mitigation MODE is either allowed to function, or is placed in a condition where the relay contact operation can be verified without operation of the equipment. Actuation equipment that may not be operated in the design mitigation MODE is prevented from operation by the SLAVE RELAY TEST circuit. For this latter case, contact operation is verified by a continuity check of the circuit containing the slave relay.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.2.6

SR 3.3.2.6 is the performance of a TADOT. This test is a check of the Loss of Offsite Power Function. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable TADOT of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions.

The test also includes trip channels that provide actuation signals directly to the SSPS. The SR is modified by a Note that excludes verification of setpoints for relays. Relay setpoints require elaborate bench calibration and are verified during CHANNEL CALIBRATION.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.3.2.7

SR 3.3.2.7 is the performance of a TADOT. This test is a check of the Manual Actuation Functions and AFW pump start on trip of all MFW pumps. Each Manual Actuation Function is tested up to, and including, the master relay coils. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable TADOT of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. In some instances, the test includes actuation of the end device (i.e., pump starts, valve cycles, etc.).

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

The SR is modified by a Note that excludes verification of setpoints during the TADOT for manual initiation Functions. The manual initiation Functions have no associated setpoints.

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.2.8

SR 3.3.2.8 is the performance of a CHANNEL CALIBRATION.

CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to measured parameter within the necessary range and accuracy.

CHANNEL CALIBRATIONS must be performed consistent with the assumptions of the unit specific setpoint methodology. The difference between the current "as-found" values and the previous test "as-left" values must be consistent with the drift allowance used in the setpoint methodology.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

This SR is modified by a Note stating that this test should include verification that the time constants are adjusted to the prescribed values where applicable.

SR 3.3.2.8 is modified by two Notes as identified in Table 3.3.2-1. The first Note requires evaluation of channel performance for the condition where the as-found setting for the channel setpoint is outside its as-found tolerance but conservative with respect to the Allowable Value. Evaluation of channel performance will verify that the channel will continue to behave in accordance with safety analysis assumptions and the channel performance assumptions in the setpoint methodology. The purpose of the assessment is to ensure confidence in the channel performance prior to returning the channel to service. For channels determined to be OPERABLE but degraded, after returning the channel to service the performance of these channels will be evaluated under the plant Corrective Action Program. Entry into the Corrective Action Program will ensure required review and documentation of the condition. The second Note requires that the as-left setting for the channel be returned to within the as-left tolerance of the NTSP. Where a setpoint more conservative than the NTSP is used in the plant surveillance procedures (field setting), the as-left and as-found tolerances, as applicable, will be applied to the surveillance procedure setpoint. This will ensure that sufficient margin to the Safety Limit and/or Analytical Limit is maintained. If the as-left channel setting cannot be returned to a setting within the as-left tolerance of the NTSP, then the channel shall be declared inoperable.

The second Note also requires that the methodologies for calculating the as-left and the as-found tolerances be in UFSAR Section 7.1.2.

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.2.9

This SR ensures the individual channel ESF RESPONSE TIMES are less than or equal to the maximum values assumed in the accident analysis. Response Time testing acceptance criteria are included in the Updated Final Safety Analysis Report, Section 7.3 (Ref. 13). Individual component response times are not modeled in the analyses. The analyses model the overall or total elapsed time, from the point at which the parameter exceeds the Trip Setpoint value at the sensor, to the point at which the equipment in both trains reaches the required functional state (e.g., pumps at rated discharge pressure, valves in full open or closed position).

For channels that include dynamic transfer functions (e.g., lag, lead/lag, rate/lag, etc.), the response time test may be performed with the transfer functions set to one with the resulting measured response time compared to the appropriate UFSAR response time. Alternately, the response time test can be performed with the time constants set to their nominal value provided the required response time is analytically calculated assuming the time constants are set at their nominal values. The response time may be measured by a series of overlapping tests such that the entire response time is measured.

Response time may be verified by actual response time tests in any series of sequential, overlapping or total channel measurements, or by the summation of allocated sensor, signal processing and actuation logic response times with actual response time tests on the remainder of the channel. Allocations for sensor response times may be obtained from: (1) historical records based on acceptable response time tests (hydraulic, noise, or power interrupt tests), (2) in place, onsite, or offsite (e.g., vendor) test measurements, or (3) utilizing vendor engineering specifications. WCAP-13632-P-A, Revision 2, "Elimination of Pressure Sensor Response Time Testing Requirements," (Ref. 14) dated January 1996, provides the basis and methodology for using allocated sensor response times in the overall verification of the channel response time for specific sensors identified in the WCAP. Response time verification for other sensor types must be demonstrated by test.

WCAP-14036-P, Revision 1, "Elimination of Periodic Protection Channel Response Time Tests," (Ref. 15) provides the basis and methodology for using allocated signal processing and actuation logic response times in the overall verification of the protection system channel response time. The allocations for sensor, signal conditioning, and actuation logic response times must be verified prior to placing the component in operational service and re-verified following maintenance that may adversely affect response time. In general, electrical repair work does not

BASES

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SURVEILLANCE REQUIREMENTS (continued)

impact response time provided the parts used for repair are of the same type and value. Specific components identified in the WCAP may be replaced without verification testing. One example where response time could be affected is replacing the sensing assembly of a transmitter.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

This SR is modified by a Note that clarifies that the turbine driven AFW pump is tested within 24 hours after reaching 842 psig in the SGs.

SR 3.3.2.10

SR 3.3.2.10 is the performance of a TADOT as described in SR 3.3.2.7, except that it is performed for the P-4 Reactor Trip Interlock, and the Frequency is once per reactor trip breaker cycle. A successful test of the required contact(s) of a channel may be performed by the verification of the change of state of a single contact. This clarifies what is an acceptable TADOT. This Frequency is based on operating experience demonstrating that undetected failure of the P-4 interlock sometimes occurs when the reactor trip breaker is cycled.

The SR is modified by a Note that excludes verification of setpoints during the TADOT. The Function tested has no associated setpoint.

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REFERENCES

1. Regulatory Guide 1.105, "Setpoint for Safety Related Instrumentation," Revision 3.
2. UFSAR, Chapter 6.
3. UFSAR, Chapter 7.
4. UFSAR, Chapter 15.
5. IEEE-279-1971.
6. 10 CFR 50.49.
7. Calculation SQN-EEB-PL&S, Precautions, Limitations, and Setpoints for NSSS.
8. NUREG-1218, April 1988.
9. WCAP-14333-P-A, Rev. 1, October 1998.

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REFERENCES (continued)

10. WCAP-10271-P-A, Supplement 2, Rev. 1, June 1990.
  11. License Amendment dated June 13, 1995, Issuance of Amendments to Technical Specifications – Sequoyah Nuclear Plant, Units 1 and 2 (TAC NOS. M91990 and 91991) (ML013320052).
  12. WCAP-15376, Rev. 0. October 2000.
  13. UFSAR, Section 7.3.
  14. WCAP-13632-P-A, Revision 2, "Elimination of Pressure Sensor Response Time Testing Requirements," January 1996.
  15. WCAP-14036-P, Revision 1, "Elimination of Periodic Protection Channel Response Time Tests," December 1995.
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## B 3.3 INSTRUMENTATION

### B 3.3.3 Post Accident Monitoring (PAM) Instrumentation

#### BASES

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##### BACKGROUND

The primary purpose of the PAM instrumentation is to display unit variables that provide information required by the control room operators during accident situations. This information provides the necessary support for the operator to take the manual actions for which no automatic control is provided and that are required for safety systems to accomplish their safety functions for Design Basis Accidents (DBAs).

The OPERABILITY of the accident monitoring instrumentation ensures that there is sufficient information available on selected unit parameters to monitor and to assess unit status and behavior following an accident.

The availability of accident monitoring instrumentation is important so that responses to corrective actions can be observed and the need for, and magnitude of, further actions can be determined. These essential instruments are identified by unit specific documents (Ref. 1) addressing the recommendations of Regulatory Guide 1.97 (Ref. 2) as required by Supplement 1 to NUREG-0737 (Ref. 3).

The instrument channels required to be OPERABLE by this LCO include two classes of parameters identified during unit specific implementation of Regulatory Guide 1.97 as Type A and Category 1 variables.

Type A variables are included in this LCO because they provide the primary information required for the control room operator to take specific manually controlled actions for which no automatic control is provided, and that are required for safety systems to accomplish their safety functions for DBAs.

Category 1 variables are the key variables deemed risk significant because they are needed to:

- Permit the operator to take preplanned manual actions to accomplish safe plant shutdown,
- Determine whether other systems important to safety are performing their intended functions,
- Monitor the process of accomplishing or maintaining critical safety functions, and

## BASES

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### BACKGROUND (continued)

- Provide information to the operators that will enable them to determine the likelihood of a gross breach of the barriers to radioactivity release and to determine if a gross breach of a barrier has occurred.

These key variables are identified by the unit specific Regulatory Guide 1.97 analyses (Ref. 1). These analyses identify the unit specific Type A and Category 1 variables and provide justification for deviating from the NRC proposed list of Category 1 variables.

The specific instrument Functions listed in Table 3.3.3-1 are discussed in the LCO section.

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### APPLICABLE SAFETY ANALYSES

The PAM instrumentation ensures the operability of Regulatory Guide 1.97 Type A and Category 1 variables so that the control room operating staff can:

- Take the specified, pre-planned, manually controlled actions, for which no automatic control is provided, and that are required for safety systems to accomplish their safety function,
- Determine whether systems important to safety are performing their intended functions,
- Monitor the process of accomplishing or maintaining critical safety functions,
- Determine the likelihood of a gross breach of the barriers to radioactivity release, and
- Determine if a gross breach of a barrier has occurred.

PAM instrumentation that meets the definition of Type A in Regulatory Guide 1.97 satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii). Category 1, non-Type A, instrumentation must be retained in TS because it is intended to assist operators in minimizing the consequences of accidents. Therefore, Category 1, non-Type A, variables are important for reducing public risk.

## BASES

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### LCO

The PAM instrumentation LCO provides OPERABILITY requirements for Regulatory Guide 1.97 Type A monitors, which provide information required by the control room operators to perform certain manual actions specified in the unit Emergency Operating Procedures. These manual actions ensure that a system can accomplish its safety function, and are credited in the safety analyses. Additionally, this LCO addresses Regulatory Guide 1.97 instruments that have been designated Category 1, non-Type A.

The OPERABILITY of the PAM instrumentation ensures there is sufficient information available on selected unit parameters to monitor and assess unit status following an accident. This capability is consistent with the recommendations of Reference 2.

LCO 3.3.3 requires two OPERABLE channels for most Functions. Two OPERABLE channels ensure no single failure prevents operators from getting the information necessary for them to determine the safety status of the unit, and to bring the unit to and maintain it in a safe condition following an accident.

Furthermore, OPERABILITY of two channels allows a CHANNEL CHECK during the post accident phase to confirm the validity of displayed information. More than two channels may be required at some units if the unit specific Regulatory Guide 1.97 analyses (Ref. 1) determined that failure of one accident monitoring channel results in information ambiguity (that is, the redundant displays disagree) that could lead operators to defeat or fail to accomplish a required safety function.

The exception to the two channel requirement is Containment Isolation Valve (CIV) Position. In this case, the important information is the status of the containment penetrations. The LCO requires one position indicator for each active CIV. This is sufficient to redundantly verify the isolation status of each isolable penetration either via indicated status of the active valve and prior knowledge of a passive valve, or via system boundary status. If a normally active CIV is known to be closed and deactivated, position indication is not needed to determine status. Therefore, the position indication for valves in this state is not required to be OPERABLE.

Type A and Category 1 variables are required to meet Regulatory Guide 1.97 Category 1 (Ref. 2) design and qualification requirements for seismic and environmental qualification, single failure criterion, utilization of emergency standby power, immediately accessible display, continuous readout, and recording of display.

Listed below are discussions of the specified instrument Functions listed in Table 3.3.3-1.

## BASES

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### LCO (continued)

1, 2. Reactor Coolant T<sub>HOT</sub> and Reactor Coolant T<sub>COLD</sub> (Wide Range)

Reactor Coolant T<sub>HOT</sub> and Reactor Coolant T<sub>COLD</sub> (Wide Range) are Category 1 variables provided for verification of core cooling and long term surveillance.

Reactor Coolant T<sub>HOT</sub> temperatures are used to determine reactor coolant subcooling margin. Reactor coolant subcooling margin will allow termination of safety injection (SI), if still in progress, or reinitiation of SI if it has been stopped. Reactor coolant subcooling margin is also used for unit stabilization and cooldown control.

In addition, RCS cold leg temperature is used in conjunction with RCS hot leg temperature to verify the unit conditions necessary to establish natural circulation in the RCS.

Reactor outlet temperature inputs to the Reactor Protection System are provided by two fast response resistance elements and associated transmitters in each loop. The channels provide indication over a range of 0°F to 700°F.

There are a total of four Reactor Coolant System (RCS) Hot and Cold Leg Temperature channels each. One RCS Hot leg temperature channel per loop and one Cold Leg temperature channel per loop. The instrument loops associated with RCS T<sub>HOT</sub> are 68-001, -024, -043, and -065. The instrument loops associated with RCS T<sub>COLD</sub> are 68-018, 68-041, 68-060, and 68-083.

3. Containment Pressure (Wide Range)

Containment Pressure (Wide Range) is provided for determination of potential for containment breach.

The channels provide indication over a range of -5 to 60 psig. There are a total of two Containment Pressure (Wide Range) channels. The instrument loops associated with Containment Pressure (Wide Range) are 30-310 and 30-311.

## BASES

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### LCO (continued)

#### 4. Containment Pressure (Narrow Range)

Containment Pressure (Narrow Range) is provided for determination of an actual containment breach and if a break is inside or outside containment. Additionally it is provided to monitor containment conditions following a break inside containment and verifying the accident is properly controlled.

The channels provide indication over a range of -1 to 15 psig. There are a total of two Containment Pressure (Narrow Range) channels. The instrument loops associated with Containment Pressure (Narrow Range) 30-044 and 30-045.

#### 5. Refueling Water Storage Tank Level

Refueling Water Storage Tank Level is provided to verify a water source to emergency core cooling systems and containment spray system, determine the time for initiation of cold leg recirculation following a loss of coolant accident, and for event diagnosis.

The channels provide indication over a range of 0% to 100%. There are a total of two Refueling Water Storage Tank Level channels. The instrument loops associated with Refueling Water Storage Tank Level are 63-050 and 63-051.

#### 6. Reactor Coolant Pressure (Wide Range)

Reactor Coolant Pressure (Wide Range) is provided to determine if the plant is in safe shutdown condition. It is also used for maintaining the proper relationship between RCS pressure and temperature, verifying vessel nondestructive testing criteria, maintain primary inventory subcooled (particularly with loss of offsite power), establish correct conditions for residual heat removal operation, determine whether reactor coolant pump operation should be continued, and determine whether high-head SI should be terminated or reinitiated.

The channels provide indication over a range of 0 to 3000 psig. There are a total of three Reactor Coolant Pressure (Wide Range) channels. The instrument loops associated with Reactor Coolant Pressure (Wide Range) are 68-062, 68-066, and 68-069.

## BASES

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### LCO (continued)

#### 7. Pressurizer Level (Wide Range)

Pressurizer Level (Wide Range) is provided to confirm if plant is in a safe shutdown condition. It is also provided to monitor RCS inventory, maintain pressurizer water level, and determine whether SI should be terminated or reinitiated.

The channels provide indication over a range of 0% to 100%. There are a total of three Pressurizer Level (Wide Range) channels. The instrument loops associated with Pressurizer Level (Wide Range) are 68-320, 68-335, and 68-339.

#### 8. Steam Line Pressure

Steam Line Pressure is provided to determine if a high-energy secondary line rupture occurred. It is also provided to maintain an adequate reactor heat sink and verify auxiliary feedwater to steam generator associated with pipe rupture is isolated. It can be used to monitor secondary side pressure to: (1) verify operation of pressure control steam dump system, (2) maintain plant in safe shutdown condition, and (3) monitor RCS cooldown rate. It is diverse to  $T_{\text{COLD}}$  for natural circulation determination. In addition, it can be used for identification of steam generator tube rupture and determination that faulted steam generator is isolated.

The channels provide indication over a range of 0 to 1200 psig. There are a total of eight Steam Line Pressure channels, two per loop. The instrument loops associated with Steam Line Pressure are 1-002A, 1-002B, 1-009A, 1-009B, 1-020A, 1-020B, 1-027A, and 1-027B.

#### 9. Steam Generator Level - (Wide Range)

Steam Generator Level (Wide Range) is provided to determine if heat sink is being maintained and is used for SI termination for secondary break outside containment.

The channels provide indication over a range of 0 to 100 percent. There are a total of four Steam Generator Level - (Wide Range) channels, one per steam generator. The instrument loops associated with Steam Generator Level - (Wide Range) are 3-043, 3-056, 3-098, and 3-111.

BASES

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LCO (continued)

10. Steam Generator Level - (Narrow Range)

Steam Generator Level (Narrow Range) is provided to monitor heat sink, maintain steam generator water level, determine whether SI should be terminated, and determine which loop has SG tube rupture.

The channels provide indication over a range of 0 to 100 percent. There are a total of eight Steam Generator Level - (Narrow Range) channels, two per steam generator. The instrument loops associated with Steam Generator Level - (Narrow Range) are 3-039, 3-042, 3-052, 3-055, 3-094, 3-097, 3-107, and 3-110.

11. Auxiliary Feedwater

Auxiliary Feedwater (AFW) flow is provided to determine if sufficient flow exists to maintain heat sink and for SI termination. The channels provide indication over a range of 0 to 440 gpm. The redundant channel capability for AFW flow consists of a single AFW flow channel for each Steam Generator (four total, one per steam generator) with a diverse channel consisting of three AFW valve position indicators (two level control valves for the motor driven AFW flowpath and one level control valve for the turbine driven AFW flowpath) for each steam generator (12 total).

The instrument loops associated with AFW flow are 3-163, 3-155, 3-147, and 3-170. The instrument loops associated with AFW valve position indication are 3-164, 3-164A, 3-174, 3-156, 3-156A, 3-173, 3-148, 3-148A, 3-172, 3-171, 3-171A, and 3-175.

12. Reactor Coolant System Subcooling Margin Monitor

Reactor Coolant System Subcooling is provided for SI termination or reinitiation and maintenance of subcooling during depressurization.

The channels provide indication over a range of 200°F subcooled to 35°F superheat. There are a total of two Reactor Coolant System Subcooling Margin Monitor channels. The instrument loops associated with Reactor Coolant System Subcooling Margin Monitor are 94-101 and 94-102.

## BASES

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### LCO (continued)

#### 13. Containment Water Level (Wide Range)

Containment Water Level (Wide Range) is provided to verify water source for recirculation mode cooling, determine whether high energy line rupture has occurred inside or outside containment, and determine potential for containment breach caused by very high water levels.

The channels provide indication over a range of 0% to 100%. There are a total of two Containment Water Level (Wide Range) channels. The instrument loops associated with Containment Water Level (Wide Range) are 63-178 and 63-179.

#### 14. Incore Thermocouples

Incore thermocouples are provided to verify that the core is being adequately cooled, verify that RCS remains subcooled, and for monitoring the potential for fuel clad breach.

The channels provide indication over a range of 0°F to 2300°F. There are a total of 65 Incore Thermocouples. Each channel consists of one incore thermocouple. The minimum number of channels required is two channels per quadrant, eight per core, one/core quadrant/train. The two required channels in each quadrant shall be in different trains.

#### 15. Reactor Vessel Level Instrumentation

Reactor Vessel Level indication is provided for determination of core cooling. It is considered to be a more direct and less ambiguous indication of core cooling.

The channels provide indication over a range of 0% to 120% (dynamic range), 0% to 70% (lower range), and 64% to 120% (upper range). There are a total of six Reactor Vessel Level Instrument channels. The instrument loops associated with Reactor Vessel Level Dynamic Range are 68-367 and 68-370. The instrument loops associated with Reactor Vessel Level Lower Range are 68-368 and 68-371. The instrument loops associated with Reactor Vessel Level Upper Range are 68-369 and 68-372.

BASES

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LCO (continued)

16. Containment Area Radiation Monitors

Containment area radiation monitors are provided for accident diagnosis and SI Termination/Reinitiation.

The channels provide indication over a range of  $10^0$  to  $10^8$  R/hr. There are a total of four Containment Area Radiation Monitors. The instrument loops associated with Containment Area Radiation Monitors Upper Compartment are 90-271 and 90-272. The instrument loops associated with Containment Area Radiation Monitors Lower Compartment are 90-273 and 90-274.

17. Neutron Flux

The Intermediate Range and Source Range Neutron Flux are provided for monitoring reactivity control, determining if plant is subcritical, and to diagnose positive reactivity insertion.

The channels provide indication over a range of 1 to  $10^6$  CPS (Source Range) and  $10^{-8}$  to 200% RTP (Intermediate Range). There are a total of two Source Range channels and two Intermediate Range channels. The instrument loops associated with the Source Range are 92-5001 and 92-5002. The instrument loops associated with the Intermediate Range are 92-5003 and 92-5004

18. ERCW to AFW Valve Position

The ERCW to AFW valve position is provided for verification of heat sink availability. There is a total of four motor driven pump instrument loops. For the turbine driven AFW pump there is a total of four instrument loops. Each ERCW to AFW suction line contains two in-series isolation valves, each with its own position indication. Thus, position indication on both valves on a suction line is necessary. Each channel consists of the two valve position indications associated with the in-series valves in a single suction line.

The instrument loops associated with ERCW to AFW valve position for the Motor Driven Pumps are 3-116A, 3-116B, 3-126A, and 3-126B. The instrument loops associated with ERCW to AFW valve position for the Turbine Driven Pump are 3-136A, 3-136B, 3-179A, and 3-179B.

BASES

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LCO (continued)

19. Containment Isolation Valve Position

Containment Isolation valve position is provided for verification of containment isolation. There is one position indication instrument per isolation valve. The Containment Isolation valve position indications are located on Panels TR-A XX-55-6K and TR-B XX-55-6L.

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APPLICABILITY

The PAM instrumentation LCO is applicable in MODES 1, 2, and 3. These variables are related to the diagnosis and pre-planned actions required to mitigate DBAs. The applicable DBAs are assumed to occur in MODES 1, 2, and 3. In MODES 4, 5, and 6, unit conditions are such that the likelihood of an event that would require PAM instrumentation is low; therefore, the PAM instrumentation is not required to be OPERABLE in these MODES.

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ACTIONS

A Note has been added in the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed on Table 3.3.3-1. The Completion Time(s) of the inoperable channel(s) of a Function will be tracked separately for each Function starting from the time the Condition was entered for that Function.

A.1

Condition A applies when one or more Functions have one required channel that is inoperable. A Note is added stating that this ACTION is not applicable to Function 16, which has its own ACTION for one channel inoperable. Required Action A.1 requires restoring the inoperable channel to OPERABLE status within 30 days. The 30 day Completion Time is based on operating experience and takes into account the remaining OPERABLE channel (or in the case of a Function that has only one required channel, other non-Regulatory Guide 1.97 instrument channels to monitor the Function), the passive nature of the instrument (no critical automatic action is assumed to occur from these instruments), and the low probability of an event requiring PAM instrumentation during this interval. On a penetration where the position indication is declared inoperable on a valve but on the opposite side of the penetration a required containment isolation valve does not exist (such as with a closed system or a check valve), only Condition A must be entered. However, valves FCV-63-158 & -172 are both inboard penetration valves, but if both valves have inoperable position indication, Condition C must be entered until at least one of the valve's position indication is restored to OPERABLE status. Valves FCV-30-46 & VLV-30-571, FCV-30-47 & VLV-30-572, and FCV-30-48 & VLV-30-573 are all outboard penetration valves, but if both valves have inoperable

## BASES

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### ACTIONS (continued)

position indication, Condition C must be entered until at least one of the valve's position indication is restored to OPERABLE status.

#### B.1

Condition B applies when the Required Action and associated Completion Time for Condition A are not met. This Required Action specifies initiation of actions in Specification 5.6.5, which requires a written report to be submitted to the NRC immediately. This report discusses the results of the root cause evaluation of the inoperability and identifies proposed restorative actions. This action is appropriate in lieu of a shutdown requirement since alternative actions are identified before loss of functional capability, and given the likelihood of unit conditions that would require information provided by this instrumentation.

#### C.1

Condition C applies when one or more Functions have two inoperable required channels (i.e., two channels inoperable in the same Function). Required Action C.1 requires restoring one channel in the Function(s) to OPERABLE status within 7 days. The Completion Time of 7 days is based on the relatively low probability of an event requiring PAM instrument operation and the availability of alternate means to obtain the required information. Continuous operation with two required channels inoperable in a Function is not acceptable because the alternate indications may not fully meet all performance qualification requirements applied to the PAM instrumentation. Therefore, requiring restoration of one inoperable channel of the Function limits the risk that the PAM Function will be in a degraded condition should an accident occur.

#### D.1

Condition D applies when one or more Functions have three required channels that are inoperable. A Note is included that states that this ACTION is only applicable to Functions 6 and 7. Required Action D.1 requires restoring one inoperable channel to OPERABLE status within 48 hours.

## BASES

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### ACTIONS (continued)

#### E.1

Condition E applies when one or more steam generators have one AFW flow rate channel and one AFW valve position channel on the same steam generator inoperable. Required Action E.1 requires restoring one inoperable channel to OPERABLE status within 7 days.

#### F.1 and F.2

Condition F applies when one or more Containment Area Radiation Monitors have one required channel inoperable. Required Action F.1 requires initiating an alternate method of monitoring containment area radiation within 72 hours. Required Action F.2 requires restoring the inoperable channel(s) to OPERABLE status within 30 days.

#### G.1

Condition G applies when the Required Action and associated Completion Time of Conditions C, D, E, or F are not met. Required Action G.1 requires entering the appropriate Condition referenced in Table 3.3.3-1 for the channel immediately. The applicable Condition referenced in the Table is Function dependent. Each time an inoperable channel has not met the Required Action of Condition C, D, E, or F, and the associated Completion Time has expired, Condition G is entered for that channel and provides for transfer to the appropriate subsequent Condition.

#### H.1 and H.2

If the Required Action and associated Completion Time of Condition C, D, E, or F is not met and Table 3.3.3-1 directs entry into Condition H, the unit must be brought to a MODE where the requirements of this LCO do not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and 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.

BASES

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ACTIONS (continued)

I.1

Condition F requires initiation of alternate means of monitoring Containment Area Radiation. These alternate means may be temporarily installed if the normal PAM channel cannot be restored to OPERABLE status within the allotted time. If these alternate means are used, the Required Action is not to shut down the unit but rather to follow the directions of Specification 5.6.5, in the Administrative Controls section of the TS. The report provided to the NRC should discuss the alternate means used, describe the degree to which the alternate means are equivalent to the installed PAM channels, justify the areas in which they are not equivalent, and provide a schedule for restoring the normal PAM channels.

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SURVEILLANCE  
REQUIREMENTS

A Note has been added to the SR Table to clarify that SR 3.3.3.1 and SR 3.3.3.2 apply to each PAM instrumentation Function in Table 3.3.3-1.

SR 3.3.3.1

Performance of the CHANNEL CHECK ensures that a gross instrumentation failure has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION. The high radiation instrumentation should be compared to similar unit instruments located throughout the unit.

Agreement criteria are determined by the unit staff, based on a combination of the channel instrument uncertainties, including isolation, indication, and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit. If the channels are within the criteria, it is an indication that the channels are OPERABLE.

As specified in the SR, a CHANNEL CHECK is only required for those channels that are normally energized.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.3.2

CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to measured parameter with the necessary range and accuracy. This SR is modified by two Notes. The first Note excludes neutron detectors. The calibration method for neutron detectors is specified in the Bases of LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation." The second Note excludes the Containment Area Radiation Monitors detectors from a CHANNEL CALIBRATION for decade ranges above 10R/h. A CHANNEL CALIBRATION for the Containment Area Radiation Monitors detectors for decade ranges below 10R/h is performed by a single calibration check with either an installed or portable gamma source. Whenever a sensing element is replaced, the next required CHANNEL CALIBRATION of the Incore thermocouple sensors is accomplished by an inplace cross calibration that compares the other sensing elements with the recently installed sensing element.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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REFERENCES

1. SQN-EEB-PS-PAM-0001, PAM Variable QA Data – Base.
  2. Regulatory Guide 1.97, Revision 2, December 1980.
  3. NUREG-0737, Supplement 1, "TMI Action Items."
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## B 3.3 INSTRUMENTATION

### B 3.3.4 Remote Shutdown Monitoring Instrumentation

#### BASES

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**BACKGROUND** The Remote Shutdown Monitoring Instrumentation provides the control room operator with sufficient instrumentation to support placing and maintaining the unit in a safe shutdown condition from a location other than the control room. This capability is necessary to protect against the possibility that the control room becomes inaccessible. A safe shutdown condition is defined as MODE 3. With the unit in MODE 3, the Auxiliary Feedwater (AFW) System and the Main Steam Safety Valves (MSSV) or the atmospheric relief valves (ARVs) can be used to remove core decay heat and meet all safety requirements. The long term supply of water for the AFW System and the ability to borate the Reactor Coolant System (RCS) from outside the control room allows extended operation in MODE 3.

If the control room becomes inaccessible, the operators can monitor the status of the reactor at the locations shown on Table B 3.3.4-1. The unit automatically reaches MODE 3 following a unit shutdown and can be maintained safely in MODE 3 for an extended period of time.

The OPERABILITY of the remote shutdown monitoring instrumentation functions ensures there is sufficient information available on selected unit parameters to support placing and maintaining the unit in MODE 3 should the control room become inaccessible.

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**APPLICABLE SAFETY ANALYSES** The Remote Shutdown Monitoring Instrumentation is required to provide equipment at appropriate locations outside the control room with a capability to support placing and maintaining the unit in a safe condition in MODE 3.

The criteria governing the design and specific system requirements of the Remote Shutdown Monitoring Instrumentation are located in 10 CFR 50, Appendix A, GDC 19 (Ref. 1).

The Remote Shutdown Monitoring Instrumentation satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

BASES

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LCO

The Remote Shutdown Monitoring Instrumentation LCO provides the OPERABILITY requirements of the instrumentation necessary to support placing and maintaining the unit in MODE 3 from a location other than the control room. The instrumentation required are listed in Table B 3.3.4-1.

The monitoring instrumentation is required for:

- Core reactivity control (initial and long term);
- RCS pressure control;
- Reactor heat removal; and
- RCS makeup.

A Function of remote shutdown monitoring instrumentation is OPERABLE if each instrument needed to support the remote shutdown monitoring instrumentation is OPERABLE. For Table B 3.3.4-1 Function 7, the required information is available from several alternate sources. In this case, the Function is OPERABLE as long as one channel of any of the alternate information sources is OPERABLE.

The remote shutdown monitoring instrumentation covered by this LCO does not need to be energized to be considered OPERABLE. This LCO is intended to ensure the instruments will be OPERABLE if unit conditions require that the Remote Shutdown Monitoring Instrumentation be placed in operation.

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APPLICABILITY

The Remote Shutdown Monitoring Instrumentation LCO is applicable in MODES 1, 2, and 3. This is required so that the unit can be placed and maintained in MODE 3 for an extended period of time from a location other than the control room.

This LCO is not applicable in MODE 4, 5, or 6. In these MODES, the facility is already subcritical and in a condition of reduced RCS energy. Under these conditions, considerable time is available to restore the necessary instrument channels if control room instruments become unavailable.

## BASES

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### ACTIONS

The Remote Shutdown Monitoring Instrumentation LCO is not met when any required channel does not satisfy its OPERABILITY criteria for the channel's Function. These criteria are outlined in the LCO section of the Bases.

A Note has been added to the ACTIONS to clarify the application of Completion Time rules. Separate Condition entry is allowed for each Function. The Completion Time(s) of the inoperable channel(s) of a Function will be tracked separately for each Function starting from the time the Condition was entered for that Function.

#### A.1

Condition A addresses the situation where one or more required Functions of Remote Shutdown Monitoring Instrumentation are inoperable.

The Required Action is to restore the required Function to OPERABLE status within 30 days. The Completion Time is based on operating experience and the low probability of an event that would require evacuation of the control room.

#### B.1 and B.2

If the Required Action and associated Completion Time of Condition A is not met, 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 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

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### SURVEILLANCE REQUIREMENTS

#### SR 3.3.4.1

Performance of the CHANNEL CHECK ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

BASES

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SURVEILLANCE REQUIREMENTS (continued)

Agreement criteria are determined by the unit staff, based on a combination of the channel instrument uncertainties, including indication and readability. If the channels are within the criteria, it is an indication that the channels are OPERABLE. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

As specified in the Surveillance, a CHANNEL CHECK is only required for those channels which are normally energized.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.3.4.2

CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy.

Whenever a temperature sensing element is replaced, the next required CHANNEL CALIBRATION of the resistance temperature detectors (RTD) sensors is accomplished by an in-place cross calibration that compares the other sensing elements with the recently installed sensing element.

This SR is modified by two Notes, Note 1 excludes the neutron detectors and Note 2 excludes the reactor trip breaker indication from the CHANNEL CALIBRATION.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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REFERENCES      1. 10 CFR 50, Appendix A, GDC 19.

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Table B 3.3.4-1 (page 1 of 1)  
Remote Shutdown Monitoring Instrumentation

FUNCTION	READOUT LOCATION	MEASUREMENT RANGE	REQUIRED NUMBER OF CHANNELS
1. Source Range Nuclear Flux	Auxiliary Control Room Panel 2-L-10	1 to $1 \times 10^6$ cps	1
2. Reactor Trip Breaker Indication	at trip switchgear	OPEN-CLOSE	1/trip breaker
3. Reactor Coolant Temperature - Hot Leg	Auxiliary Control Room Panel 2-L-10	0-650°F	1/loop
4. Pressurizer Pressure	Auxiliary Control Room Panel 2-L-10	0-3000 psig	1
5. Pressurizer Level	Auxiliary Control Room Panel 2-L-10	0-100%	1
6. Steam Generator Pressure	Auxiliary Control Room Panel 2-L-10	0-1200 psig	1/steam generator
7. Steam Generator Level	Auxiliary Control Room Panels 2-L-11A and 2-L-11B or near Auxiliary Feedwater Pump	0-100%	1/steam generator
8. RHR Flow Rate	Auxiliary Control Room Panel 2-L-10	0-4500 gpm	1
9. RHR Temperature	Auxiliary Control Room Panel 2-L-10	50-400°F	1
10. Auxiliary Feedwater Flow Rate	Auxiliary Control Room Panel 2-L-10	0-440 gpm	1/steam generator
11. Pressurizer Relief Tank Pressure	Auxiliary Control Room Panel 2-L-10	0-100 psig	1
12. Containment Pressure	Auxiliary Control Room Panel 2-L-10	-1 to +15 psig	1

## B 3.3 INSTRUMENTATION

### B 3.3.5 Loss of Power (LOP) Diesel Generator (DG) Start Instrumentation

#### BASES

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#### BACKGROUND

The DGs provide a source of emergency power when offsite power is either unavailable or is insufficiently stable to allow safe unit operation. Undervoltage protection will generate a LOP start if a loss of voltage, unbalanced voltage, or degraded voltage condition occurs in the switchyard. There are three LOP start signals for each 6.9 kV Shutdown Board.

Six undervoltage relays (two per phase) and three unbalanced voltage relays (each three phase) are provided on each 6.9 kV Shutdown Board for detecting a sustained degraded voltage condition, unbalanced voltage condition, or a loss of bus voltage. The relays are combined in different logic configurations. Loss of Voltage Function and Degraded Voltage Functions are both two-out-of-three logic circuits. Unbalanced Voltage Relay is a permissive one-out-of-two logic circuit. The Loss of Voltage Function (Function 1.a) logic generates a LOP signal if the voltage is below a nominal 80% for a short time while the Degraded Voltage Function (Function 2.a) logic generates a LOP signal if the voltage is below a nominal 93.5% for a longer time. The Unbalanced Voltage Function generates a LOP signal if the alarm relay (1.30 V at 2.95 sec) and either the Low (2.96 V at 9.95 sec) or High (18.13 V at 3.95 sec) relay actuate to the determined voltage unbalance settings.

Six timers are provided on each 6.9 kV Shutdown Board, two timers associated with the Loss of Voltage Function logic and four timers associated with the Degraded Voltage Function logic. The two Loss of Voltage timers (Diesel Start and Load Shed Timers, Function 1.b) are arranged in a one-out-of-two logic with each timer set at a nominal 1.25 seconds. The Degraded Voltage timers are arranged in two sets of two; each set in a one-out-of-two logic. One set of Degraded Voltage timers (Diesel Start and Load Shed Timers, Function 2.b) are set at a nominal 300 seconds. The other set of Degraded Voltage timers (SI/Degraded Voltage Logic Enable Timers, Function 2.c) are set at a nominal 9.5 seconds. The timing functions for the Unbalanced Voltage relays are internal to the relays and set accordingly. These timers along with the under voltage relays, ensure adequate voltage is available to the safety related loads and that unintended actuations from degraded voltage or voltage perturbations will not occur.

## BASES

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### BACKGROUND (continued)

The Loss of Voltage Function voltage sensors monitor 6.9kV Shutdown Board voltage and actuate if voltage drops below 5520 volts. If two-out-of-three Loss of Voltage, Voltage Sensors detect less than 5520 volts, a signal is sent to the Diesel Generator Start and Load Shed Timers starting the 1.25 second timer. If Shutdown Board voltage increases to above the Loss of Voltage, Voltage Sensors setpoint before the Diesel Generator Start and Load Shed Timers reach their set time, the circuit returns to normal and the timers reset. If Shutdown Board voltage does not increase above the Loss of Voltage, Voltage Sensor setpoint within 1.25 seconds, a LOP signal is generated that trips the normal and alternate feeder breakers, starts the diesel generator, and trips major 6.9kV and 480V Shutdown Board loads.

The Degraded Voltage Function voltage sensors monitor 6.9kV Shutdown Board voltage and actuate if voltage drops below 6456 volts. If two-out-of-three Degraded Voltage, Voltage Sensors detect less than 6456 volts, a signal is sent to the Diesel Generator Start and Load Shed Timers starting their 300 second timer and to the SI/Degraded Voltage Logic Enable Timers starting their 9.5 second timer. If Shutdown Board voltage increases to above the Degraded Voltage, Voltage Sensors setpoint before the Diesel Generator Start and Load Shed Timers or the SI/Degraded Voltage Logic Enable Timers reach their set time, the circuit returns to normal and the timers reset. If Shutdown Board voltage does not increase above the Degraded Voltage, Voltage Sensor setpoint within 300 seconds a LOP signal is generated that trips the normal and alternate feeder breakers, starts the Diesel Generator, and trips major 6.9kV and 480V Shutdown Board loads. If Shutdown Board voltage does not increase above the Degraded Voltage, Voltage Sensor setpoint within 9.5 seconds and a safety injection signal is present or if a safety injection signal is generated after 9.5 seconds, a signal is generated that trips major 6.9kV and 480V Shutdown Board loads.

The Unbalanced Voltage Relay Function monitors 6.9 kV Shutdown Board voltage and actuates if the alarm relay (1.30 V unbalance) and either the Low (2.96 V unbalance) or High (18.13 V unbalance) Relays have actuated. If the voltage unbalance levels drop below setpoint values before the relays time settings are met, the circuit returns to normal and timers reset. If Shutdown Board voltage unbalance remains following the pre-determined time settings, a LOP signal is generated that is sent to the Degraded Voltage trip circuitry and trips the normal and alternate feeder breakers, starts the diesel generator, and trips major 6.9 kV and 480 V Shutdown Board loads.

The LOP start actuation is described in UFSAR, Section 8.3(Ref. 1).

## BASES

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### BACKGROUND (continued)

The Allowable Value in conjunction with the trip setpoint and LCO establishes the threshold for Engineered Safety Features Actuation System (ESFAS) action to prevent exceeding acceptable limits such that the consequences of Design Basis Accidents (DBAs) will be acceptable. The Allowable Value is considered a limiting value such that a channel is OPERABLE if the setpoint is found not to exceed the Allowable Value during the CHANNEL CALIBRATION. Note that although a channel is OPERABLE under these circumstances, the setpoint must be left adjusted to within the established calibration tolerance band of the setpoint in accordance with uncertainty assumptions stated in the referenced setpoint methodology, (as-left-criteria) and confirmed to be operating within the statistical allowances of the uncertainty terms assigned.

#### Allowable Values and LOP DG Start Instrumentation Setpoints

The Trip Setpoints used in the relays are based on the analytical limits presented in the associated setpoint scaling document/calculation. The selection of these trip setpoints is such that adequate protection is provided when all sensor and processing time delays are taken into account.

Setpoints adjusted consistent with the requirements of the Allowable Value ensure that the consequences of accidents will be acceptable, providing the unit is operated from within the LCOs at the onset of the accident and that the equipment functions as designed.

Allowable Values and/or Nominal Trip Setpoints are specified for each Function in Table 3.3.5-1. Nominal Trip Setpoints are also specified in the unit specific setpoint calculations. The trip setpoints are selected to ensure that the setpoint measured by the surveillance procedure does not exceed the Allowable Value if the relay is performing as required. If the measured setpoint does not exceed the Allowable Value, the relay is considered OPERABLE. Operation with a trip setpoint less conservative than the nominal Trip Setpoint, but within the Allowable Value, is acceptable provided that operation and testing is consistent with the assumptions of the unit specific setpoint calculation (Refs. 2, 3, 4, 6 and 7).

BASES

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APPLICABLE  
SAFETY  
ANALYSES

The LOP DG start instrumentation is required for the Engineered Safety Features (ESF) Systems to function in any accident with a loss of offsite power. Its design basis is that of the ESF Actuation System (ESFAS).

Accident analyses credit the loading of the DG based on the loss of offsite power during a loss of coolant accident (LOCA). The actual DG start has historically been associated with the ESFAS actuation. The DG loading has been included in the delay time associated with each safety system component requiring DG supplied power following a loss of offsite power. The analyses assume a non-mechanistic DG loading, which does not explicitly account for each individual component of loss of power detection and subsequent actions.

The required channels of LOP DG start instrumentation, in conjunction with the ESF systems powered from the DGs, provide unit protection in the event of any of the analyzed accidents discussed in Reference 5, in which a loss of offsite power is assumed.

The delay times assumed in the safety analysis for the ESF equipment include the 10 second DG start delay, and the appropriate sequencing delay, if applicable. The response times for ESFAS actuated equipment in LCO 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," include the appropriate DG loading and sequencing delay.

The LOP DG start instrumentation channels satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

BASES

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LCO The LCO for LOP DG start instrumentation requires that the loss of voltage, unbalanced voltage relay, and degraded voltage Functions shall be OPERABLE in MODES 1, 2, 3, and 4 when the LOP DG start instrumentation supports safety systems associated with the ESFAS, as required by Table 3.3.5-1. In MODES 5 and 6, the Functions must be OPERABLE, as required by Table 3.3.5-1, whenever the associated DG is required to be OPERABLE to ensure that the automatic start of the DG is available when needed. A channel is OPERABLE with a trip setpoint value outside its calibration tolerance band provided the trip setpoint "as-found" value does not exceed its associated Allowable Value and provided the trip setpoint "as-left" value is adjusted to a value within the "as-left" calibration tolerance band of the Nominal Trip Setpoint. A trip setpoint may be set more conservative than the Nominal Trip Setpoint as necessary in response to plant conditions. Loss of the LOP DG Start Instrumentation Function could result in the delay of safety systems initiation when required. This could lead to unacceptable consequences during accidents. During the loss of offsite power the DG powers the motor driven auxiliary feedwater pumps. Failure of these pumps to start would leave only one turbine driven pump, as well as an increased potential for a loss of decay heat removal through the secondary system.

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APPLICABILITY The LOP DG Start Instrumentation Functions are required in MODES 1, 2, 3, and 4 because ESF Functions are designed to provide protection in these MODES. Actuation in MODE 5 or 6 is required whenever the required DG must be OPERABLE so that it can perform its function on a LOP or degraded power to the associated 6.9 kV Shutdown Boards.

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ACTIONS In the event a channel's trip setpoint is found nonconservative with respect to the Allowable Value, or the channel is found inoperable, then the function that channel provides must be declared inoperable and the LCO Condition entered for the particular protection function affected.

Because the required channels are specified on a per shutdown board basis, the Condition may be entered separately for each shutdown board as appropriate.

A Note has been added in the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed in the LCO. The Completion Time(s) of the inoperable channel(s) of a Function will be tracked separately for each Function starting from the time the Condition was entered for that Function.

BASES

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ACTIONS (continued)

A.1

Condition A applies to the LOP DG start Functions with one or more Functions with one voltage sensor channel inoperable.

If one channel of the voltage sensors is inoperable, Required Action A.1 requires that channel to be restored to OPERABLE status within 6 hours.

The specified Completion Time is reasonable considering the Function remains fully OPERABLE and the low probability of an event occurring during these intervals.

B.1

Condition B applies when one or more Functions have two or more voltage sensor channels inoperable or one or more Functions have one required timer inoperable.

Required Action B.1.1 requires restoring all but one voltage sensor channel to OPERABLE status. Required Action B.1.2 requires restoring the required load shed timer to OPERABLE status. The 1 hour Completion Time takes into account the low probability of an event requiring a LOP start occurring during this interval.

C.1

Condition C applies when one or more unbalanced voltage relays are inoperable.

Required Action C.1 requires restoring all unbalanced voltage relays to OPERABLE status. The 1 hour Completion Time takes into account the low probability of an event requiring a LOP start occurring during this interval.

BASES

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ACTIONS (continued)

D.1

Condition D applies to each of the LOP DG start Functions when the Required Action and associated Completion Time for Condition A, B, or C are not met.

In these circumstances the Conditions specified in LCO 3.8.1, "AC Sources - Operating," or LCO 3.8.2, "AC Sources - Shutdown," for the DG made inoperable by failure of the LOP DG start instrumentation are required to be entered immediately. The actions of those LCOs provide for adequate compensatory actions to assure unit safety.

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SURVEILLANCE  
REQUIREMENTS

SR 3.3.5.1

SR 3.3.5.1 is the performance of a TADOT. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable TADOT of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. The test checks trip devices that provide actuation signals directly, bypassing the analog process control equipment. The SR is modified by a Note that excludes verification of setpoints for relays. Relay setpoints require elaborate bench calibration and are verified during CHANNEL CALIBRATION. There is a plant specific program which verifies that the instrument channel functions as required by verifying the as-left and as-found setting are consistent with those established by the setpoint methodology.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.3.5.2

SR 3.3.5.2 is the performance of a CHANNEL CALIBRATION.

The setpoints, as well as the response to a loss of voltage, unbalanced voltage, and a degraded voltage test, shall include a single point verification that the trip occurs within the required time delay, as shown in Reference 1.

BASES

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SURVEILLANCE REQUIREMENTS (continued)

CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy. There is a plant specific program which verifies that the instrument channel functions as required by verifying the as-left and as-found setting are consistent with those established by the setpoint methodology.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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REFERENCES

1. UFSAR, Section 8.3.
  2. TVA Calculation 27 DAT, "Demonstrated Accuracy Calculation 27 DAT."
  3. TVA Calculation DS1-2, "Demonstrated Accuracy Calculation DS1-2."
  4. TVA Calculation SQN-EEB-MS-TI06-0008, "Degraded Voltage Analysis."
  5. UFSAR, Chapter 15.
  6. TVA Calculation EDQ0002022016000331, "Determination of Unbalance Voltage Relay Analytical Limits."
  7. TVA Calculation EDQ0002022016000329, "Demonstrated Accuracy Calculation for Voltage Unbalance Relays."
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### B 3.3 INSTRUMENTATION

#### B 3.3.6 Containment Ventilation Isolation Instrumentation

##### BASES

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**BACKGROUND** Containment Ventilation isolation instrumentation closes the containment isolation valves in the Containment Purge System. This action isolates the containment atmosphere from the environment to minimize releases of radioactivity in the event of an accident. The Containment Purge System may be in use during reactor operation and with the reactor shutdown.

Containment Ventilation isolation initiates on a automatic safety injection (SI) signal or by manual actuation. The Bases for LCO 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," discuss initiation of SI signals.

The containment purge system has inner and outer containment isolation valves in its supply and exhaust ducts. A high radiation signal initiates containment ventilation isolation, which closes both inner and outer containment isolation valves in the Containment Purge System. This system is described in the Bases for LCO 3.6.3, "Containment Isolation Valves."

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**APPLICABLE SAFETY ANALYSES** The safety analyses assume that the containment remains intact with containment purge isolated early in the event, within approximately 300 seconds. The containment ventilation isolation radiation monitors, in addition to the SI signal, ensure closing of the containment purge supply and exhaust valves. They are also the primary means for automatically isolating containment in the event of a fuel handling accident during shutdown. Containment isolation in turn ensures meeting the containment leakage rate assumptions of the safety analyses, and ensures that the calculated accidental offsite radiological doses are below 10 CFR 100 (Ref. 1) limits (10 CFR 50.67 limits for a fuel handling accident).

The containment ventilation isolation instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

## BASES

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### LCO

The LCO requirements ensure that the instrumentation necessary to initiate Containment Ventilation Isolation, listed in Table 3.3.6-1, is OPERABLE.

#### 1. Manual Initiation

The LCO requires two channels OPERABLE. The operator can initiate Containment Ventilation Isolation at any time by using one of three sets of manual initiation switches in the control room. Either of the two Phase A and Containment Ventilation Isolation switches (HS-30-63A and HS-30-63B) or, both Phase B and Containment Ventilation Isolation switches (HS-30-64A and HS-30-64B), or both Phase B Containment Isolation switches (HS-30-68A and HS-30-68B), will actuate both trains of CVI. This action will cause actuation of all components in the same manner as any of the automatic actuation signals.

The LCO for Manual Initiation ensures the proper amount of redundancy is maintained in the manual actuation circuitry to ensure the operator has manual initiation capability.

Each channel consists of one selector switch and the interconnecting wiring to the actuation logic cabinet.

#### 2. Automatic Actuation Logic and Actuation Relays

The LCO requires two trains of Automatic Actuation Logic and Actuation Relays OPERABLE to ensure that no single random failure can prevent automatic actuation.

Automatic Actuation Logic and Actuation Relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b, SI. The applicable MODES and specified conditions for the containment ventilation isolation portion of the SI Function is different and less restrictive than those for the SI role. If one or more of the SI Functions becomes inoperable in such a manner that only the Containment Ventilation Isolation Function is affected, the Conditions applicable to the SI Functions need not be entered. The less restrictive Actions specified for inoperability of the Containment Ventilation Isolation Functions specify sufficient compensatory measures for this case.

BASES

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LCO (continued)

3. Containment Radiation

Table 3.3.6-1 specifies the number of required channels of radiation monitors to ensure that the radiation monitoring instrumentation necessary to initiate Containment Ventilation Isolation remains OPERABLE.

For sampling systems, channel OPERABILITY involves more than OPERABILITY of the channel electronics. OPERABILITY also requires correct valve lineup and sample pump operation, as well as detector OPERABILITY, for trip to occur under the conditions assumed by the safety analyses.

4. Safety Injection (SI)

Refer to LCO 3.3.2, Function 1, for all initiating Functions and requirements.

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APPLICABILITY

The Manual Initiation, Automatic Actuation Logic and Actuation Relays, Safety Injection, and Containment Radiation Functions are required OPERABLE as annotated on Table 3.3.6-1. Under these conditions, the potential exists for an accident that could release significant fission product radioactivity into containment. Therefore, the containment ventilation isolation instrumentation must be OPERABLE in these MODES.

While in MODES 5 and 6 without fuel handling in progress, the containment ventilation isolation instrumentation need not be OPERABLE since the potential for radioactive releases is minimized and operator action is sufficient to ensure post accident offsite doses are maintained within the limits of Reference 1.

The Applicability for the containment ventilation isolation on the ESFAS Safety Injection Functions are specified in LCO 3.3.2. Refer to the Bases for LCO 3.3.2 for discussion of the Safety Injection Function Applicability.

## BASES

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### ACTIONS

The most common cause of channel inoperability is outright failure or drift sufficient to exceed the tolerance allowed by unit specific calibration procedures. Typically, the drift is found to be small and results in a delay of actuation rather than a total loss of function. This determination is generally made during the performance of a COT, when the process instrumentation is set up for adjustment to bring it within specification. If the Trip Setpoint is less conservative than the tolerance specified by the calibration procedure, the channel must be declared inoperable immediately and the appropriate Condition entered.

A Note has been added to the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed in Table 3.3.6-1. The Completion Time(s) of the inoperable channel(s)/train(s) of a Function will be tracked separately for each Function starting from the time the Condition was entered for that Function.

#### A.1

Condition A applies to all Containment Ventilation Isolation Functions and addresses the train orientation of the Solid State Protection System (SSPS) and the master and slave relays for these Functions. It also addresses the failure of required radiation monitoring channel.

If a train is inoperable or the required channel is inoperable, operation may continue as long as the Required Action for the applicable Conditions of LCO 3.6.3 is met for each valve made inoperable by failure of isolation instrumentation.

A Note is added stating that Condition A is only applicable in MODE 1, 2, 3, or 4.

#### B.1 and B.2

Condition B applies to all Containment Ventilation Isolation Functions and addresses the train orientation of the SSPS and the master and slave relays for these Functions. It also addresses the failure of the required radiation monitoring channel. If a train or the required radiation monitoring channel is inoperable, operation may continue as long as the Required Action to place and maintain containment ventilation isolation valves in their closed position is met or the applicable Conditions of LCO 3.9.4, "Containment Penetrations," are met for each valve made inoperable by failure of isolation instrumentation. The Completion Time for these Required Actions is Immediately.

BASES

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ACTIONS (continued)

A Note states that Condition B is applicable during movement of irradiated fuel assemblies within containment.

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SURVEILLANCE  
REQUIREMENTS

A Note has been added to the SR Table to clarify that Table 3.3.6-1 determines which SRs apply to which Containment Ventilation Isolation Functions.

SR 3.3.6.1

Performance of the CHANNEL CHECK ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the unit staff, based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.3.6.2

SR 3.3.6.2 is the performance of an ACTUATION LOGIC TEST. The train being tested is placed in the bypass condition, thus preventing inadvertent actuation. Through the semiautomatic tester, all possible logic combinations, with and without applicable permissives, are tested for each protection function. In addition, the master relay coil is pulse tested for continuity. This verifies that the logic modules are OPERABLE and there is an intact voltage signal path to the master relay coils.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

## BASES

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### SURVEILLANCE REQUIREMENTS (continued)

The SR is modified by a Note stating that the Surveillance is only applicable to the actuation logic of the ESFAS Instrumentation.

#### SR 3.3.6.3

SR 3.3.6.3 is the performance of a MASTER RELAY TEST. The MASTER RELAY TEST is the energizing of the master relay, verifying contact operation and a low voltage continuity check of the slave relay coil. Upon master relay contact operation, a low voltage is injected to the slave relay coil. This voltage is insufficient to pick up the slave relay, but large enough to demonstrate signal path continuity.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

The SR is modified by a Note stating that the Surveillance is only applicable to the master relays of the ESFAS Instrumentation.

#### SR 3.3.6.4

A COT is performed on each required channel to ensure the entire channel will perform the intended Function. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable COT of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

This test verifies the capability of the instrumentation to provide the containment ventilation system isolation. There is a plant specific program which verifies that the instrument channel functions as required by verifying the as-left and as-found setting are consistent with those established by the setpoint methodology.

BASES

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.6.5

SR 3.3.6.5 is the performance of a SLAVE RELAY TEST. The SLAVE RELAY TEST is the energizing of the slave relays. Contact operation is verified in one of two ways. Actuation equipment that may be operated in the design mitigation mode is either allowed to function or is placed in a condition where the relay contact operation can be verified without operation of the equipment. Actuation equipment that may not be operated in the design mitigation mode is prevented from operation by the SLAVE RELAY TEST circuit. For this latter case, contact operation is verified by a continuity check of the circuit containing the slave relay.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.3.6.6

SR 3.3.6.6 is the performance of a TADOT. This test is a check of the Manual Actuation Functions. Each Manual Actuation Function is tested up to, and including, the master relay coils. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable TADOT of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. In some instances, the test includes actuation of the end device (i.e., pump starts, valve cycles, etc.).

The SR is modified by a Note that excludes verification of setpoints during the TADOT. The Functions tested have no setpoints associated with them.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.3.6.7

CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy. There is a plant specific program which verifies that the instrument channel functions as required by verifying the as-left and as-found setting are consistent with those established by the setpoint methodology.

BASES

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SURVEILLANCE REQUIREMENTS (continued)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.3.6.8

This SR ensures the individual channel response times are less than or equal to the maximum values assumed in the accident analysis. Response Time testing acceptance criteria are included in the Updated Final Safety Analysis Report, Table 7.3.1-4 (Ref. 2). Individual component response times are not modeled in the analyses. The analyses model the overall or total elapsed time, from the point at which the parameter exceeds the Trip Setpoint value to the point at which the equipment in both trains reaches the required functional state (e.g., valves in full open or closed position).

Response time may be verified by actual response time tests in any series of sequential, overlapping or total channel measurements, or by the summation of allocated signal processing and actuation logic response times with actual response time tests on the remainder of the channel.

WCAP-14036-P, Revision 1, "Elimination of Periodic Protection Channel Response Time Tests," (Ref. 3) provides the basis and methodology for using allocated signal processing and actuation logic response times in the overall verification of the protection system channel response time. The allocations for signal conditioning and actuation logic response times must be verified prior to placing the component in operational service and re-verified following maintenance that may adversely affect response time. In general, electrical repair work does not impact response time provided the parts used for repair are of the same type and value. Specific components identified in the WCAP may be replaced without verification testing.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.3.6.8 is modified by a Note stating that radiation detectors are excluded from response time testing.

BASES

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- REFERENCES
1. 10 CFR 100.11.
  2. UFSAR Table 7.3.1-4.
  3. WCAP-14036-P-A, Revision 1, "Elimination of Periodic Protection Channel Response Time Tests," December 1995.
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## B 3.3 INSTRUMENTATION

### B 3.3.7 Control Room Emergency Ventilation System (CREVS) Actuation Instrumentation

#### BASES

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**BACKGROUND** The CREVS provides an enclosed control room environment from which the unit can be operated following an uncontrolled release of radioactivity. During normal operation, the Control Building Ventilation System provides control room ventilation. Upon receipt of an actuation signal, the CREVS initiates filtered ventilation and pressurization of the control room. This system is described in the Bases for LCO 3.7.10, "Control Room Emergency Ventilation System (CREVS)."

The actuation instrumentation consists of redundant radiation monitors in the air intake. A high radiation signal from any detector will initiate its associated train of the CREVS. The control room operator can also initiate CREVS trains by manual switches in the control room. The CREVS is also actuated by a safety injection (SI) signal. The SI Function is discussed in LCO 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation."

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**APPLICABLE SAFETY ANALYSES** The control room must be kept habitable for the operators stationed there during accident recovery and post accident operations.

The CREVS acts to terminate the supply of unfiltered outside air to the control room, initiate filtration, and pressurize the control room. These actions are necessary to ensure the control room is kept habitable for the operators stationed there during accident recovery and post accident operations by minimizing the radiation exposure of control room personnel.

In MODES 1, 2, 3, and 4, the radiation monitor actuation of the CREVS is a backup for the SI signal actuation. This ensures initiation of the CREVS during a loss of coolant accident or main steam line break.

The radiation monitor actuation of the CREVS in MODES 5 and 6, during movement of irradiated fuel assemblies, and during CORE ALTERATIONS are the primary means to ensure control room habitability in the event of a fuel handling accident.

The CREVS actuation instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

BASES

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LCO

The LCO requirements ensure that instrumentation necessary to initiate the CREVS is OPERABLE.

1. Manual Initiation

The LCO requires two channels OPERABLE. The operator can initiate the CREVS at any time by using either of two switches in the control room. This action will cause actuation of all components in the same manner as any of the automatic actuation signals.

The LCO for Manual Initiation ensures the proper amount of redundancy is maintained in the manual actuation circuitry to ensure the operator has manual initiation capability.

Each channel consists of one hand switch and the interconnecting wiring to the actuation logic cabinet.

2. Control Room Radiation

The LCO specifies two required Control Room Air Intake Radiation Monitors to ensure that the radiation monitoring instrumentation necessary to initiate the CREVS remains OPERABLE.

For sampling systems, channel OPERABILITY involves more than OPERABILITY of channel electronics. OPERABILITY also requires correct valve lineups and sample pump operation, as well as detector OPERABILITY.

3. Safety Injection

Refer to LCO 3.3.2, Function 1, for all initiating Functions and requirements.

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APPLICABILITY

The CREVS Functions must be OPERABLE in MODES 1, 2, 3, 4, 5, and 6, during movement of irradiated fuel assemblies, and during CORE ALTERATIONS to ensure a habitable environment for the control room operators.

The Applicability for the CREVS actuation on the ESFAS Safety Injection Functions are specified in LCO 3.3.2. Refer to the Bases for LCO 3.3.2 for discussion of the Safety Injection Function Applicability.

## BASES

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### ACTIONS

The most common cause of channel inoperability is outright failure or drift sufficient to exceed the tolerance allowed by the unit specific calibration procedures. Typically, the drift is found to be small and results in a delay of actuation rather than a total loss of function. This determination is generally made during the performance of a COT, when the process instrumentation is set up for adjustment to bring it within specification. If the Trip Setpoint is less conservative than the tolerance specified by the calibration procedure, the channel must be declared inoperable immediately and the appropriate Condition entered.

A Note has been added to the ACTIONS indicating that separate Condition entry is allowed for each Function. The Conditions of this Specification may be entered independently for each Function listed in Table 3.3.7-1 in the accompanying LCO. The Completion Time(s) of the inoperable channel(s)/train(s) of a Function will be tracked separately for each Function starting from the time the Condition was entered for that Function.

#### A.1

Condition A applies to the actuation logic train Function of the CREVS, the radiation monitor channel Functions, and the manual channel Functions.

If one train is inoperable, or one radiation monitor channel is inoperable in one or more Functions, 7 days are permitted to restore it to OPERABLE status. The 7 day Completion Time is the same as is allowed if one train of the mechanical portion of the system is inoperable. The basis for this Completion Time is the same as provided in LCO 3.7.10. If the channel/train cannot be restored to OPERABLE status, one CREVS train must be placed in the recirculation mode of operation. This accomplishes the actuation instrumentation Function and places the unit in a conservative mode of operation.

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

Condition B applies to the failure of two CREVS actuation trains, two radiation monitor channels, or two manual channels. The first Required Action is to place one CREVS train in the recirculation mode of operation immediately. This accomplishes the actuation instrumentation Function that may have been lost and places the unit in a conservative mode of operation. The applicable Conditions and Required Actions of LCO 3.7.10 must also be entered for the CREVS train made inoperable by the inoperable actuation instrumentation. This ensures appropriate limits are placed upon train inoperability as discussed in the Bases for LCO 3.7.10.

## BASES

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### ACTIONS (continued)

Alternatively, both trains may be placed in the recirculation mode. This ensures the CREVS function is performed even in the presence of a single failure.

#### C.1 and C.2

Condition C applies when the Required Action and associated Completion Time for Condition A or B have not been met and the unit is in MODE 1, 2, 3, or 4. The unit must be brought to a MODE in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

#### D.1 and D.2

Condition D applies when the Required Action and associated Completion Time for Condition A or B have not been met when irradiated fuel assemblies are being moved or when CORE ALTERATIONS are being performed. Movement of irradiated fuel assemblies and CORE ALTERATIONS must be suspended immediately to reduce the risk of accidents that would require CREVS actuation.

#### E.1

Condition E applies when the Required Action and associated Completion Time for Condition A or B have not been met in MODE 5 or 6. Actions must be initiated to restore the inoperable train(s) to OPERABLE status immediately to ensure adequate isolation capability in the event of a fuel handling accident.

BASES

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SURVEILLANCE  
REQUIREMENTS

A Note has been added to the SR Table to clarify that Table 3.3.7-1 determines which SRs apply to which CREVS Actuation Functions.

SR 3.3.7.1

Performance of the CHANNEL CHECK ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the unit staff, based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.3.7.2

A COT is performed on each required channel to ensure the entire channel will perform the intended function. This test verifies the capability of the instrumentation to provide the CREVS actuation. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable COT of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. There is a plant specific program which verifies that the instrument channel functions as required by verifying the as-left and as-found setting are consistent with those established by the setpoint methodology.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.7.3

SR 3.3.7.3 is the performance of a TADOT. This test is a check of the Manual Actuation Functions. Each Manual Actuation Function is tested up to, and including, the master relay coils. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable TADOT of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. In some instances, the test includes actuation of the end device (i.e., pump starts, valve cycles, etc.).

The test also includes trip devices that provide actuation signals directly to the Solid State Protection System, bypassing the analog process control equipment.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

The SR is modified by a Note that excludes verification of setpoints during the TADOT. The Functions tested have no setpoints associated with them.

SR 3.3.7.4

CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy.

There is a plant specific program which verifies that the instrument channel functions as required by verifying the as-left and as-found setting are consistent with those established by the setpoint methodology.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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REFERENCES

1. WCAP-15376, Rev. 0, October 2000.
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### B 3.3 INSTRUMENTATION

#### B 3.3.8 Auxiliary Building Gas Treatment System (ABGTS) Actuation Instrumentation

##### BASES

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**BACKGROUND** The ABGTS ensures that radioactive materials in the auxiliary building atmosphere following a fuel handling accident involving handling irradiated fuel or a loss of coolant accident (LOCA) are filtered and adsorbed prior to exhausting to the environment. The system is described in the Bases for LCO 3.7.12, "Auxiliary Building Gas Treatment System (ABGTS)." The system initiates filtered ventilation of exhaust air from the fuel handling area, ECCS pump rooms, and waste packaging area automatically following receipt of a spent fuel pool area high radiation signal or a Containment Phase A Isolation signal. Initiation may also be performed manually as needed from the main control room.

High area radiation, monitored by either of two monitors, provides ABGTS initiation. Each ABGTS train is initiated by high radiation detected by a channel dedicated to that train. There are a total of two channels, one for each train. High radiation detected by the required monitor or Containment Phase A Isolation signal from the Engineered Safety Features Actuation System (ESFAS) initiates auxiliary building isolation and starts the ABGTS. These actions function to prevent exfiltration of contaminated air by initiating filtered ventilation, which imposes a negative pressure on the Auxiliary Building Secondary Containment Enclosure (ABSCE). During plant operations with the containment open to the auxiliary building, the ABSCE boundary is extended to include the area inside the containment building and the shield building.

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**APPLICABLE SAFETY ANALYSES** The ABGTS ensures that radioactive materials in the ABSCE atmosphere following a LOCA are filtered and adsorbed prior to being exhausted to the environment. This action reduces the radioactive content in the auxiliary building exhaust following a LOCA so that offsite doses remain within the limits specified in 10 CFR 100 (Ref. 1).

High radiation initiation of ABGTS for a fuel handling accident will ensure the auxiliary building secondary containment enclosure (ABSCE) boundary is established such that the release point for the fission products will correspond to the release point assumed in the fuel handling accident analysis so that control room doses remain within the limits specified in 10 CFR 50.67. (Ref. 2)

The ABGTS actuation instrumentation for LOCA satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

## BASES

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### LCO

The LCO requirements ensure that instrumentation necessary to initiate the ABGTS is OPERABLE.

#### 1. Spent Fuel Pool Area Radiation

The LCO specifies one Spent Fuel Pool Area Radiation Monitor channel to ensure that the radiation monitoring instrumentation necessary to initiate the ABGTS remains OPERABLE. One radiation monitor is dedicated to each train of ABGTS.

For sampling systems, channel OPERABILITY involves more than OPERABILITY of channel electronics. OPERABILITY may also require correct valve lineups, sample pump operation, filter motor operation, detector OPERABILITY, if these supporting features are necessary for actuation to occur under the conditions assumed by the safety analyses. The measurement range for the Spent Fuel Pool Area Monitors is  $10^{-1}$  to  $10^4$  mR/hr.

The Required Channels value is modified by a footnote stating that the Required Channel shall be associated with the ABGTS train required OPERABLE per LCO 3.7.12. This ensures a valid actuation signal will start a train of ABGTS.

#### 2. Containment Isolation - Phase A

Refer to LCO 3.3.2, Function 3.a., for all initiating Functions and requirements.

Only the Trip Setpoint is specified for each ABGTS Function in the LCO. The Trip Setpoint limits account for instrument uncertainties, which are defined in TI-18, Radiation Monitoring Procedure (Ref. 3).

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### APPLICABILITY

The automatic ABGTS actuation instrumentation is required to be OPERABLE in MODES 1, 2, 3, and 4 to remove fission products caused by post LOCA Emergency Core Cooling Systems leakage and is addressed in LCO 3.3.2.

High radiation initiation of the ABGTS must be OPERABLE in any MODE during movement of irradiated fuel assemblies or storage of fuel assemblies in the spent fuel pool to ensure automatic initiation of the ABGTS when the potential for a fuel handling accident exists. High radiation initiation of ABGTS for a fuel handling accident will ensure the auxiliary building secondary containment enclosure (ABSCE) boundary is established such that the release point for the fission products will correspond to the release point assumed in the dose consequences analysis for the fuel handling accident.

## BASES

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### ACTIONS

The most common cause of channel inoperability is outright failure or drift sufficient to exceed the tolerance allowed by unit specific calibration procedures. Typically, the drift is found to be small and results in a delay of actuation rather than a total loss of function. This determination is generally made during the performance of a COT, when the process instrumentation is set up for adjustment to bring it within specification.

If the Trip Setpoint is less conservative than the tolerance specified by the calibration procedure, the channel must be declared inoperable immediately and the appropriate Condition entered.

LCO 3.0.3 is not applicable while in MODE 5 or 6. However, since irradiated fuel assembly movement and fuel assembly storage in the spent fuel pool can occur in MODE 1, 2, 3, or 4, the ACTIONS have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving or storing irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving or storing irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement or fuel storage is independent of reactor operations. Entering LCO 3.0.3, while in MODE 1, 2, 3, or 4 would require the unit to be shutdown unnecessarily.

#### A.1

Condition A applies to the area radiation monitor being inoperable solely due to the trip setpoint not within limits. Condition A applies to the failure of the required radiation monitor channel. If the required channel is inoperable, a period of 4 hours is allowed to adjust the setpoint and restore it to OPERABLE status. If the channel cannot be restored to OPERABLE status, the Required Action and associated Completion Time of Condition B would apply.

#### B.1

Condition B applies when the Required Action and associated Completion Time for Condition A has not been met or the required channel is inoperable for reasons other than the trip setpoint not within limit. Entry into the applicable Conditions and Required Actions of LCO 3.7.12 will require that movement of irradiated fuel assemblies or movement of loads over the spent fuel pool be suspended immediately to eliminate the potential for events that could require ABGTS actuation.

BASES

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SURVEILLANCE  
REQUIREMENTS

A Note has been added to the SR Table to clarify that Table 3.3.8-1 determines which SRs apply to which ABGTS Actuation Functions.

SR 3.3.8.1

Performance of the CHANNEL CHECK ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the unit staff, based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.3.8.2

A COT is performed on each required channel to ensure the entire channel will perform the intended function. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable COT of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. This test verifies the capability of the instrumentation to provide the ABGTS actuation. There is a plant specific program which verifies that the instrument channel functions as required by verifying the as-left and as-found setting are consistent with those established by the setpoint methodology.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.8.3

CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy. There is a plant specific program which verifies that the instrument channel functions as required by verifying the as-left and as-found setting are consistent with those established by the setpoint methodology.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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- REFERENCES
1. 10 CFR 100.11.
  2. 10 CFR 50.67.
  3. TI-18, Radiation Monitoring Procedure.
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## B 3.3 INSTRUMENTATION

### B 3.3.9 Boron Dilution Monitoring Instrumentation (BDMI)

#### BASES

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**BACKGROUND** The primary purpose of the BDMI is to provide an indication of inadvertent positive reactivity changes when the reactor is in a shutdown condition (i.e., MODES 3, 4, and 5). Based on this indication, operator action can be taken to mitigate the consequences of the inadvertent addition of unborated primary grade water into the Reactor Coolant System (RCS).

The required BDMI consists of one OPERABLE channel of the two channels of source range instrumentation. The requirement for an OPERABLE source range channel ensures the capability to monitor core reactivity and detect a boron dilution event. In order to promptly detect the event in MODES 3, 4, and 5 the required source range instrumentation must provide visual, audible (count rate indication), and alarm in the control room.

The source range neutron flux monitors are used to monitor the core reactivity condition. The installed source range neutron flux monitors are part of the Nuclear Instrumentation System (NIS). These detectors are located external to the reactor vessel and detect neutrons leaking from the core.

The installed source range neutron flux monitors are dual chamber unguarded fission chamber detectors. The detectors monitor the neutron flux in counts per second. The instrument range covers six decades of neutron flux (1E+6 cps). The detectors also provide continuous visual indication in the control room, an audible count rate (selectable between the two source range neutron flux channels), and a high flux at shutdown alarm to alert operators to a possible dilution accident.

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**APPLICABLE SAFETY ANALYSES** The BDMI senses abnormal increases in source range counts per minute (flux rate). One OPERABLE source range neutron flux channel is required to provide a signal to alert the operator to unexpected changes in core reactivity. Following reactor shutdown the high flux at shutdown alarm setting will be automatically adjusted downward to a nominal value of 3 times the background count rate as the background count rate reduces. The operator does not depend entirely on this alarm setpoint but has audible indication of increasing neutron flux from the audible count rate drawer and visual indication from counts per second meters for each channel on the main control board and source range drawer. The audible count rate from the source range neutron flux monitors provides prompt and definite indication of any boron dilution. The count rate increase is proportional to the subcritical multiplication factor and allows

BASES

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APPLICABLE SAFETY ANALYSES (continued)

operators to promptly recognize the initiation of a boron dilution event. Two cases are analyzed for a boron dilution accident following reactor shutdown, beginning of life (BOL) with equilibrium xenon and BOL with a clean core. The analysis shows that for both cases > 15 minutes following the high flux at shutdown alarm for operator action time is available before reaching a  $k_{\text{eff}}$  of 1.0. Therefore, the acceptance criterion for this event is met. The source range neutron flux monitoring channel is credited in the boron dilution accident in UFSAR, Section 15.2.4 (Ref. 1).

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

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LCO

LCO 3.3.9 provides the requirements for OPERABILITY for instrumentation necessary to detect a boron dilution event and monitor core reactivity. In the applicable plant condition, the specified instrumentation is required to provide a core reactivity monitoring function and is not required to provide a trip function. Therefore, in MODES 3, 4, and 5 a single OPERABLE source range channel with visual, audible (count rate indication), and alarm in the control room is required to provide prompt indication of an inadvertent boron dilution.

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APPLICABILITY

The BDMI must be OPERABLE in MODES 3, 4, and 5 because the safety analysis identifies this system as the primary means to indicate the need for operator action to mitigate an inadvertent boron dilution of the RCS.

The BDMI OPERABILITY requirements are not applicable in MODES 1 and 2 because an inadvertent boron dilution would be terminated by a source range trip, a trip on the Power Range Neutron Flux - High (low setpoint nominally 25% RTP), or Overtemperature  $\Delta T$ . These RTS Functions are discussed in LCO 3.3.1, "RTS Instrumentation."

In MODE 6, a dilution event is precluded by locked valves that isolate the RCS from the potential source of unborated water (according to LCO 3.9.2, "Unborated Water Source Isolation Valves").

The Applicability is modified by a Note that allows the high flux at shutdown alarm to be blocked during reactor startup in MODE 3. Blocking the high flux at shutdown alarm is acceptable during startup while in MODE 3, provided the reactor trip breakers are closed with the intent to withdraw rods for startup.

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ACTIONS

The most common cause of channel inoperability is outright failure or drift of the bistable or process module sufficient to exceed the tolerance allowed by the unit specific calibration procedure. Typically, the drift is found to be small and results in a delay of actuation rather than a total loss of function. This determination of drift is generally made during the

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BASES

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## ACTIONS (continued)

performance of a COT when the process instrumentation is set up for adjustment to bring it to within specification. If the channel is outside the tolerance specified by the calibration procedure, the channel must be evaluated and the appropriate Condition entered.

A.1

Required Action A.1 requires the verification of SDM according to SR 3.1.1.1 within 1 hour and once per 12 hours thereafter. This action is intended to confirm that no unintended boron dilution has occurred while the BDMI was inoperable, and that the required SDM has been maintained. The specified Completion Time takes into consideration sufficient time for the initial determination of SDM and other information available in the control room related to SDM.

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SURVEILLANCE  
REQUIREMENTSSR 3.3.9.1

Performance of the CHANNEL CHECK ensures that gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the unit staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.9.2

SR 3.3.9.2 requires the performance of a COT to ensure that the required channel of the BDMI and associated alarm setpoint are fully operational. This test shall include verification that the High Flux at Shutdown alarm setpoint is three times the background count rate. There is a plant specific program which verifies that the instrument channel functions as required by verifying the as-left and as-found settings are consistent with those established by the setpoint methodology.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. (Ref. 2)

SR 3.3.9.3

SR 3.3.9.3 is the performance of a CHANNEL CALIBRATION. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. There is a plant specific program which verifies that the instrument channel functions as required by verifying the as-left and as-found setting are consistent with those established by the setpoint methodology. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy. The CHANNEL CALIBRATION shall include checking the discriminator voltage and adjusting if necessary.

This SR is modified by a Note that states that neutron detectors are excluded from the CHANNEL CALIBRATION.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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REFERENCES

1. UFSAR, Section 15.2.4.
  2. WCAP-15376, Revision 0, October 2000.
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## B 3.4 REACTOR COOLANT SYSTEM (RCS)

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

#### BASES

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

The RCS pressure limit is consistent with operation within the nominal operational envelope. Each pressurizer pressure indication is compared to the limit. A lower pressure will cause the reactor core to approach DNB limits.

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

The RCS flow rate normally remains constant during an operational fuel cycle with all pumps running. The minimum RCS flow limit corresponds to that assumed for DNB analyses. Each OPERABLE flow rate indication is compared to the limit. If one or more flow rate indications are unavailable, the remaining flow rate indications are averaged to come up with a value for comparison to the limit. A lower RCS flow will cause the core to approach DNB limits.

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

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**APPLICABLE SAFETY ANALYSES** The requirements of this LCO represent the initial conditions for DNB limited transients analyzed in the plant safety analyses (Ref. 1). The safety analyses have shown that transients initiated from the limits of this LCO will result in meeting the DNBR criterion. This is the acceptance limit for the RCS DNB parameters. Changes to the unit that could impact these parameters must be assessed for their impact on the DNBR criteria. The transients analyzed for include loss of coolant flow events and dropped or misaligned rod events. A key assumption for the analysis of these events is that the core power distribution is within the limits of LCO 3.1.6, "Control Bank Insertion Limits," LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)."

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BASES

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APPLICABLE SAFETY ANALYSES (continued)

The pressurizer pressure limit and RCS average temperature limit correspond to the analytical limits used in the safety analyses, with allowance for measurement uncertainty.

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

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LCO

This LCO specifies limits on the monitored process variables - pressurizer pressure, RCS average temperature, and RCS total flow rate - to ensure the core operates within the limits assumed in the safety analyses. The minimum RCS flow is based on the maximum analyzed steam generator tube plugging. Operating within these limits will result in meeting the DNBR criterion in the event of a DNB limited transient.

RCS flow indication calibration must include appropriate considerations for the accuracy of feedwater flow measurement. Sequoyah Nuclear Plant (SQN) can employ either of two methods to measure feedwater flow; an installed Leading Edge Flow Meter (LEFM), or in-line feedwater flow venturis. Unlike the feedwater venturis, the LEFM is not susceptible to fouling during use and possesses a higher accuracy. These attributes make the LEFM the preferred method of measuring feedwater flow as an input to the determination of RCS flow.

In the event the LEFM is unavailable, the feedwater venturis are used to calibrate the RCS flow indicators. However, the calibration assumptions for flow measurement uncertainties is not applicable to the case where the power calorimetric is based on the venturi feedwater flow indication, even if the LEFM is used to correct the venturi feedwater flow indications for the effects of fouling. For those instances where the LEFM is unavailable, SQN Technical Requirements Manual (TRM) specifies the appropriate actions to be taken.

The numerical values for pressure, temperature, and flow rate specified in the LCO are given for the measurement location and have been adjusted for instrument error.

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APPLICABILITY

In MODE 1, the limits on pressurizer pressure, RCS coolant average temperature, and RCS flow rate must be maintained during steady state operation in order to ensure 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 that DNB is not a concern.

BASES

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APPLICABILITY (continued)

A Note has been added to indicate the limit on pressurizer pressure is not applicable during short term operational transients such as a THERMAL POWER ramp increase > 5% RTP per minute or a THERMAL POWER step increase > 10% RTP. Additionally, the limit on pressurizer pressure is not applicable during PHYSICS TESTS and during the performance of SR 3.1.3.2, moderator temperature coefficient (MTC) determination. Measurement of MTC has a high probability of causing a drop in pressure below the specified value, because the reactor coolant system temperature must be dropped several degrees below  $T_{avg}$  for an accurate MTC measurement. This results in an associated drop in pressurizer level and in a downswing of pressurizer pressure, making it difficult to maintain pressurizer pressure above the limit. These conditions represent short term perturbations where actions to control pressure variations might be counterproductive. Also, since they represent transients initiated from power levels < 100% RTP, an increased DNBR margin exists to offset the temporary pressure variations.

The DNBR limit is provided in SL 2.1.1, "Reactor Core SLs." The conditions which define the DNBR limit are less restrictive than the limits of this LCO, but violation of a Safety Limit (SL) merits a stricter, more severe Required Action. Should a violation of this LCO occur, the operator must check whether or not an SL may have been exceeded.

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ACTIONS

A.1

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

RCS total flow rate is not a controllable parameter and is not expected to vary during steady state operation. If the indicated RCS total flow rate is below the LCO limit, power must be reduced, 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 for the off normal condition, and to restore the readings within limits, and 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

BASES

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ACTIONS (continued)

potential for violation of the accident analysis bounds. The Completion Time of 6 hours is reasonable to reach the required plant conditions in an orderly manner.

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SURVEILLANCE  
REQUIREMENTS

SR 3.4.1.1

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.4.1.2

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.4.1.3

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.4.1.4

Measurement of RCS total flow rate by performance of an elbow tap differential flow method (Ref. 2) allows the installed RCS flow instrumentation to be calibrated and verifies the actual RCS flow rate is greater than or equal to the minimum required RCS flow rate.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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REFERENCES

1. UFSAR, Chapter 15.
  2. UFSAR, Section 7.2.2.2.2.
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## B 3.4 REACTOR COOLANT SYSTEM (RCS)

### B 3.4.2 RCS Minimum Temperature for Criticality

#### BASES

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BACKGROUND	<p>This LCO is based upon meeting several major considerations before the reactor can be made critical and while the reactor is critical.</p>
	<p>The first consideration is moderator temperature coefficient (MTC), LCO 3.1.3, "Moderator Temperature Coefficient (MTC)." In the transient and accident analyses, the MTC is assumed to be in a range from slightly positive to negative and the operating temperature is assumed to be within the nominal operating envelope while the reactor is critical. The LCO on minimum temperature for criticality helps ensure the plant is operated consistent with these assumptions.</p>
	<p>The second consideration is the protective instrumentation. Because certain protective instrumentation (e.g., excore neutron detectors) can be affected by moderator temperature, a temperature value within the nominal operating envelope is chosen to ensure proper indication and response while the reactor is critical.</p>
	<p>The third consideration is the pressurizer operating characteristics. The transient and accident analyses assume that the pressurizer is within its normal startup and operating range (i.e., saturated conditions and steam bubble present). It is also assumed that the RCS temperature is within its normal expected range for startup and power operation. Since the density of the water, and hence the response of the pressurizer to transients, depends upon the initial temperature of the moderator, a minimum value for moderator temperature within the nominal operating envelope is chosen.</p>
	<p>The fourth consideration is that the reactor vessel is above its minimum nil ductility reference temperature when the reactor is critical.</p>
APPLICABLE SAFETY ANALYSES	<p>Although the RCS minimum temperature for criticality is not itself an initial condition assumed in Design Basis Accidents (DBAs), the closely aligned temperature for hot zero power (HZP) is a process variable that is an initial condition of DBAs, such as the rod cluster control assembly (RCCA) withdrawal, RCCA ejection, and main steam line break accidents performed at zero power that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier.</p>

BASES

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APPLICABLE SAFETY ANALYSES (continued)

All low power safety analyses assume initial RCS loop temperatures  $\geq$  the HZP temperature of 547°F (Ref. 1). The minimum temperature for criticality limitation provides a small band, 6°F, for critical operation below HZP. This band allows critical operation below HZP during plant startup and does not adversely affect any safety analyses since the MTC is not significantly affected by the small temperature difference between HZP and the minimum temperature for criticality.

The RCS minimum temperature for criticality satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

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LCO

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

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APPLICABILITY

In MODE 1 and MODE 2 with  $k_{\text{eff}} \geq 1.0$ , LCO 3.4.2 is applicable since the reactor can only be critical ( $k_{\text{eff}} \geq 1.0$ ) in these MODES.

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ACTIONS

A.1

If the parameters that are outside the limit cannot be restored, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 2 with  $k_{\text{eff}} < 1.0$  within 30 minutes. Rapid reactor shutdown can be readily and practically achieved within a 30 minute period. The allowed time is reasonable, based on operating experience, to reach MODE 2 with  $k_{\text{eff}} < 1.0$  in an orderly manner and without challenging plant systems.

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SURVEILLANCE REQUIREMENTS

SR 3.4.2.1

RCS loop average temperature is required to be verified at or above 541°F.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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REFERENCES

1. UFSAR, Section 15.1.
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## B 3.4 REACTOR COOLANT SYSTEM (RCS)

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

#### BASES

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

The PTLR contains P/T limit curves for heatup, cooldown, 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 specific material fracture toughness requirements of the RCPB materials. Reference 2 requires an adequate margin to brittle failure during normal operation, anticipated operational occurrences, and system hydrostatic tests. It mandates the use of the American Society of Mechanical Engineers (ASME) Code, Section III, Appendix G (Ref. 3).

The neutron embrittlement effect on the material toughness is reflected by increasing the nil ductility reference temperature ( $RT_{NDT}$ ) as exposure to neutron fluence increases.

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, as necessary, based on the evaluation findings and the recommendations of Regulatory Guide 1.99 (Ref. 6).

## BASES

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### BACKGROUND (continued)

The P/T limit curves are composite curves established by superimposing limits derived from stress analyses of those portions of the reactor vessel and head that are the most restrictive. At any specific pressure, temperature, and temperature rate of change, one location within the reactor vessel will dictate the most restrictive limit. Across the 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 curve includes the Reference 2 requirement that it be  $\geq 40^{\circ}\text{F}$  above the heatup curve or the cooldown curve, and not less than the minimum permissible temperature for ISLH testing. However, the criticality curve is not operationally limiting; a more restrictive limit exists in LCO 3.4.2, "RCS Minimum Temperature for Criticality."

The 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. 7), provides a recommended methodology for evaluating an operating event that causes an excursion outside the limits.

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### APPLICABLE SAFETY ANALYSES

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

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

## BASES

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### 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 follow:

- a. The severity of the departure from the allowable operating P/T regime or the severity of the rate of change of temperature,
- b. The length of time the limits were violated (longer violations allow the temperature gradient in the thick vessel walls to become more pronounced), and
- c. The existences, sizes, and orientations of flaws in the vessel material.

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### APPLICABILITY

The RCS P/T limits LCO 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

## BASES

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### APPLICABILITY (continued)

maximum pressure. Furthermore, MODES 1 and 2 are above the temperature range of concern for nonductile failure, and stress analyses have been performed for normal maneuvering profiles, such as power ascension or descent.

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### ACTIONS

#### A.1 and A.2

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

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

Besides restoring operation 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 within 72 hours. 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. 7), 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.

## BASES

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### ACTIONS (continued)

#### B.1 and B.2

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

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

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

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

#### C.1 and C.2

Actions must be initiated immediately to correct operation outside of the P/T limits at times other than when in MODE 1, 2, 3, or 4, so that the RCPB is returned to a condition that has been verified by stress analysis.

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

## BASES

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### ACTIONS (continued)

Besides restoring operation within limits, an evaluation is required to determine if RCS operation can continue. The evaluation must verify that the RCPB integrity remains acceptable and must be completed prior to entry into MODE 4. Several methods may be used, including comparison with pre-analyzed transients in the stress analyses, or inspection of the components.

ASME Code, Section XI, Appendix E (Ref. 7), may be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline.

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

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### SURVEILLANCE REQUIREMENTS

#### SR 3.4.3.1

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

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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

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### REFERENCES

1. WCAP-7924-A, April 1975.
  2. 10 CFR 50, Appendix G.
  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. Regulatory Guide 1.99, Revision 2, May 1988.
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BASES

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REFERENCES (continued)

7. ASME, Boiler and Pressure Vessel Code, Section XI, Appendix E.
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## B 3.4 REACTOR COOLANT SYSTEM (RCS)

### B 3.4.4 RCS Loops - MODES 1 and 2

#### BASES

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

The secondary functions of the RCS include:

- Moderating the neutron energy level to the thermal state, to increase the probability of fission,
- Improving the neutron economy by acting as a reflector,
- Carrying the soluble neutron poison, boric acid,
- Providing a second barrier against fission product release to the environment, and
- Removing the heat generated in the fuel due to fission product decay following a unit shutdown.

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

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

Both transient and steady state analyses have been performed to establish the effect of flow on the departure from nucleate boiling (DNB). The transient and accident analyses for the plant have been performed assuming four RCS loops are in operation. The majority of the plant safety analyses are based on initial conditions at high core power or zero power. The accident analyses that are most important to RCP operation are the partial and complete loss of reactor coolant flow, rod withdrawal

BASES

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APPLICABLE SAFETY ANALYSES (continued)

events, startup of an inactive reactor coolant loop, and single RCP locked rotor (Ref. 1).

Steady state DNB analysis has been performed for the four RCS loop operation. These analyses establish typical allowable RCS loop average temperature and  $\Delta T$  for the design power distribution and flow as a function of RCS pressure. These analyses also establish a locus of power, pressure, and temperature conditions for which the departure from nucleate boiling ratio (DNBR) is equal to its Safety Limit value. The area of permissible operation is bounded by the combination of assumed reactor trips for high neutron flux (fixed setpoint), high pressure (fixed setpoint), low pressure (fixed setpoint), overtemperature  $\Delta T$  (variable setpoint), and overpower  $\Delta T$  (variable setpoint). The difference between the reactor trip values assumed in the safety analyses and the nominal reactor trip setpoints provides an allowance for instrumentation channel error and setpoint error.

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

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

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LCO

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

An OPERABLE RCS loop consists of an OPERABLE RCP in operation providing forced flow for heat transport and an associated OPERABLE SG.

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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, and 5.

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BASES

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APPLICABILITY (continued)

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 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level", and
  - LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level".
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ACTIONS

A.1

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

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

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SURVEILLANCE  
REQUIREMENTS

SR 3.4.4.1

This SR requires verification that each RCS loop is in operation. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal while maintaining the margin to DNB.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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REFERENCES

1. UFSAR, Chapter 15.
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## B 3.4 REACTOR COOLANT SYSTEM (RCS)

### B 3.4.5 RCS Loops - MODE 3

#### BASES

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BACKGROUND	<p>In MODE 3, the primary function of the reactor coolant is removal of decay heat and transfer of this heat, via the steam generator (SG), to the secondary plant fluid. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid.</p> <p>The reactor coolant is circulated through four RCS loops, connected in parallel to the reactor vessel, each containing an SG, a reactor coolant pump (RCP), and appropriate flow, pressure, level, and temperature instrumentation for control, protection, and indication. The reactor vessel contains the clad fuel. The SGs provide the heat sink. The RCPs circulate the water through the reactor vessel and SGs at a sufficient rate to ensure proper heat transfer and prevent fuel damage.</p> <p>In MODE 3, RCPs are used to provide forced circulation for heat removal during heatup and cooldown. The MODE 3 decay heat removal requirements are low enough that a single RCS loop with one RCP running is sufficient to remove core decay heat. However, two RCS loops are required to be OPERABLE to ensure redundant capability for decay heat removal.</p>
APPLICABLE SAFETY ANALYSES	<p>Whenever the reactor trip breakers (RTBs) are in the closed position and the control rod drive mechanisms (CRDMs) are energized, an inadvertent rod withdrawal from subcritical, resulting in a power excursion, is possible. Such a transient could be caused by a malfunction of the rod control system. In addition, the possibility of a power excursion due to the ejection of an inserted control rod is possible with the breakers closed or open. Such a transient could be caused by the mechanical failure of a CRDM.</p> <p>Therefore, in MODE 3 with the Rod Control System capable of rod withdrawal, accidental control rod withdrawal from subcritical is postulated and requires at least two RCS loops to be OPERABLE and in operation to ensure that the accident analyses limits are met. For those conditions when the Rod Control System is not capable of rod withdrawal, two RCS loops are required to be OPERABLE, but only one RCS loop is required to be in operation to ensure removal of decay heat from the core and a homogeneous boron concentration throughout the RCS.</p>

BASES

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APPLICABLE SAFETY ANALYSES (continued)

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

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

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LCO

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

When the Rod Control System is not capable of rod withdrawal, only one RCS loop in operation is necessary to ensure removal of decay heat from the core and homogenous boron concentration throughout the RCS. An additional RCS loop is required to be OPERABLE to ensure that safety analyses limits are met.

The Note permits all RCPs to be removed from operation for  $\leq 1$  hour per 8 hour period. The purpose of the Note is to perform tests that are designed to validate various accident analyses values. One of these tests is validation of the pump coastdown curve used as input to a number of accident analyses including a loss of flow accident. This test is generally performed in MODE 3 during the initial startup testing program, and as such should only be performed once. If, however, changes are made to the RCS that would cause a change to the flow characteristics of the RCS, the input values of the coastdown curve must be revalidated by conducting the test again. Another test performed during the startup testing program is the validation of rod drop times during cold conditions, both with and without flow.

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

## BASES

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### LCO (continued)

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

- a. No operations are permitted that would dilute the RCS boron concentration with coolant at boron concentrations less than required to assure the SDM of LCO 3.1.1, thereby maintaining the margin to criticality. Boron reduction with coolant at boron concentrations less than required to assure SDM is maintained is prohibited because a uniform concentration distribution throughout the RCS cannot be ensured when in natural circulation and
- b. Core outlet temperature is maintained at least 10°F below saturation temperature, so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

An OPERABLE RCS loop consists of one OPERABLE RCP and one associated OPERABLE SG, which has the minimum water level specified in SR 3.4.5.2. An RCP is OPERABLE if it is capable of being powered and is able to provide forced flow if required.

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### APPLICABILITY

In MODE 3, this LCO ensures forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. The most stringent condition of the LCO, that is, two RCS loops OPERABLE and two RCS loops in operation, applies to MODE 3 with the Rod Control System capable of rod withdrawal. The least stringent condition, that is, two RCS loops OPERABLE and one RCS loop in operation, applies to MODE 3 with the Rod Control System not capable of rod withdrawal.

Operation in other MODES is covered by:

- LCO 3.4.4, "RCS Loops - MODES 1 and 2;"
- LCO 3.4.6, "RCS Loops - MODE 4;"
- LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled;"
- LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled;"
- LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level"; and
- LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level".

BASES

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ACTIONS

A.1

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

B.1

If restoration for Required Action A.1 is not possible within 72 hours, the unit must be brought to MODE 4. In MODE 4, the unit may be placed on the Residual Heat Removal System. The additional Completion Time of 12 hours is compatible with required operations to achieve cooldown and depressurization from the existing plant conditions in an orderly manner and without challenging plant systems.

C.1

If one required RCS loop is not in operation, and the Rod Control System is capable of rod withdrawal, the Required Action is to place the Rod Control System in a condition incapable of rod withdrawal (e.g., de-energize all CRDMs by opening the RTBs or de-energizing the motor generator (MG) sets). When the Rod Control System is capable of rod withdrawal, it is postulated that a power excursion could occur in the event of an inadvertent control rod withdrawal. This mandates having the heat transfer capacity of two RCS loops in operation. If only one loop is in operation, the Rod Control System must be rendered incapable of rod withdrawal. The Completion Time of 1 hour to defeat the Rod Control System is adequate to perform this operation in an orderly manner without exposing the unit to risk for an undue time period.

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

If two required RCS loops are inoperable or a required RCS loop is not in operation, except as during conditions permitted by the Note in the LCO section, the Rod Control System must be placed in a condition incapable of rod withdrawal (e.g., all CRDMs must be de-energized by opening the RTBs or de-energizing the MG sets). All operations involving

BASES

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ACTIONS (continued)

introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 must be suspended, and action to restore one of the RCS loops to OPERABLE status and operation must be initiated. Boron dilution requires forced circulation for proper mixing, and opening the RTBs or de-energizing the MG sets removes the possibility of an inadvertent rod withdrawal. Suspending the introduction of coolant into the RCS of coolant with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 is required to assure continued safe operation. With coolant added without forced circulation, unmixed coolant could be introduced to the core, however coolant added with boron concentration meeting the minimum SDM maintains acceptable margin to subcritical operations. The immediate Completion Time reflects the importance of maintaining operation for heat removal. The action to restore must be continued until one loop is restored to OPERABLE status and operation.

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SURVEILLANCE  
REQUIREMENTS

SR 3.4.5.1

This SR requires verification that the required loops are in operation. Verification includes flow rate, temperature, and pump status monitoring, which help ensure that forced flow is providing heat removal.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.4.5.2

SR 3.4.5.2 requires verification of SG OPERABILITY. SG OPERABILITY is verified by ensuring that the secondary side narrow range water level is  $\geq 21\%$  for required RCS loops. If the SG secondary side narrow range water level is  $< 21\%$  the tubes may become uncovered and the associated loop may not be capable of providing the heat sink for removal of the decay heat.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.5.3

Verification that each required RCP is OPERABLE ensures that safety analyses limits are met. The requirement also ensures that an additional RCP can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power availability to each required RCP. Alternatively, verification that a pump is in operation also verifies proper breaker alignment and power availability.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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

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REFERENCES      None.

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## B 3.4 REACTOR COOLANT SYSTEM (RCS)

### B 3.4.6 RCS Loops - MODE 4

#### BASES

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

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

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

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

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

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**LCO** The purpose of this LCO is to require that at least two loops be OPERABLE in MODE 4 and that one of these loops be in operation. The LCO allows the two loops that are required to be OPERABLE to consist of any combination of RCS loops and RHR loops. Any one loop in operation provides enough flow to remove the decay heat from the core with forced circulation. An additional loop is required to be OPERABLE to provide redundancy for heat removal.

Note 1 permits all RCPs or RHR pumps to be removed from operation for  $\leq 1$  hour per 8 hour period. The purpose of the Note is to permit tests that are designed to validate various accident analyses values. One of the tests performed during the startup testing program is the validation of rod drop times during cold conditions, both with and without flow. The no flow test may be performed in MODE 3, 4, or 5 and requires that the pumps be

## BASES

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### LCO (continued)

stopped for a short period of time. The Note permits the stopping of the pumps in order to perform this test and validate the assumed analysis values. If changes are made to the RCS that would cause a change to the flow characteristics of the RCS, the input values must be revalidated by conducting the test again. The 1 hour time period is adequate to perform the test, and operating experience has shown that boron stratification is not a problem during this short period with no forced flow.

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

- a. No operations are permitted that would dilute the RCS boron concentration with coolant with boron concentrations less than required to meet SDM of LCO 3.1.1, "SHUTDOWN MARGIN", thereby maintaining the margin to criticality. Boron reduction with coolant at boron concentrations less than required to assure SDM is maintained is prohibited because a uniform concentration distribution throughout the RCS cannot be ensured when in natural circulation; and
- b. Core outlet temperature is maintained at least 10°F below saturation temperature, so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

Note 2 requires that a steam bubble exists in the pressurizer prior to starting any RCP.

An OPERABLE RCS loop comprises an OPERABLE RCP and an associated OPERABLE SG, which has the minimum water level specified in SR 3.4.6.2.

Similarly for the RHR System, an OPERABLE RHR loop (either A or B) comprises an OPERABLE RHR pump capable of providing forced flow to an OPERABLE RHR heat exchanger. RCPs and RHR pumps are OPERABLE if they are capable of being powered and are able to provide forced flow if required.

## BASES

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

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 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level"; and

LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level".

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## ACTIONS

### A.1

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

### A.2

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

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

BASES

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ACTIONS (continued)

B.1 and B.2

If two required loops are inoperable or a required loop is not in operation, except during conditions permitted by Note 1 in the LCO section, all operations involving introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 must be suspended and action to restore one RCS or RHR loop to OPERABLE status and operation must be initiated. The required margin to criticality must not be reduced in this type of operation. Suspending operations that would cause the introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 is required to assure continued safe operation. With coolant added without forced circulation, unmixed coolant could be introduced to the core, however coolant added with boron concentration meeting the minimum SDM maintains acceptable margin to subcritical operations. The immediate Completion Times reflect the importance of maintaining operation for decay heat removal. The action to restore must be continued until one loop is restored to OPERABLE status and operation.

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SURVEILLANCE  
REQUIREMENTS

SR 3.4.6.1

This SR requires verification that the required RCS or RHR loop is in operation and circulating reactor coolant. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.4.6.2

SR 3.4.6.2 requires verification of SG OPERABILITY. SG OPERABILITY is verified by ensuring that the secondary side water level is  $\geq 21\%$  (narrow range indication). If the SG secondary side water level is  $< 21\%$  (narrow range indication), the tubes may become uncovered and the associated loop may not be capable of providing the heat sink necessary for removal of decay heat.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.6.3

Verification that each required pump is OPERABLE ensures that an additional RCS or RHR 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 each required pump. Alternatively, verification that a pump is in operation also verifies proper breaker alignment and power availability.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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

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REFERENCES      None.

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## B 3.4 REACTOR COOLANT SYSTEM (RCS)

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

#### BASES

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**BACKGROUND** In MODE 5 with the RCS loops filled, the primary function of the reactor coolant is the removal of decay heat and transfer of this heat either to the steam generator (SG) secondary side coolant via natural circulation (Ref. 1) or the component cooling water via the residual heat removal (RHR) heat exchangers. While the principal means for decay heat removal is via the RHR System, the SGs via natural circulation (Ref. 1) are specified as a backup means for redundancy. Even though the SGs cannot produce steam in this MODE, they are capable of being a heat sink due to their large contained volume of secondary water. As long as the SG 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 reactor coolant is circulated by means of two RHR loops connected to the RCS, each loop containing an RHR heat exchanger, an RHR pump, and appropriate flow and temperature instrumentation for control, protection, and indication. One RHR pump circulates the water through the RCS at a sufficient rate to prevent boric acid stratification.

The number of loops in operation can vary to suit the operational needs. The intent of this LCO is to provide forced flow from at least one RHR loop for decay heat removal and transport. The flow provided by one RHR loop is adequate for decay heat removal. The other intent of this LCO is to require that a second path be OPERABLE to provide redundancy for heat removal.

The LCO provides for redundant paths of decay heat removal capability. The first path is an RHR loop that must be OPERABLE and in operation. The second path can be another OPERABLE RHR loop or maintaining two SGs with secondary side water levels  $\geq 21\%$  (narrow range indication) to provide an alternate method for decay heat removal via natural circulation (Ref. 1).

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

RCS Loops - MODE 5 (Loops Filled) satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

BASES

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LCO

The purpose of this LCO is to require that at least one of the RHR loops be OPERABLE and in operation with an additional RHR loop OPERABLE or two SGs with secondary side water level  $\geq 21\%$  (narrow range indication). One RHR loop provides sufficient forced circulation to perform the safety functions of the reactor coolant under these conditions. An additional RHR loop is required to be OPERABLE to meet single failure considerations. However, if the standby RHR loop is not OPERABLE, an acceptable alternate method is two SGs with their secondary side water levels  $\geq 21\%$  (narrow range indication). Should the operating RHR loop fail, the SGs could be used to remove the decay heat via natural circulation.

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

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

- a. No operations are permitted that would dilute the RCS boron concentration with coolant with boron concentrations less than required to meet SDM of LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," thereby maintaining the margin to criticality. Boron reduction with coolant at boron concentrations less than required to assure SDM is maintained is prohibited because a uniform concentration distribution throughout the RCS cannot be ensured when in natural circulation; and
- b. Core outlet temperature is maintained at least  $10^{\circ}\text{F}$  below saturation temperature, so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

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

BASES

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LCO (continued)

Note 3 requires that a steam bubble exists in the pressurizer or that the secondary side water temperature of each SG be  $\leq 25^{\circ}\text{F}$  above each of the RCS cold leg temperatures before the start of a reactor coolant pump (RCP). This restriction is to prevent a low temperature overpressure event due to a thermal transient when an RCP is started.

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

RHR pumps are OPERABLE if they are capable of being powered and are able to provide flow if required. A SG can perform as a heat sink via natural circulation when it has an adequate water level and is OPERABLE.

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APPLICABILITY

In MODE 5 with RCS loops filled, this LCO requires forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. One loop of RHR provides sufficient circulation for these purposes. However, one additional RHR loop is required to be OPERABLE, or the secondary side water level of at least two SGs is required to be  $\geq 21\%$  (narrow range indication).

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 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level"; and
  - LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level".
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ACTIONS

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

If one RHR loop is OPERABLE and either the required SGs have secondary side water levels  $< 21\%$  (narrow range indication), or one required RHR loop is inoperable, redundancy for heat removal is lost. Action must be initiated immediately to restore a second RHR loop to OPERABLE status or to restore the required SG secondary side water levels. Either Required Action will restore redundant heat removal paths. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal.

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BASES

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ACTIONS

C.1 and C.2

If a required RHR loop is not in operation, except during conditions permitted by Note 1, or if no required loop is OPERABLE, all operations involving introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 must be suspended and action to restore one RHR loop to OPERABLE status and operation must be initiated. Suspending the introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 is required to assure continued safe operation. With coolant added without forced circulation, unmixed coolant could be introduced to the core, however coolant added with boron concentration meeting the minimum SDM maintains acceptable margin to subcritical operations. The immediate Completion Times reflect the importance of maintaining operation for heat removal.

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SURVEILLANCE  
REQUIREMENTS

SR 3.4.7.1

This SR requires verification that the required loop is in operation and circulating reactor coolant. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.4.7.2

Verifying that at least two SGs are OPERABLE by ensuring their secondary side water levels are  $\geq 21\%$  (narrow range indication) ensures an alternate decay heat removal method via natural circulation in the event that the second RHR loop is not OPERABLE. If both RHR loops are OPERABLE, this Surveillance is not needed.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.7.3

Verification that each required RHR pump is OPERABLE ensures that an additional 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 each required RHR pump. Alternatively, verification that a pump is in operation also verifies proper breaker alignment and power availability. If secondary side water level is  $\geq 21\%$  (narrow range indication) in at least two SGs, this Surveillance is not needed.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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

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REFERENCES

1. NRC Information Notice 95-35, "Degraded Ability of Steam Generators to Remove Decay Heat by Natural Circulation."
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## B 3.4 REACTOR COOLANT SYSTEM (RCS)

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

#### BASES

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**BACKGROUND** In MODE 5 with the RCS loops not filled, the primary function of the reactor coolant is the removal of decay heat generated in the fuel, and the transfer of this heat to the component cooling water via the residual heat removal (RHR) heat exchangers. The steam generators (SGs) are not available as a heat sink when the loops are not filled. The secondary function of the reactor coolant is to act as a carrier for the soluble neutron poison, boric acid.

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

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

RCS loops in MODE 5 (loops not filled) satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

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

Note 1 permits all RHR pumps to be removed from operation for  $\leq 15$  minutes when switching from one loop to another. The circumstances for stopping both RHR pumps are to be limited to situations when the outage time is short and core outlet temperature is maintained  $\geq 10^\circ\text{F}$  below saturation temperature. The Note prohibits boron dilution with coolant at boron concentrations less than required to assure SDM of LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," is maintained or draining operations when RHR forced flow is stopped.

BASES

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LCO (continued)

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

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

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APPLICABILITY

In MODE 5 with loops not filled, this LCO requires core heat removal and coolant circulation by the RHR 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 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level"; and

LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level".

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ACTIONS

A.1

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

B.1 and B.2

If no required loop is OPERABLE or the required loop is not in operation, except during conditions permitted by Note 1, all operations involving introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 must be suspended and action must be initiated immediately to restore an RHR loop to OPERABLE status and operation. The required margin to criticality must not be reduced in this type of operation. Suspending the introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 is required to assure continued safe operation. With coolant added without forced circulation, unmixed coolant could be introduced to the core, however coolant added with boron concentration meeting the minimum SDM maintains acceptable margin to

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BASES

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ACTIONS (continued)

subcritical operations. The immediate Completion Time reflects the importance of maintaining operation for heat removal. The action to restore must continue until one loop is restored to OPERABLE status and operation.

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SURVEILLANCE  
REQUIREMENTS

SR 3.4.8.1

This SR requires verification that the required loop is in operation and circulating reactor coolant. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.4.8.2

Verification that each required pump is OPERABLE ensures that an additional 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 each required pump. Alternatively, verification that a pump is in operation also verifies proper breaker alignment and power availability.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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

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REFERENCES

None.

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## B 3.4 REACTOR COOLANT SYSTEM (RCS)

### B 3.4.9 Pressurizer

#### BASES

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

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

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

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

## BASES

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### APPLICABLE SAFETY ANALYSES

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

Safety analyses presented in the UFSAR (Ref. 1) do not take credit for pressurizer heater operation; however, an implicit initial condition assumption of the safety analyses is that the RCS is operating at normal pressure.

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

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### LCO

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

The LCO requires two groups of OPERABLE pressurizer heaters, each with a capacity  $\geq 150$  kW. The minimum heater capacity required is sufficient to provide assurance that the heaters can be energized during a loss of offsite power condition to provide adequate subcooling margin in the RCS to maintain natural circulation conditions in MODE 3.

## BASES

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**APPLICABILITY** The need for pressure control is most pertinent when core heat can cause the greatest effect on RCS temperature, resulting in the greatest effect on pressurizer level and RCS pressure control. Thus, applicability has been designated for MODES 1 and 2. The applicability is also provided for MODE 3. 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.

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

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**ACTIONS** A.1, A.2, A.3, and A.4

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

If the pressurizer water level is not within the limit, action must be taken to bring the plant to a MODE in which the LCO does not apply. To achieve this status, within 6 hours the unit must be brought to MODE 3 with all rods fully inserted and incapable of withdrawal. Additionally, the unit must be brought to MODE 4 within 12 hours. This takes the unit out of the applicable MODES.

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.

### 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 the anticipation that a demand caused by loss of offsite power would be unlikely in this period. Pressure control may be maintained during this time using pressurizer control heaters.

## BASES

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### ACTIONS (continued)

#### C.1 and C.2

If one group of pressurizer heaters are inoperable and cannot be restored in 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 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

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### SURVEILLANCE REQUIREMENTS

#### SR 3.4.9.1

This SR requires that during steady state operation, pressurizer level is maintained below the nominal upper limit to provide a minimum space for a steam bubble. The Surveillance is performed by observing the indicated level on the narrow range instrumentation.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

#### SR 3.4.9.2

The SR 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 specified capacity. This may be done by measuring circuit current and voltage to calculate kW capacity.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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### REFERENCES

1. UFSAR, Section 15.1.
  2. NUREG-0737, November 1980.
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## B 3.4 REACTOR COOLANT SYSTEM (RCS)

### B 3.4.10 Pressurizer Safety Valves

#### BASES

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

Because the safety valves are totally enclosed and self actuating, they are considered independent components. The relief capacity for each valve, 420,000 lb/hr, is based on postulated overpressure transient conditions resulting from a complete loss of steam flow to the turbine. This event results in the maximum surge rate into the pressurizer, which specifies the minimum relief capacity for the safety valves. The discharge flow from the pressurizer safety valves is directed to the pressurizer relief tank. This discharge flow is indicated by an increase in temperature downstream of the pressurizer safety valves or increase in the pressurizer relief tank temperature or level.

Overpressure protection is required in MODES 1, 2, 3, 4, and 5; however, in MODE 4, MODE 5, and MODE 6 with the reactor vessel head on, overpressure protection is provided by operating procedures and by meeting the requirements of LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."

The upper and lower pressure limits are based on the  $\pm 3\%$  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 (nominal operating temperature and pressure). 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 American Society of Mechanical Engineers (ASME) pressure limit (Ref. 1) could include damage to RCS components, increased leakage, or a requirement to perform additional stress analyses prior to resumption of reactor operation.

BASES

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APPLICABLE  
SAFETY  
ANALYSES

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

- a. Uncontrolled rod withdrawal from full power,
- b. Loss of reactor coolant flow,
- c. Loss of external electrical load,
- d. Loss of normal feedwater,
- e. Loss of all AC power to station auxiliaries, and
- f. Locked rotor.

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

Pressurizer safety valves satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

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LCO

The three pressurizer safety valves are set to open at the RCS design pressure (2485 psig), and within the ASME specified tolerance, to avoid exceeding the maximum design pressure SL, to maintain accident analyses assumptions, and to comply with ASME requirements. The upper and lower pressure tolerance limits are based on the  $\pm 3\%$  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 more valves could result in exceeding the SL if a transient were to occur. The consequences of exceeding the ASME pressure limit could include damage to one or more RCS components, increased leakage, or additional stress analysis being required prior to resumption of reactor operation.

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APPLICABILITY

In MODES 1, 2, and 3 OPERABILITY of three 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 is conservatively included, although the listed accidents may not require the safety valves for protection.

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## BASES

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### APPLICABILITY (continued)

The LCO is not applicable in MODE 4 or in MODE 5 because LTOP is provided. Overpressure protection is not required in MODE 6 with reactor vessel head detensioned.

The Note allows entry into MODE 3 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 54 hour exception is based on 18 hour outage time for each of the three valves. The 18 hour period is derived from operating experience that hot testing can be performed in this timeframe.

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### ACTIONS

#### A.1

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

#### B.1 and B.2

If the Required Action of A.1 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 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. In MODE 4, overpressure protection is provided by the LTOP System. The change from MODE 1, 2, or 3 to MODE 4 reduces the RCS energy (core power and pressure), lowers the potential for large pressurizer insurges, and thereby removes the need for overpressure protection by three pressurizer safety valves.

BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.4.10.1

SRs are specified in the Inservice Testing Program. Pressurizer safety valves are to be tested in accordance with the requirements of the ASME Code (Ref. 4), which provides the activities and Frequencies necessary to satisfy the SRs. No additional requirements are specified.

The pressurizer safety valve setpoint is  $\pm 3\%$  of 2485 psig for OPERABILITY; however, the valves are reset to  $\pm 1\%$  during the Surveillance to allow for drift.

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REFERENCES

1. ASME, Boiler and Pressure Vessel Code, Section III.
  2. UFSAR, Chapter 15.
  3. WCAP-7769, Rev. 1, June 1972.
  4. ASME Code for Operation and Maintenance of Nuclear Power Plants.
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## B 3.4 REACTOR COOLANT SYSTEM (RCS)

### B 3.4.11 Pressurizer Power Operated Relief Valves (PORVs)

#### BASES

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

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

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

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

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

The plant has two PORVs, each having a relief capacity of 210,000 lb/hr at 2335 psig. The functional design of the PORVs is based on maintaining pressure below the Pressurizer Pressure - High reactor trip setpoint following a step reduction of 50% of full load with steam dump. In addition, the PORVs minimize challenges to the pressurizer safety valves and also may be used for low temperature overpressure protection (LTOP). See LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."

BASES

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APPLICABLE  
SAFETY  
ANALYSES

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

The PORVs are also modeled in safety analyses for events that result in increasing RCS pressure for which departure from nucleate boiling ratio (DNBR) criteria are critical (Ref. 2). By assuming PORV actuation, the primary pressure remains below the high pressurizer pressure trip setpoint; thus, the DNBR calculation is more conservative. As such, this actuation is not required to mitigate these events, and PORV automatic operation is, therefore, not an assumed safety function.

Pressurizer PORVs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

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LCO

The LCO requires the PORVs and their associated block valves to be OPERABLE for manual operation to mitigate the effects associated with an SGTR.

By maintaining two PORVs and their associated block valves OPERABLE, the single failure criterion is satisfied. An OPERABLE block valve may be either open and energized with the capability to be closed, or closed and energized with the capability to be opened, since the required safety function is accomplished by manual operation. Although typically open to allow PORV operation, the block valves may be OPERABLE when closed to isolate the flow path of an inoperable PORV that is capable of being manually cycled (e.g., as in the case of excessive PORV leakage). Similarly, isolation of an OPERABLE PORV does not render that PORV or block valve inoperable provided the relief function remains available with manual action.

An OPERABLE PORV is required to be capable of manually opening and closing, and not experiencing excessive seat leakage. Excessive seat leakage, although not associated with a specific acceptance criteria, exists when conditions dictate closure of the block valve to limit leakage.

Satisfying the LCO helps minimize challenges to fission product barriers.

## BASES

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**APPLICABILITY** In MODES 1, 2, and 3, the PORV and its block valve are required to be OPERABLE to limit the potential for a small break LOCA through the flow path. The most likely cause for a PORV small break LOCA is a result of a pressure increase transient that causes the PORV to open. Imbalances in the energy output of the core and heat removal by the secondary system can cause the RCS pressure to increase to the PORV opening setpoint. The most rapid increases will occur at the higher operating power and pressure conditions of MODES 1 and 2. The PORVs are also required to be OPERABLE in MODES 1, 2, and 3 for manual actuation to mitigate a steam generator tube rupture event.

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

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**ACTIONS** A Note has been added to clarify that all pressurizer PORVs and block valves are treated as separate entities, each with separate Completion Times (i.e., the Completion Time is on a component basis).

### A.1

PORVs may be inoperable and capable of being manually cycled (e.g., excessive seat leakage). In this condition, the associated block valve is required to be closed within 1 hour, but power must be maintained to the associated block valve, since removal of power would render the block valve inoperable.

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 has shown that minor problems can be corrected or closure accomplished in this time period.

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

If one PORV is inoperable and not capable of being manually cycled, it must be either restored, or isolated by closing the associated block valve and removing the power to the associated block valve. The Completion Times of 1 hour are reasonable, based on challenges to the PORVs during this time period, and provide 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

## BASES

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### ACTIONS (continued)

least one PORV that remains OPERABLE, an additional 72 hours is provided to restore the inoperable PORV to OPERABLE status. If the PORV cannot be restored within this additional time, the plant must be brought to a MODE in which the LCO does not apply, as required by Condition D.

#### C.1 and C.2

If one block valve is inoperable, then it is necessary to either restore the block valve(s) to OPERABLE status within the Completion Time of 1 hour or place the associated PORV in manual control. The prime importance for the capability to close the block valve is to isolate a stuck open PORV. Therefore, if the block valve(s) cannot be restored to OPERABLE status within 1 hour, the Required Action is to place the PORV in manual control to preclude its automatic opening for an overpressure event and to avoid the potential for a stuck open PORV at a time that the block valve is inoperable. The Completion Time of 1 hour is reasonable, based on the small potential for challenges to the system during this time period, and provides the operator time to correct the situation. Because at least one PORV remains OPERABLE, the operator is permitted 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 may not be capable of mitigating an event if the inoperable block valve is not full open. If the block valve is restored within the Completion Time of 72 hours, the PORV may be restored to automatic operation. If it cannot be restored within this additional time, the plant must be brought to a MODE in which the LCO does not apply, as required by Condition D.

The Required Actions C.1 and C.2 are modified by a Note stating that the Required Actions do not apply if the sole reason for the block valve being declared inoperable is as a result of power being removed to comply with other Required Actions. In this event, the Required Actions for inoperable PORV(s) (which require the block valve power to be removed once it is closed) are adequate to address the condition. While it may be desirable to also place the PORV(s) in manual control, this may not be possible for all causes of Condition B or E entry with PORV(s) inoperable and not capable of being manually cycled (e.g., as a result of failed control power fuse(s) or control switch malfunctions(s)).

## BASES

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### ACTIONS (continued)

#### D.1 and D.2

If the Required Action of Condition A, B, or C is not met, 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 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. In MODES 4 and 5, automatic PORV OPERABILITY may be required. See LCO 3.4.12.

#### E.1, E.2, E.3, and E.4

If two PORVs are 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 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 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. In MODES 4 and 5, automatic PORV OPERABILITY may be required. See LCO 3.4.12.

#### F.1

If two block valve(s) are inoperable, it is necessary to restore at least one block valve within 1 hour. The Completion Time is reasonable, based on the small potential for challenges to the system during this time and provides the operator time to correct the situation.

Required Action F.1 is modified by a Note stating that the Required Action does not apply if the sole reason for the block valve being declared inoperable is a result of power being removed to comply with other Required Actions. In this event, the Required Actions for inoperable PORV(s) (which require the block valve power to be removed once it is closed) are adequate to address the condition. While it may be desirable to also place the PORV(s) in manual control, this may not be possible for all causes of Condition B or E entry with PORV(s) inoperable and not

BASES

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ACTIONS (continued)

capable of being manually cycled (e.g., as a result of failed control power fuse(s) or control switch malfunctions(s)).

G.1 and G.2

If the Required Action of Condition F is not met, 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 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. In MODES 4 and 5, automatic PORV OPERABILITY may be required. See LCO 3.4.12.

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SURVEILLANCE  
REQUIREMENTS

SR 3.4.11.1

Block valve cycling verifies that the valve(s) can be opened and closed if needed.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

This SR is modified by two Notes. Note 1 modifies this SR by stating that it is not required to be performed with the block valve closed in accordance with the Required Actions of this LCO. Opening the block valve in this condition increases the risk of an unisolable leak from the RCS since the PORV is already inoperable. Note 2 modifies this SR to allow entry into and operation in MODE 3 prior to performing the SR. This allows the test to be performed in MODE 3 under operating temperature and pressure conditions, prior to entering MODE 1 or 2. In accordance with Reference 4, administrative controls require this test be performed in MODE 3 or 4 to adequately simulate operating temperature and pressure effects on PORV operation.

SR 3.4.11.2

SR 3.4.11.2 requires a complete cycle of each PORV. Operating a PORV through one complete cycle ensures that the PORV can be manually actuated for mitigation of an SGTR.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

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SURVEILLANCE REQUIREMENTS (continued)

The Note modifies this SR to allow entry into and operation in MODE 3 prior to performing the SR. This allows the test to be performed in MODE 3 under operating temperature and pressure conditions, prior to entering MODE 1 or 2. In accordance with Reference 4, administrative controls require this test be performed in MODE 3, 4, or 5 with a steam bubble in the pressurizer to adequately simulate operating temperature and pressure effects on PORV operation.

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REFERENCES

1. Regulatory Guide 1.32, February 1977.
  2. UFSAR, Section 15.2.
  3. ASME Code for Operation and Maintenance of Nuclear Power Plants.
  4. Letter from R.W. Hernan (NRC Staff) to J.A. Scalice (TVA), "Issuance of Technical Specification Amendments for Sequoyah Nuclear Plant, Units 1 and 2 (TAC Nos. MA2164 and MA2169) (TS 98-01)," dated November 19, 1998 (ADAMS Accession No. ML013320468).
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## B 3.4 REACTOR COOLANT SYSTEM (RCS)

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

#### BASES

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**BACKGROUND** 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. The PTLR provides the maximum allowable actuation logic setpoints for the power operated relief valves (PORVs) and the maximum RCS pressure for the existing RCS cold leg temperature during cooldown, shutdown, and heatup to meet the Reference 1 requirements during the LTOP MODES.

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

The potential for vessel overpressurization is most acute when the RCS is water solid, occurring only 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, "RCS Pressure and Temperature (P/T) Limits," requires administrative control of RCS pressure and temperature during heatup and cooldown to prevent exceeding the PTLR limits.

This LCO provides RCS overpressure protection by having a minimum coolant input capability and having adequate pressure relief capacity. Limiting coolant input capability requires all safety injection pumps and all but one centrifugal charging pump incapable of injection into the RCS and isolating the accumulators. The pressure relief capacity requires either two redundant PORVs or a depressurized RCS and an RCS vent of sufficient size. One PORV or the open RCS vent is the overpressure protection device that acts to terminate an increasing pressure event.

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

## BASES

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### BACKGROUND (continued)

decay heat levels, the makeup system can provide adequate flow via the makeup control valve. If conditions require the use of more than one 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 a depressurized RCS and an RCS vent of sufficient size. Two PORVs are required for redundancy. One PORV has adequate relieving capability to keep from overpressurization for the required coolant input capability.

#### PORV Requirements

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

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

The PTLR 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 in the PTLR ensures that the Reference 1 limits will not be exceeded in any analyzed event.

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

BASES

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BACKGROUND (continued)

RCS Vent Requirements

Once the RCS is depressurized, a vent exposed to the containment atmosphere will maintain the RCS at 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 flow capacity requirement, it requires an RCS vent opening of a least three square inches. This may be accomplished by 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.

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APPLICABLE  
SAFETY  
ANALYSES

Safety analyses (Ref. 3) demonstrate that the reactor vessel is adequately protected against exceeding the Reference 1 P/T limits. In MODES 1, 2, and 3 the pressurizer safety valves will prevent RCS pressure from exceeding the Reference 1 limits. At about the LTOP arming temperature specified in the PTLR and below, overpressure prevention falls to two 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 PTLR curves are revised, the LTOP System must be re-evaluated to ensure its functional requirements can still be met using the PORV method or the depressurized and vented RCS condition.

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

BASES

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

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

- a. Rendering all safety injection pumps and all but one charging pump incapable of injection,
- b. Deactivating the accumulator discharge isolation valves in their closed positions, and
- c. Disallowing start of an RCP unless a steam bubble exists in the pressurizer or the secondary side water temperature of each SG is  $\leq 25^{\circ}\text{F}$  above each of the RCS cold leg temperatures. LCO 3.4.6, "RCS Loops – MODE 4," and LCO 3.4.7, "RCS Loops – MODE 5, Loops Filled," provides this protection.

## BASES

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### APPLICABLE SAFETY ANALYSES (continued)

The Reference 3 analyses demonstrate that either one PORV or the depressurized RCS and RCS vent can maintain RCS pressure below limits when only one charging pump is actuated. Thus, the LCO allows only one charging pump OPERABLE during the LTOP MODES. Since neither one PORV nor the RCS vent can handle the pressure transient need from accumulator injection, when RCS temperature is low, the LCO also requires the accumulator's 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 accumulators must have their discharge valves closed and the valve power supply breakers fixed in their open positions.

Fracture mechanics analyses established the temperature of LTOP Applicability at the LTOP arming temperature specified in the PTLR.

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 charging pump OPERABLE and SI actuation available.

#### PORV Performance

The fracture mechanics analyses show that the vessel is protected when the PORVs are set to open at or below the limit shown in the PTLR. The setpoints are derived by analyses that model the performance of the LTOP System, assuming the limiting LTOP transient of one charging pump injecting into the RCS. These analyses consider pressure overshoot and undershoot beyond the PORV opening and closing, resulting from signal processing and valve stroke times. The PORV setpoints at or below the derived limit ensures the Reference 1 P/T limits will be met.

The PORV setpoints in the PTLR will be updated when the revised P/T limits conflict with the LTOP analysis limits. The P/T limits are periodically modified as the reactor vessel material toughness decreases due to neutron embrittlement caused by neutron irradiation. Revised 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 is assumed to represent the worst case, single active failure.

BASES

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APPLICABLE SAFETY ANALYSES (continued)

RCS Vent Performance

With the RCS depressurized, analyses show a vent size of 3.0 square inches is capable of mitigating the allowed LTOP overpressure transient. The capacity of a vent this size is greater than the flow of the limiting transient for the LTOP configuration, one charging pump OPERABLE, maintaining RCS pressure less than the maximum pressure on the P/T limit curve.

The RCS vent size will 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.

The LTOP System satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

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LCO

This LCO requires 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 that no safety injection pumps and a maximum of one charging pump be capable of injecting into the RCS, and all accumulator discharge isolation valves be closed and immobilized (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 LCO is modified by three Notes. Note 1 allows two charging pumps to be made capable of injecting for  $\leq 1$  hour during pump swap operations. One hour provides sufficient time to safely complete the actual transfer and to complete the administrative controls and Surveillance Requirements associated with the swap. The intent is to minimize the actual time that more than one charging pump is physically capable of injection. Note 2 states that accumulator isolation is only required when the accumulator pressure is more than or at the maximum RCS pressure for the existing temperature, as allowed by the P/T limit curves. This Note permits the accumulator discharge isolation valve Surveillance to be performed only under these pressure and temperature conditions. Note 3 allows a 4 hour maximum time period for rendering both safety injection and one centrifugal charging pump inoperable after entry in MODE 4 from MODE 3. RCS temperature must remain above 325°F until the pumps are rendered incapable of inadvertent injection.

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## BASES

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### LCO (continued)

The 4 hour time period is sufficient for completing this activity and is based on low probability for inadvertent pump start.

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

- a. Two OPERABLE PORVs,

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

- b. A depressurized RCS and an RCS vent.

An RCS vent is OPERABLE when open with an area of  $\geq 3.0$  square inches.

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

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### APPLICABILITY

This LCO is applicable in MODE 4 when any RCS cold leg temperature is  $\leq$  the LTOP arming temperature specified in the PTLR, 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 the LTOP arming temperature specified in the PTLR. 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.

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

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### ACTIONS

A Note prohibits the application of LCO 3.0.4.b to an inoperable LTOP System. There is an increased risk associated with entering MODE 4 from MODE 5 with LTOP inoperable and the provisions of LCO 3.0.4.b, which allow entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, should not be applied in this circumstance.

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## BASES

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### ACTIONS (continued)

#### A.1 and B.1

With any safety injection pump or more than one charging pump capable of injecting into the RCS, RCS overpressurization is possible.

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

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

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

If isolation is needed and cannot be accomplished in 1 hour, Required Action D.1 and Required Action D.2 provide two options, either of which must be performed in the next 12 hours. By increasing the RCS temperature to > the LTOP arming temperature specified in the PTLR, an accumulator pressure of 600 psig cannot exceed the LTOP limits if the accumulators are fully injected. Depressurizing the accumulators below the LTOP limit from the PTLR also gives this protection.

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

#### E.1

In MODE 4 when any RCS cold leg temperature is  $\leq$  the LTOP arming temperature specified in the PTLR, with one required PORV inoperable, the PORV must be restored to OPERABLE status within a Completion Time of 7 days. Two PORVs are required to provide low temperature overpressure mitigation while withstanding a single failure of an active component.

The Completion Time considers the facts that only one of the PORVs 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.

## BASES

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### ACTIONS (continued)

#### F.1

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

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

#### G.1

The RCS must be depressurized and a vent must be established within 12 hours when:

- a. Both required PORVs are inoperable,
- b. A Required Action and associated Completion Time of Condition A, B, D, E, or F is not met, or
- c. The LTOP System is inoperable for any reason other than Condition A, B, C, D, E, or F.

The vent must be sized  $\geq 3.0$  square inches to ensure that the flow capacity is greater than that required for the worst case mass input transient reasonable during the applicable MODES. This action is needed to protect the RCPB from a low temperature overpressure event and a possible brittle failure of the reactor vessel.

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

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### 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, no safety injection pumps and a maximum of one charging pump are verified incapable of injecting into the RCS and the accumulator discharge isolation valves are verified closed and locked out.

## BASES

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### SURVEILLANCE REQUIREMENTS (continued)

The SI pumps and charging pump are rendered incapable 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 Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

The additional frequency for SR 3.4.12.1 and SR 3.4.12.2 is necessary to allow time during the transition from MODE 3 to MODE 4 to make the pumps inoperable.

#### SR 3.4.12.4

The RCS vent of  $\geq 3.0$  square inches is proven OPERABLE by verifying its open condition.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

The passive vent path arrangement must only be open to be OPERABLE. This Surveillance is required to be met if the vent is being used to satisfy the pressure relief requirements of the LCO 3.4.12b.

#### SR 3.4.12.5

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

The block valve is a remotely controlled, motor operated valve. The power to the valve operator is not required removed, and the manual operator is not required locked in the inactive position. Thus, the block valve can be closed in the event the PORV develops excessive leakage or does not close (sticks open) after relieving an overpressure situation.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.12.6

Performance of a COT is required within 12 hours after decreasing RCS temperature to  $\leq$  the LTOP arming temperature specified in the PTLR on each required PORV to verify and, as necessary, adjust its lift setpoint. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable COT of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. The COT will verify the setpoint is within the PTLR allowed maximum limits in the PTLR. PORV actuation could depressurize the RCS and is not required.

The 12 hour Frequency considers the unlikelihood of a low temperature overpressure event during this time.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

A Note has been added indicating that this SR is required to be performed 12 hours after decreasing RCS cold leg temperature to  $\leq$  the LTOP arming temperature specified in the PTLR. The COT cannot be performed until in the LTOP MODES when the PORV lift setpoint can be reduced to the LTOP setting. The test must be performed within 12 hours after entering the LTOP MODES.

SR 3.4.12.7

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

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

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- REFERENCES
1. 10 CFR 50, Appendix G.
  2. Generic Letter 88-11.
  3. UFSAR, Chapter 15.
  4. 10 CFR 50, Section 50.46.
  5. 10 CFR 50, Appendix K.
  6. Generic Letter 90-06.
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## B 3.4 REACTOR COOLANT SYSTEM (RCS)

### B 3.4.13 RCS Operational LEAKAGE

#### BASES

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

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

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

The safety significance of RCS LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring reactor coolant LEAKAGE into the containment area 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 that is 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 analyses radiation release assumptions from being exceeded. The consequences of violating this LCO include the possibility of a loss of coolant accident (LOCA).

BASES

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APPLICABLE  
SAFETY  
ANALYSES

Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event. The safety analysis for an event resulting in steam discharge to the atmosphere assumes 150 gallons per day (gpd) per steam generator (i.e., a total of 0.4 gpm).

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 UFSAR (Ref. 3) analysis for SGTR assumes the contaminated secondary fluid is only briefly released via the atmospheric relief valve for the affected steam generator. The 0.4 gpm operational primary to secondary leakage from all four steam generators is relatively inconsequential.

The safety analysis for the SLB accident assumes the 150 gpd primary to secondary LEAKAGE is through the affected generator as an initial condition. The dose consequences resulting from the SLB accident are well within the limits defined in 10 CFR 100 or the staff approved licensing basis (i.e., a small fraction of these limits).

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

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LCO

RCS operational LEAKAGE shall be limited to:

a. Pressure Boundary LEAKAGE

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

b. Unidentified LEAKAGE

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

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BASES

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LCO (continued)

c. Identified LEAKAGE

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

d. Primary to Secondary LEAKAGE Through Any One SG

The limit of 150 gallons per day per SG is based on the operational LEAKAGE performance criterion in NEI 97-06, Steam Generator Program Guidelines (Ref. 4). The Steam Generator Program operational LEAKAGE performance criterion in NEI 97-06 states, "The RCS operational primary to secondary leakage through any one SG shall be limited to 150 gallons per day." The limit is based on operating experience with SG tube degradation mechanisms that result in tube leakage. The operational leakage rate criterion in conjunction with the implementation of the Steam Generator Program is an effective measure for minimizing the frequency of steam generator tube ruptures.

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APPLICABILITY

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

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

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

BASES

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ACTIONS

A.1

Unidentified LEAKAGE or identified 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 primary to secondary LEAKAGE is not within limit, or if unidentified or identified 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. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. The reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. This action reduces the LEAKAGE and also reduces the factors that tend to degrade the pressure boundary.

The 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. In MODE 5, the pressure stresses acting on the RCPB are much lower, and further deterioration is much less likely.

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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. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. Unidentified LEAKAGE and identified LEAKAGE are determined by performance of an RCS water inventory balance.

The RCS water inventory balance must be met with the reactor at steady state operating conditions (stable temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows). The Surveillance is modified by two Notes. Note 1 states that this SR is not required to be performed until 12 hours after establishing steady state operation. The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established.

BASES

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SURVEILLANCE REQUIREMENTS (continued)

Steady state operation is required to perform a proper inventory balance since calculations during maneuvering are not useful. For RCS operational LEAKAGE determination by water inventory balance, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP 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. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. These leakage detection systems are specified in LCO 3.4.15, "RCS Leakage Detection Instrumentation."

Note 2 states that this SR is not applicable to primary to secondary LEAKAGE because LEAKAGE of 150 gallons per day cannot be measured accurately by an RCS water inventory balance.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.4.13.2

This SR verifies that primary to secondary LEAKAGE is less or equal to 150 gallons per day through any one SG. Satisfying the primary to secondary LEAKAGE limit ensures that the operational LEAKAGE performance criterion in the Steam Generator Program is met. If this SR is not met, compliance with LCO 3.4.17, "Steam Generator Tube Integrity," should be evaluated. The 150 gallons per day limit is measured at ambient temperature as described in Reference 5. The operational LEAKAGE rate limit applies to LEAKAGE through any one SG. If it is not practical to assign the LEAKAGE to an individual SG, all the primary to secondary LEAKAGE should be conservatively assumed to be from one SG.

The Surveillance is modified by a Note which states that the Surveillance is not required to be performed until 12 hours after establishment of steady state operation. For RCS primary to secondary LEAKAGE determination, 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.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

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SURVEILLANCE REQUIREMENTS (continued)

The primary to secondary LEAKAGE is determined using continuous process radiation monitors or radiochemical grab sampling in accordance with the EPRI guidelines (Ref. 5).

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REFERENCES

1. 10 CFR 50, Appendix A, GDC 30.
  2. Regulatory Guide 1.45, May 1973.
  3. UFSAR, Section 15.4.3.
  4. NEI 97-06, "Steam Generator Program Guidelines."
  5. EPRI, "Pressurized Water Reactor Primary-to-Secondary Leak Guidelines."
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## B 3.4 REACTOR COOLANT SYSTEM (RCS)

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

#### BASES

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

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

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

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

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

- a. Residual Heat Removal (RHR) System,

BASES

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BACKGROUND (continued)

- b. Safety Injection System, and
- c. Chemical and Volume Control System.

The PIVs are listed in Table B 3.4.14-1

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

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APPLICABLE  
SAFETY  
ANALYSES

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

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

RCS PIV leakage satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

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LCO

RCS PIV leakage is identified LEAKAGE into closed systems connected to the RCS. Isolation valve leakage is usually on the order of drops per minute. Leakage that increases significantly suggests that something is operationally wrong and corrective action must be taken.

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

## BASES

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### LCO (continued)

Reference 6 permits leakage testing at a lower pressure differential than between the specified maximum RCS pressure and the normal pressure of the connected system during RCS operation (the maximum pressure differential) in those types of valves in which the higher service pressure will tend to diminish the overall leakage channel opening. In such cases, the observed rate may be adjusted to the maximum pressure differential by assuming leakage is directly proportional to the pressure differential to the one half power.

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### APPLICABILITY

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

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

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### ACTIONS

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

#### A.1 and A.2

The flow path must be isolated by two valves. Required Actions A.1 and A.2 are modified by a Note that the valves used for isolation must meet the same leakage requirements as the PIVs and must be within the RCPB.

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

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

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## BASES

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### ACTIONS (continued)

#### B.1 and B.2

If Required Actions and associated Completion Times of Condition A are not met, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. This Action may reduce the leakage and also reduces the potential for a LOCA outside the containment. The allowed Completion Times are reasonable based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

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### SURVEILLANCE REQUIREMENTS

#### SR 3.4.14.1

Performance of leakage testing on each RCS PIV or isolation valve used to satisfy Required Action A.1 and Required Action 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, if the plant does not go into MODE 5 for at least 7 days. The Frequency is consistent with 10 CFR 50.55.a(g) (Ref. 7) as contained in the Inservice Testing Program, is within frequency allowed by the American Society of Mechanical Engineers (ASME) Code (Ref.6), and is based on the need to perform such surveillances under the conditions that apply during an outage and the potential for an unplanned transient if the surveillance were performed with the reactor at power.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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

## BASES

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### SURVEILLANCE REQUIREMENTS (continued)

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 complementary 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 RHR System when the RHR System is aligned to the RCS in the shutdown cooling mode of operation. PIVs contained in the RHR shutdown cooling flow path must be leakage rate tested after RHR is secured and stable unit conditions and the necessary differential pressures are established.

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### 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. ASME Code for Operation and Maintenance of Nuclear Power Plants.
  7. 10 CFR 50.55a(g).
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Table B 3.4.14-1 (page 1 of 1)  
Reactor Coolant System Pressure Isolation Valves

<u>VALVE NUMBER</u>	<u>FUNCTION</u>
63-560	Accumulator Discharge
63-561	Accumulator Discharge
63-562	Accumulator Discharge
63-563	Accumulator Discharge
63-622	Accumulator Discharge
63-623	Accumulator Discharge
63-624	Accumulator Discharge
63-625	Accumulator Discharge
63-551	Safety Injection (Cold Leg)
63-553	Safety Injection (Cold Leg)
63-557	Safety Injection (Cold Leg)
63-555	Safety Injection (Cold Leg)
63-632	Residual Heat Removal (Cold Leg)
63-633	Residual Heat Removal (Cold Leg)
63-634	Residual Heat Removal (Cold Leg)
63-635	Residual Heat Removal (Cold Leg)
63-641	Residual Heat Removal/Safety Injection (Hot Leg)
63-644	Residual Heat Removal/Safety Injection (Hot Leg)
63-558	Safety Injection (Hot Leg)
63-559	Safety Injection (Hot Leg)
63-543	Safety Injection (Hot Leg)
63-545	Safety Injection (Hot Leg)
63-547	Safety Injection (Hot Leg)
63-549	Safety Injection (Hot Leg)
63-640	Residual Heat Removal (Hot Leg)
63-643	Residual Heat Removal (Hot Leg)
FCV-74-1	Residual Heat Removal
FCV-74-2	Residual Heat Removal

## B 3.4 REACTOR COOLANT SYSTEM (RCS)

### B 3.4.15 RCS Leakage Detection Instrumentation

#### BASES

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**BACKGROUND** GDC 30 of Appendix A to 10 CFR 50 (Ref. 1) requires means for detecting and, to the extent practical, identifying the location of the source of RCS LEAKAGE. Regulatory Guide 1.45, Revision 0, (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.

The containment pocket sump used to collect unidentified LEAKAGE is instrumented to alarm for increases above the normal flow rates.

The reactor coolant contains radioactivity that, when released to the containment, may be detected by radiation monitoring instrumentation. A radioactivity detection system is included for monitoring particulate activity because of its sensitivity and rapid response to RCS LEAKAGE.

Other indications may be used to detect an increase in unidentified LEAKAGE; however, they are not required to be OPERABLE by this LCO. An increase in humidity of the containment atmosphere would indicate release of water vapor to the containment. Dew point temperature measurements can thus be used to monitor humidity levels of the containment atmosphere as an indicator of potential RCS LEAKAGE.

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 pocket sump. 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.

BASES

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BACKGROUND (continued)

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 is 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.

The above-mentioned LEAKAGE detection methods or systems differ in sensitivity and response time. Some of these systems could serve as early alarm systems signaling the operators that closer examination of other detection systems is necessary to determine the extent of any corrective action that may be required.

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APPLICABLE  
SAFETY  
ANALYSES

The need to evaluate the severity of an alarm or an indication is important to the operators, and the ability to compare and verify with indications from other systems is necessary.

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

RCS leakage detection instrumentation satisfies Criterion 1 of 10 CFR 50.36(c)(2)(ii).

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LCO

This LCO requires instruments of diverse monitoring principles to be OPERABLE to provide confidence that small amounts of unidentified LEAKAGE are detected in time to allow actions to place the plant in a safe condition, when RCS LEAKAGE indicates possible RCPB degradation.

The LCO requires two instruments to be OPERABLE.

The containment pocket sump is used to collect unidentified LEAKAGE. The monitor on the containment pocket sump detects level and is instrumented to detect when there is an increase above the normal value by 1 gpm. The identification of an increase in unidentified LEAKAGE will be delayed by the time required for the unidentified LEAKAGE to travel to the containment pocket sump and it may take longer than one hour to detect a 1 gpm increase in unidentified LEAKAGE, depending on the

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BASES

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LCO (continued)

origin and magnitude of the LEAKAGE. This sensitivity is acceptable for containment pocket sump level monitor OPERABILITY.

The reactor coolant contains radioactivity that, when released to the containment, can be detected by the lower containment atmosphere particulate radioactivity monitor. A radioactivity detection system is included for monitoring particulate activity because of the sensitivity and rapid response to RCS LEAKAGE, but has recognized limitations. 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. If there are few fuel element cladding defects and low levels of activation products, it may not be possible for the lower containment atmosphere particulate radioactivity monitor to detect a 1 gpm increase within 1 hour during normal operation. However, the lower containment atmosphere particulate radioactivity monitor is OPERABLE when it is capable of detecting a 1 gpm increase in unidentified LEAKAGE within 1 hour given an RCS activity equivalent to that assumed in the design calculations for the monitor (Reference 3).

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

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APPLICABILITY

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

In MODE 5 or 6, the temperature is to be  $\leq 200^{\circ}\text{F}$  and pressure is maintained low or at atmospheric pressure. Since the temperatures and pressures are far lower than those for MODES 1, 2, 3, and 4, the likelihood of leakage and crack propagation are much smaller. Therefore, the requirements of this LCO are not applicable in MODES 5 and 6.

BASES

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ACTIONS

A.1 and A.2

With the required containment pocket sump level monitor inoperable, no other form of sampling can provide the equivalent information; however, the lower containment atmosphere particulate radioactivity monitor will provide indications of changes in leakage. Together with the lower containment atmosphere particulate radioactivity monitor, the periodic surveillance for RCS water inventory balance, SR 3.4.13.1, must be performed at an increased frequency of 24 hours to provide information that is adequate to detect leakage. A Note is added allowing that SR 3.4.13.1 is not required to be performed until 12 hours after establishing steady state operation (stable temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows). The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established.

Restoration of the required pocket sump level monitor to OPERABLE status within a Completion Time of 30 days is required to regain the function 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.

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

With the lower containment atmosphere particulate radioactivity monitoring instrumentation channel 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 water inventory balance performed every 24 hours, the reactor may be operated for up to 30 days to allow restoration of the required lower containment atmosphere particulate radioactivity monitors.

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

## BASES

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### ACTIONS (continued)

#### C.1 and C.2

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

#### D.1

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

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### SURVEILLANCE REQUIREMENTS

#### SR 3.4.15.1

SR 3.4.15.1 requires the performance of a CHANNEL CHECK of the required lower containment atmosphere particulate radioactivity monitor. The check gives reasonable confidence that the channel is operating properly.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

#### SR 3.4.15.2

SR 3.4.15.2 requires the performance of a COT on the required lower containment atmosphere particulate radioactivity monitor. The test ensures that the monitor can perform its function in the desired manner. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable COT of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. The test verifies the alarm setpoint and relative accuracy of the instrument string.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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BASES

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.15.3 and SR 3.4.15.4

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 Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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REFERENCES

1. 10 CFR 50, Appendix A, Section IV, GDC 30.
  2. Regulatory Guide 1.45, Revision 0, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.
  3. UFSAR, Section 5.2.7.
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## B 3.4 REACTOR COOLANT SYSTEM (RCS)

### B 3.4.16 RCS Specific Activity

#### BASES

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**BACKGROUND** The maximum dose that an individual at the exclusion area boundary can receive for 2 hours following an accident, or at the low population zone outer boundary for the radiological release duration, is specified in 10 CFR 100.11 (Ref. 1). Doses to control room operators must be limited per GDC 19. The limits on specific activity ensure that the offsite and control room doses are appropriately limited during analyzed transients and accidents.

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

The LCO contains specific activity limits for both DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133. The allowable levels are intended to ensure that offsite and control room doses meet the appropriate acceptance criteria in the Standard Review Plan (SRP) (Ref. 2).

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**APPLICABLE SAFETY ANALYSES** The LCO limits on the specific activity of the reactor coolant ensures that the resulting offsite and control room doses meet the appropriate SRP acceptance criteria following a SLB or SGTR accident. The safety analyses (Refs. 3 and 4) assume the specific activity of the reactor coolant is at the LCO limits, and an existing reactor coolant steam generator (SG) tube leakage rate of 150 gpd exists. The safety analyses assume the specific activity of the secondary coolant is at its limit of 0.1  $\mu\text{Ci/gm}$  DOSE EQUIVALENT I-131 from LCO 3.7.16, "Secondary Specific Activity."

The analyses for the SLB and SGTR accidents establish the acceptance limits for RCS specific activity. Reference to these analyses is used to assess changes to the unit that could affect RCS specific activity, as they relate to the acceptance limits.

The safety analyses consider two cases of reactor coolant iodine specific activity. One case assumes specific activity at 0.35  $\mu\text{Ci/gm}$  DOSE EQUIVALENT I-131 with a concurrent large iodine spike that increases the rate of release of iodine from the fuel rods containing cladding defects to the primary coolant immediately after the SLB (by a factor of 500), or SGTR (by a factor of 500), respectively. The second case assumes the

## BASES

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### APPLICABLE SAFETY ANALYSES (continued)

initial reactor coolant iodine activity at 21.0  $\mu\text{Ci/gm}$  DOSE EQUIVALENT I-131 due to an iodine spike caused by a reactor or an RCS transient prior to the accident. In both cases, the noble gas activity is assumed to be 1612.6  $\mu\text{Ci/gm}$  DOSE EQUIVALENT XE-133.

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

The loss of offsite power causes the steam dump valves to close to protect the condenser. The rise in pressure in the ruptured SG discharges radioactively contaminated steam to the atmosphere through the SG power operated relief valves and the main steam safety valves. The unaffected SGs remove core decay heat by venting steam to the atmosphere until the cooldown ends and the Residual Heat Removal (RHR) system is placed in service.

The SLB radiological analysis assumes that offsite power is lost at the same time as the pipe break occurs outside containment. Reactor trip occurs after the generation of an SI signal on low steam line pressure. The affected SG blows down completely and steam is vented directly to the atmosphere. The unaffected SGs remove core decay heat by venting steam to the atmosphere until the cooldown ends and the RHR system is placed in service.

Operation with iodine specific activity levels greater than the LCO limit is permissible, if the activity levels do not exceed 21.0  $\mu\text{Ci/gm}$  for more than 48 hours.

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

RCS specific activity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

BASES

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LCO The iodine specific activity in the reactor coolant is limited to 0.35  $\mu\text{Ci/gm}$  DOSE EQUIVALENT I-131, and the noble gas specific activity in the reactor coolant is limited to 1612.6  $\mu\text{Ci/gm}$  DOSE EQUIVALENT XE-133. The limits on specific activity ensure that offsite and control room doses will meet the appropriate SRP acceptance criteria (Ref. 2).

The SLB and SGTR accident analyses (Refs. 3 and 4) show that the calculated doses are within acceptable limits. Violation of the LCO may result in reactor coolant radioactivity levels that could, in the event of a SLB or SGTR, lead to doses that exceed the SRP acceptance criteria (Ref. 2).

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APPLICABILITY In MODES 1, 2, 3, and 4, operation within the LCO limits for DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133 is necessary to limit the potential consequences of a SLB or SGTR to within the SRP acceptance criteria (Ref. 2).

In MODES 5 and 6, the steam generators are not being used for decay heat removal, the RCS and steam generators are depressurized, and primary to secondary leakage is minimal. Therefore, the monitoring of RCS specific activity is not required.

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ACTIONS A.1 and A.2

With the DOSE EQUIVALENT I-131 greater than the LCO limit, samples at intervals of 4 hours must be taken to demonstrate that the specific activity is  $\leq 21.0 \mu\text{Ci/gm}$ . An isotopic analysis of a reactor coolant sample must be performed for at least I-131, I-133, and I-135. The Completion Time of 4 hours is required to obtain and analyze a sample. Sampling is continued every 4 hours to provide a trend.

The DOSE EQUIVALENT I-131 must be restored to within limits within 48 hours. The Completion Time of 48 hours is acceptable since it is expected that, if there were an iodine spike, the normal coolant iodine concentration would be restored within this time period. Also, there is a low probability of a SLB or SGTR occurring during this time period.

A Note permits the use of the provisions of LCO 3.0.4.c. This allowance permits entry into the applicable MODE(S), relying on Required Actions A.1 and A.2 while the DOSE EQUIVALENT I-131 LCO limit is not met. This allowance 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.

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BASES

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ACTIONS (continued)

B.1 and B.2

If a Required Action and the associated Completion Time of Condition A is not met, or if the DOSE EQUIVALENT I-131 is  $> 21.0 \mu\text{Ci/gm}$ , or if the DOSE EQUIVALENT XE-133 is  $> 1612.6 \mu\text{Ci/gm}$ , the reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

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SURVEILLANCE  
REQUIREMENTS

SR 3.4.16.1

SR 3.4.16.1 requires performing a gamma isotopic analysis and calculating the DOSE EQUIVALENT XE-133 using the dose conversion factors in the DOSE EQUIVALENT XE-133 definition. 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 the noble gas specific activity.

Trending the results of this Surveillance allows proper remedial action to be taken before reaching the LCO limit under normal operating conditions.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

Due to the inherent difficulty in detecting Kr-85 in a reactor coolant sample due to masking from radioisotopes with similar decay energies, such as F-18 and I-134, it is acceptable to include the minimum detectable activity for Kr-85 in the SR 3.4.16.1 calculation. If a specific noble gas nuclide listed in the definition of DOSE EQUIVALENT XE-133 is not detected, it should be assumed to be present at the minimum detectable activity.

A Note modifies the SR to only require the surveillance to be performed in MODES 1, 2, and 3 with  $T_{\text{avg}} \geq 500^\circ\text{F}$ .

BASES

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.16.2

This Surveillance is performed to ensure iodine specific activity remains within the LCO limit during normal operation and following fast power changes when iodine spiking is more apt to occur.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

The Frequency, between 2 and 6 hours after a power change  $\geq 15\%$  RTP within a 1 hour period, is established because the iodine levels peak during this time following iodine spiking initiation; samples at other times would provide inaccurate results.

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REFERENCES

1. 10 CFR 100.11.
  2. Standard Review Plan (SRP) Section 15.1.5 Appendix A (SLB) and Section 15.6.3 (SGTR).
  3. UFSAR, Section 15.5.4.
  4. UFSAR, Section 15.5.5.
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## B 3.4 REACTOR COOLANT SYSTEM (RCS)

### B 3.4.17 Steam Generator (SG) Tube Integrity

#### BASES

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**BACKGROUND** Steam generator (SG) tubes are small diameter, thin walled tubes that carry primary coolant through the primary to secondary heat exchangers. The SG tubes have a number of important safety functions. Steam generator tubes are an integral part of the reactor coolant pressure boundary (RCPB) and, as such, are relied on to maintain the primary system's pressure and inventory. The SG tubes isolate the radioactive fission products in the primary coolant from the secondary system. In addition, as part of the RCPB, the SG tubes are unique in that they act as the heat transfer surface between the primary and secondary systems to remove heat from the primary system. This Specification addresses only the RCPB integrity function of the SG. The SG heat removal function is addressed 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," and LCO 3.4.7, "RCS Loops – MODE 5, Loops Filled."

SG tube integrity means that the tubes are capable of performing their intended RCPB safety function consistent with the licensing basis, including applicable regulatory requirements.

Steam generator tubing is subject to a variety of degradation mechanisms. Steam generator tubes may experience tube degradation related to corrosion phenomena, such as wastage, pitting, intergranular attack, and stress corrosion cracking, along with other mechanically induced phenomena such as denting and wear. These degradation mechanisms can impair tube integrity if they are not managed effectively. The SG performance criteria are used to manage SG tube degradation.

Specification 5.5.7, "Steam Generator (SG) Program," requires that a program be established and implemented to ensure that SG tube integrity is maintained. Pursuant to Specification 5.5.7, tube integrity is maintained when the SG performance criteria are met. There are three SG performance criteria: structural integrity, accident induced leakage, and operational LEAKAGE. The SG performance criteria are described in Specification 5.5.7. Meeting the SG performance criteria provides reasonable assurance of maintaining tube integrity at normal and accident conditions.

The processes used to meet the SG performance criteria are defined by the Steam Generator Program Guidelines (Ref. 1).

## BASES

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### APPLICABLE SAFETY ANALYSES

The steam generator tube rupture (SGTR) accident is the limiting design basis event for SG tubes and avoiding an SGTR is the basis for this Specification. The analysis of a SGTR event assumes a bounding primary to secondary LEAKAGE rate equal to the operational LEAKAGE rate limits in LCO 3.4.13, "RCS Operational LEAKAGE," plus the leakage rate associated with a double-ended rupture of a single tube. The accident analysis for a SGTR assumes the contaminated secondary fluid is only briefly released to the atmosphere via SG atmospheric relief valves and safety valves.

The analysis for design basis accidents and transients other than a SGTR assume the SG tubes retain their structural integrity (i.e., they are assumed not to rupture.) In these analyses, the steam discharge to the atmosphere is based on the total primary to secondary LEAKAGE from all SGs of 0.4 gallons per minute or is assumed to increase to 1 gallon per minute as a result of accident induced conditions. For accidents that do not involve fuel damage, the primary coolant activity level of DOSE EQUIVALENT I-131 is assumed to be equal to the LCO 3.4.16, "RCS Specific Activity," limits. For accidents that assume fuel damage, the primary coolant activity is a function of the amount of activity released from the damaged fuel. The dose consequences of these events are within the limits of GDC 19 (Ref. 2) and 10 CFR 100 (Ref. 3).

Steam generator tube integrity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

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### LCO

The LCO requires that SG tube integrity be maintained. The LCO also requires that all SG tubes that satisfy the plugging criteria be plugged in accordance with the Steam Generator Program.

During an SG inspection, any inspected tube that satisfies the Steam Generator Program plugging criteria is removed from service by plugging. If a tube was determined to satisfy the plugging criteria but was not plugged, the tube may still have tube integrity.

In the context of this Specification, a SG tube is defined as the entire length of the tube, including the tube wall, between the tube-to-tubesheet weld at the tube inlet and the tube-to-tubesheet weld at the tube outlet. The tube-to-tubesheet weld is not considered part of the tube.

A SG tube has tube integrity when it satisfies the SG performance criteria. The SG performance criteria are defined in Specification 5.5.7, "Steam Generator Program," and describe acceptable SG tube performance. The Steam Generator Program also provides the evaluation process for determining conformance with the SG performance criteria.

## BASES

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### LCO (continued)

There are three SG performance criteria: structural integrity, accident induced leakage, and operational LEAKAGE. Failure to meet any one of these criteria is considered failure to meet the LCO.

The structural integrity performance criterion provides a margin of safety against tube burst or collapse under normal and accident conditions, and ensures structural integrity of the SG tubes under all anticipated transients included in the design specification. Tube burst is defined as, "The gross structural failure of the tube wall. The condition typically corresponds to an unstable opening displacement (e.g., opening area increased in response to constant pressure) accompanied by ductile (plastic) tearing of the tube material at the ends of the degradation." Tube collapse is defined as, "For the load displacement curve for a given structure, collapse occurs at the top of the load versus displacement curve where the slope of the curve becomes zero." The structural integrity performance criterion provides guidance on assessing loads that have a significant effect on burst or collapse. In that context, the term "significant" is defined as "An accident loading condition other than differential pressure is considered significant when the addition of such loads in the assessment of the structural integrity performance criterion could cause a lower structural limit or limiting burst/collapse condition to be established." For tube integrity evaluations, except for circumferential degradation, axial thermal loads are classified as secondary loads. For circumferential degradation, the classification of axial thermal loads as primary or secondary loads will be evaluated on a case-by-case basis. The division between primary and secondary classifications will be based on detailed analysis and/or testing.

Structural integrity requires that the primary membrane stress intensity in a tube not exceed the yield strength for all ASME Code, Section III, Service Level A (normal operating conditions) and Service Level B (upset or abnormal conditions) transients included in the design specification. This includes safety factors and applicable design basis loads based on ASME Code, Section III, Subsection NB (Ref. 4) and Draft Regulatory Guide 1.121 (Ref. 5).

The accident induced leakage performance criterion ensures that the primary to secondary LEAKAGE caused by a design basis accident, other than a SGTR, is within the accident analysis assumptions. The accident analyses assumptions are discussed in the Applicable Safety Analyses section. The accident induced leakage rate includes any primary to secondary LEAKAGE existing prior to the accident in addition to primary to secondary LEAKAGE induced during the accident.

## BASES

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### LCO (continued)

The operational LEAKAGE performance criterion provides an observable indication of SG tube conditions during plant operation. The limit on operational LEAKAGE is contained in LCO 3.4.13, "RCS Operational LEAKAGE," and limits primary to secondary LEAKAGE through any one SG to 150 gallons per day. This limit is based on the assumption that a single crack leaking this amount would not propagate to a SGTR under the stress conditions of a LOCA or a steam line break. If this amount of LEAKAGE is due to more than one crack, the cracks are very small, and the above assumption is conservative.

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### APPLICABILITY

Steam generator tube integrity is challenged when the pressure differential across the tubes is large. Large differential pressures across SG tubes can only be experienced in MODE 1, 2, 3, or 4.

RCS conditions are far less challenging in MODES 5 and 6 than during MODES 1, 2, 3, and 4. In MODES 5 and 6, primary to secondary differential pressure is low, resulting in lower stresses and reduced potential for LEAKAGE.

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### ACTIONS

The ACTIONS are modified by a Note clarifying that the Conditions may be entered independently for each SG tube. This is acceptable because the Required Actions provide appropriate compensatory actions for each affected SG tube. Complying with the Required Actions may allow for continued operation, and subsequent affected SG tubes are governed by subsequent Condition entry and application of associated Required Actions.

#### A.1 and A.2

Condition A applies if it is discovered that one or more SG tubes examined in an inservice inspection satisfy the tube plugging criteria but were not plugged in accordance with the Steam Generator Program as required by SR 3.4.17.2. An evaluation of SG tube integrity of the affected tube(s) must be made. Steam generator tube integrity is based on meeting the SG performance criteria described in the Steam Generator Program. The SG plugging criteria define limits on SG tube degradation that allow for flaw growth between inspections while still providing assurance that the SG performance criteria will continue to be met. In order to determine if a SG tube that should have been plugged has tube integrity, an evaluation must be completed that demonstrates that the SG performance criteria will continue to be met until the next refueling outage or SG tube inspection. The tube integrity determination is based on the estimated condition of the tube at the time the situation is discovered and the estimated growth of the degradation prior to the next

BASES

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ACTIONS (continued)

SG tube inspection. If it is determined that tube integrity is not being maintained, Condition B applies.

A Completion Time of 7 days is sufficient to complete the evaluation while minimizing the risk of plant operation with a SG tube that may not have tube integrity.

If the evaluation determines that the affected tube(s) have tube integrity, Required Action A.2 allows plant operation to continue until the next refueling outage or SG inspection provided the inspection interval continues to be supported by an operational assessment that reflects the affected tubes. However, the affected tube(s) must be plugged prior to entering MODE 4 following the next refueling outage or SG inspection. This Completion Time is acceptable since operation until the next inspection is supported by the operational assessment.

B.1 and B.2

If the Required Actions and associated Completion Times of Condition A are not met or if SG tube integrity is not being maintained, the reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours.

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

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SURVEILLANCE  
REQUIREMENTS

SR 3.4.17.1

During shutdown periods the SGs are inspected as required by this SR and the Steam Generator Program. NEI 97-06, Steam Generator Program Guidelines (Ref. 1), and its referenced EPRI Guidelines, establish the content of the Steam Generator Program. Use of the Steam Generator Program ensures that the inspection is appropriate and consistent with accepted industry practices.

During SG inspections a condition monitoring assessment of the SG tubes is performed. The condition monitoring assessment determines the "as found" condition of the SG tubes. The purpose of the condition monitoring assessment is to ensure that the SG performance criteria have been met for the previous operating period.

## BASES

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### SURVEILLANCE REQUIREMENTS (continued)

The Steam Generator Program determines the scope of the inspection and the methods used to determine whether the tubes contain flaws satisfying the tube plugging criteria. Inspection scope (i.e., which tubes or areas of tubing within the SG are to be inspected) is a function of existing and potential degradation locations. The Steam Generator Program also specifies the inspection methods to be used to find potential degradation. Inspection methods are a function of degradation morphology, non-destructive examination (NDE) technique capabilities, and inspection locations.

The Steam Generator Program defines the Frequency of SR 3.4.17.1. The Frequency is determined by the operational assessment and other limits in the SG examination guidelines (Ref. 6). The Steam Generator Program uses information on existing degradations and growth rates to determine an inspection Frequency that provides reasonable assurance that the tubing will meet the SG performance criteria at the next scheduled inspection. In addition, Specification 5.5.7 contains prescriptive requirements concerning inspection intervals to provide added assurance that the SG performance criteria will be met between scheduled inspections. If crack indications are found in any SG tube, the maximum inspection interval for all affected and potentially affected SGs is restricted by Specification 5.5.7 until subsequent inspections support extending the inspection interval.

#### SR 3.4.17.2

During an SG inspection, any inspected tube that satisfies the Steam Generator Program plugging criteria is removed from service by plugging. The tube plugging criteria delineated in Specification 5.5.7 are intended to ensure that tubes accepted for continued service satisfy the SG performance criteria with allowance for error in the flaw size measurement and for future flaw growth. In addition, the tube plugging criteria, in conjunction with other elements of the Steam Generator Program, ensure that the SG performance criteria will continue to be met until the next inspection of the subject tube(s). Reference 1 provides guidance for performing operational assessments to verify that the tubes remaining in service will continue to meet the SG performance criteria.

The Frequency of prior to entering MODE 4 following a SG inspection ensures that the Surveillance has been completed and all tubes meeting the plugging criteria are plugged prior to subjecting the SG tubes to significant primary to secondary pressure differential.

BASES

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- REFERENCES
1. NEI 97-06, "Steam Generator Program Guidelines."
  2. 10 CFR 50 Appendix A, GDC 19.
  3. 10 CFR 100.
  4. ASME Boiler and Pressure Vessel Code, Section III, Subsection NB.
  5. Draft Regulatory Guide 1.121, "Basis for Plugging Degraded Steam Generator Tubes," August 1976.
  6. EPRI, "Pressurized Water Reactor Steam Generator Examination Guidelines."
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## B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

### B 3.5.1 Accumulators

#### BASES

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

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

In the refill phase of a large break LOCA, which immediately follows the blowdown phase, 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 accumulator inventory is then available to help fill voids in the lower plenum and reactor vessel downcomer so as to establish a recovery level at the bottom of the core and ongoing reflood of the core with the addition of safety injection (SI) water.

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

Each accumulator is piped into an RCS cold leg via an accumulator line and is isolated from the RCS by a motor operated isolation valve and two check valves in series.

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

## BASES

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### APPLICABLE SAFETY ANALYSES

The accumulators are assumed OPERABLE in both the large and small break LOCA analyses at full power (Ref. 1). These are the Design Basis Accidents (DBAs) that establish the acceptance limits for the accumulators. Reference to the analyses for these DBAs is used to assess changes in the accumulators as they relate to the acceptance limits.

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

The limiting large break LOCA is a double ended guillotine break. Based on deterministic studies, the worst break location is in the cold leg piping between the reactor coolant pump and the reactor vessel for the RCS loop containing the pressurizer. During this event, the accumulators discharge to the RCS as soon as RCS pressure decreases to below accumulator pressure.

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

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

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

## BASES

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### APPLICABLE SAFETY ANALYSES (continued)

small break LOCA and there is a high probability that the criteria are met following a large break LOCA:

- a. Maximum fuel element cladding temperature is  $\leq 2200^{\circ}\text{F}$ ,
- b. Maximum cladding oxidation is  $\leq 0.17$  times the total cladding thickness before oxidation,
- c. Maximum hydrogen generation from a zirconium water reaction is  $\leq 0.01$  times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react, and
- d. Core is maintained in a coolable geometry.

Since the accumulators discharge during the blowdown phase and the first few seconds of the refill phase of a large break LOCA, they do not contribute to the long term cooling requirements of 10 CFR 50.46.

For both the large and small break LOCA analyses, a nominal contained accumulator water volume is used. The contained water volume is the same as the deliverable volume for the accumulators, since the accumulators are emptied, once discharged. The large and small break LOCA safety analyses are performed with accumulator volumes that are consistent with the LOCA evaluation models. The realistic large break LOCA safety analysis takes values between 7515 gallons and 8194 gallons. To allow for instrument inaccuracy, values of 7615 gallons and 7960 gallons are specified. The small break LOCA safety analysis assumes a value from within the range of values used for the large break safety analysis.

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

BASES

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APPLICABLE SAFETY ANALYSES (continued)

The large and small break LOCA analyses are performed with accumulator pressures that are consistent with the LOCA evaluation models. The realistic large break LOCA safety analysis takes values between 600 psig and 683 psig. To allow for instrument inaccuracy, values of 624 psig and 668 psig are specified. The small break LOCA safety analysis assumes a value from the low end of the range of values taken for the large break safety analysis. The maximum nitrogen cover pressure limit prevents accumulator relief valve actuation, and ultimately preserves accumulator integrity.

The effects on containment mass and energy releases from the accumulators are accounted for in the appropriate analyses (Refs. 1 and 3).

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

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LCO

The LCO establishes the minimum conditions required to ensure that the accumulators are available to accomplish their core cooling safety function following a LOCA. Four accumulators are required to ensure that 100% of the contents of three of the accumulators will reach the core during a LOCA. This is consistent with the assumption that the contents of one accumulator spill through the break. If less than three accumulators are injected during the blowdown phase of a LOCA, the ECCS acceptance criteria of 10 CFR 50.46 (Ref. 2) could be violated.

For an accumulator to be considered OPERABLE, the isolation valve must be fully open, power removed above 2000 psig, and the limits established in the SRs for contained volume, boron concentration, and nitrogen cover pressure must be met.

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APPLICABILITY

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

This LCO is only applicable at pressures > 1000 psig. At pressures ≤ 1000 psig, the rate of RCS blowdown is such that the ECCS pumps can provide adequate injection to ensure that peak clad temperature remains below the 10 CFR 50.46 (Ref. 2) limit of 2200°F.

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

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BASES

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## ACTIONS

A.1

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

B.1

If one accumulator is inoperable for a reason other than boron concentration, the accumulator must be returned to OPERABLE status within 24 hours. In this Condition, the required contents of three accumulators cannot be assumed to reach the core during a LOCA. Due to the severity of the consequences should a LOCA occur in these conditions, the 24 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 potential for exposure of the plant to a LOCA under these conditions. The 24 hours allowed to restore an inoperable accumulator to OPERABLE status is justified in WCAP-15049-A, Rev. 1 (Ref. 4).

C.1 and C.2

If the accumulator cannot be returned to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and RCS pressure reduced to  $\leq 1000$  psig within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

BASES

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ACTIONS (continued)

D.1

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

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SURVEILLANCE  
REQUIREMENTS

SR 3.5.1.1

Each accumulator isolation valve should be verified to be fully open. This verification ensures that the accumulators are available for injection and ensures timely discovery if a valve should be less than fully open. If an isolation valve is not fully open, the rate of injection to the RCS would be reduced. Although a motor operated valve position should not change with power removed, a closed valve could result in not meeting accident analyses assumptions.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.5.1.2 and SR 3.5.1.3

Borated water volume and nitrogen cover pressure are verified for each accumulator.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.5.1.4

The boron concentration should be verified to be within required limits for each accumulator since the static design of the accumulators limits the ways in which the concentration can be changed.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

Sampling the affected accumulator within 6 hours after a 1% volume increase will identify whether inleakage has caused a reduction in boron concentration to below the required limit. It is not necessary to verify boron concentration if the added water inventory is from the refueling water storage tank (RWST), because the water contained in the RWST is within the accumulator boron concentration requirements. This is consistent with the recommendation of NUREG-1366 (Ref. 5).

BASES

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.1.5

Verification that power is removed from each accumulator isolation valve operator when the RCS pressure is  $\geq 2000$  psig ensures that an active failure could not result in the undetected closure of an accumulator motor operated isolation valve. If this were to occur, only two accumulators would be available for injection given a single failure coincident with a LOCA.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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

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REFERENCES

1. UFSAR, Chapter 6.
  2. 10 CFR 50.46.
  3. UFSAR, Chapter 15.
  4. WCAP-15049-A, Rev. 1, April 1999.
  5. NUREG-1366, February 1990.
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## B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

### B 3.5.2 ECCS - Operating

#### BASES

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#### BACKGROUND

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

- a. Loss of coolant accident (LOCA), coolant leakage greater than the capability of the normal charging system,
- b. Rod ejection accident,
- c. Loss of secondary coolant accident, including uncontrolled steam release or loss of feedwater, and
- d. Steam generator tube rupture (SGTR).

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

There are two modes of ECCS operation, injection and recirculation. The injection mode consists of the injection phase and the recirculation mode consists of the cold leg recirculation phase and hot leg recirculation phase. In the injection phase, water is taken from the refueling water storage tank (RWST) and injected into the Reactor Coolant System (RCS) through the cold legs. When sufficient water is removed from the RWST to ensure that enough boron has been added to maintain the reactor subcritical and the containment sumps have enough water to supply the required net positive suction head to the ECCS pumps, suction is switched to the containment sump for cold leg recirculation. After approximately 5.5 hours, the ECCS flow is shifted to the hot leg recirculation phase to provide a backflush, which would reduce the boiling in the top of the core and any resulting boron precipitation.

The ECCS consists of three separate subsystems: centrifugal charging (high head), safety injection (SI) (intermediate head), and residual heat removal (RHR) (low head). Each subsystem consists of two redundant, 100% capacity trains. The ECCS accumulators and the RWST are also part of the ECCS, but are not considered part of an ECCS flow path as described by this LCO.

## BASES

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### BACKGROUND (continued)

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

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

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

During the recirculation phase of LOCA recovery, RHR pump suction is transferred to the containment sump. The RHR pumps then supply the other ECCS pumps. Initially, recirculation is through the same paths as the injection phase. Subsequently, recirculation alternates injection between the hot and cold legs.

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

## BASES

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### BACKGROUND (continued)

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

The ECCS subsystems are actuated upon receipt of an SI signal. If offsite power is available, the safeguard loads start immediately in the programmed sequence. If offsite power is not available, the Engineered Safety Feature (ESF) buses shed normal operating loads and are connected to the emergency diesel generators (EDGs). Safeguard loads are then actuated in the programmed time sequence. The time delay associated with diesel starting, sequenced loading, and pump starting determines the time required before pumped flow is available to the core following a LOCA.

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

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### APPLICABLE SAFETY ANALYSES

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

- a. Maximum fuel element cladding temperature is  $\leq 2200^{\circ}\text{F}$ ,
- b. Maximum cladding oxidation is  $\leq 0.17$  times the total cladding thickness before oxidation,
- c. Maximum hydrogen generation from a zirconium water reaction is  $\leq 0.01$  times the hypothetical amount generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react,
- d. Core is maintained in a coolable geometry, and
- e. Adequate long term core cooling capability is maintained.

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

## BASES

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### APPLICABLE SAFETY ANALYSES (continued)

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

- a. A large break LOCA event, with loss of offsite power and a single failure disabling one ECCS train (both containment spray trains are assumed to operate conservatively reducing containment pressure and increasing break flow) and
- b. A small break LOCA event, with a loss of offsite power and a single failure disabling one ECCS train.

During the blowdown stage of a LOCA, the RCS depressurizes as primary coolant is ejected through the break into the containment. The nuclear reaction is terminated either by moderator voiding during large breaks or control rod insertion for small breaks. Following depressurization, emergency cooling water is injected into the cold legs, flows into the downcomer, fills the lower plenum, and refloods the core.

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

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

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### LCO

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

## BASES

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### LCO (continued)

In MODES 1, 2, and 3, an ECCS train consists of a centrifugal charging subsystem, an SI subsystem, and an RHR subsystem. Each ECCS train includes the piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the RWST upon an SI signal and automatically transferring RHR suction to the containment sump.

During an event requiring ECCS actuation, a flow path is required to provide an abundant supply of water from the RWST to the RCS via the ECCS pumps and their respective supply headers to each of the four cold leg injection nozzles. In the long term, this flow path may be switched to take its supply from the containment sump and to supply its flow to the RCS hot and cold legs.

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

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

As indicated in Note 2, operation in MODE 3 with ECCS trains made incapable of injecting in order to facilitate entry into or exit from the Applicability of LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," is necessary for plants with an LTOP arming temperature at or near the MODE 3 boundary temperature of 350°F. LCO 3.4.12 requires that certain pumps be rendered incapable of injecting at and below the LTOP arming temperature. When this temperature is at or near the MODE 3 boundary temperature, time is needed to make pumps incapable of injecting prior to entering the LTOP Applicability, and provide time to restore the inoperable pumps to OPERABLE status on exiting the LTOP Applicability.

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### APPLICABILITY

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

BASES

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APPLICABILITY (continued)

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

In MODES 5 and 6, plant conditions are such that the probability of an event requiring ECCS injection is extremely low.

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ACTIONS

A.1

With one or more ECCS trains inoperable and at least 100% of the ECCS flow equivalent to a single OPERABLE ECCS train available, the inoperable components must be returned to OPERABLE status within 72 hours. The 72 hour Completion Time is based on an NRC reliability evaluation (Ref. 5) and is a reasonable time for repair of many ECCS components.

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

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

An event accompanied by a loss of offsite power and the failure of an EDG can disable one ECCS train until power is restored. A reliability analysis (Ref. 5) has shown that the impact of having one full ECCS train inoperable is sufficiently small to justify continued operation for 72 hours.

Reference 6 describes situations in which one component, such as an RHR crossover valve, can disable both ECCS trains. With one or more component(s) inoperable such that 100% of the flow equivalent to a single OPERABLE ECCS train is not available, the facility is in a condition outside the accident analysis. Therefore, LCO 3.0.3 must be immediately entered.

## BASES

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### ACTIONS (continued)

#### B.1 and B.2

If the inoperable trains cannot be returned to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

#### C.1

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

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### SURVEILLANCE REQUIREMENTS

#### SR 3.5.2.1

Verification of proper valve position ensures that the flow path from the ECCS pumps to the RCS is maintained. Misalignment of these valves could render both ECCS trains inoperable. Securing these valves in position by removal of power ensures that they cannot change position as a result of an active failure or be inadvertently misaligned. These valves are of the type, described in Reference 6, that can disable the function of both ECCS trains and invalidate the accident analyses.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.2.2

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

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.5.2.3

With the exception of the operating centrifugal charging pump, 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. ECCS piping is verified full of water by venting and/or ultrasonic testing (UT) pump casings and accessible high point vents. This will also prevent water hammer, pump cavitation, and pumping of noncondensable gas (e.g., air, nitrogen, or hydrogen) into the reactor vessel following an SI signal or during shutdown cooling.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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 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 plant safety analysis. SRs are specified in the Inservice Testing Program of the ASME Code.

## BASES

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### SURVEILLANCE REQUIREMENTS (continued)

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

#### SR 3.5.2.5 and SR 3.5.2.6

These Surveillances demonstrate that each automatic ECCS valve actuates to the required position on an actual or simulated SI signal and that each ECCS pump starts on receipt of an actual or simulated SI signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

#### SR 3.5.2.7

Realignment of valves in the flow path on an SI signal is necessary for proper ECCS performance. These valves have stops to allow proper positioning for restricted flow to a ruptured cold leg, ensuring that the other cold legs receive at least the required minimum flow.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

#### SR 3.5.2.8

Periodic inspections of the containment sump suction inlet ensure that it is unrestricted and stays in proper operating condition.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

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- REFERENCES
1. 10 CFR 50, Appendix A, GDC 35.
  2. 10 CFR 50.46.
  3. UFSAR, Section 6.3.
  4. UFSAR, Chapter 15, "Accident Analysis."
  5. NRC Memorandum to V. Stello, Jr., from R.L. Baer, "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975.
  6. IE Information Notice No. 87-01.
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## B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

### B 3.5.3 ECCS - Shutdown

#### BASES

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BACKGROUND	<p>The Background section for Bases 3.5.2, "ECCS - Operating," is applicable to these Bases, with the following modifications.</p> <p>In MODE 4, the required ECCS train consists of two separate subsystems: centrifugal charging (high head) and residual heat removal (RHR) (low head).</p> <p>The ECCS flow paths consist of piping, valves, RHR heat exchangers, and pumps such that water from the refueling water storage tank (RWST) can be injected into the Reactor Coolant System (RCS) following the accidents described in Bases 3.5.2.</p>
APPLICABLE SAFETY ANALYSES	<p>The Applicable Safety Analyses section of Bases 3.5.2 also applies to this Bases section.</p> <p>Due to the stable conditions associated with operation in MODE 4 and the reduced probability of occurrence of a Design Basis Accident (DBA), the ECCS operational requirements are reduced. It is understood in these reductions that automatic safety injection (SI) actuation is not available. In this MODE, sufficient time exists for manual actuation of the required ECCS to mitigate the consequences of a DBA.</p> <p>Only one train of ECCS is required for MODE 4. This requirement dictates that single failures are not considered during this MODE of operation. The ECCS trains satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).</p>
LCO	<p>In MODE 4, one of the two independent (and redundant) ECCS trains is required to be OPERABLE to ensure that sufficient ECCS flow is available to the core and adequate core cooling is maintained following a DBA.</p> <p>In MODE 4, an ECCS train consists of a centrifugal charging subsystem and an RHR subsystem. Each train includes the piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the RWST and transferring suction to the containment sump.</p> <p>During an event requiring ECCS actuation, a flow path is required to provide an abundant supply of water from the RWST to the RCS via the ECCS pumps and their respective supply headers to each of the four cold leg injection nozzles. In the long term, this flow path may be switched to take its supply from the containment sump and to deliver its flow to the RCS hot and cold legs.</p>

BASES

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LCO (continued)

This LCO is modified by two Notes. Note 1 allows an RHR train to be considered OPERABLE during alignment and operation for decay heat removal, if capable of being manually realigned (remote or local) to the ECCS mode of operation and not otherwise inoperable. A second Note allows the required ECCS RHR subsystem to be inoperable because of surveillance testing of RCS pressure isolation valve leakage (FCV-74-1 and FCV-74-2). This allows testing while RCS pressure is high enough to obtain valid leakage data and following valve closure for RHR decay heat removal path. The condition requiring manual realignment capability (FCV-74-1 and FCV-74-2 can be opened from the main control room) ensures that in the unlikely event of a DBA during the one hour of surveillance testing, the RHR subsystem can be placed in ECCS recirculation mode when required to mitigate the event. This allows operation in the RHR mode during MODE 4.

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APPLICABILITY

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

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

In MODES 5 and 6, plant conditions are such that the probability of an event requiring ECCS injection is extremely low.

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ACTIONS

A Note prohibits the application of LCO 3.0.4.b to an inoperable ECCS centrifugal charging subsystem when entering MODE 4. There is an increased risk associated with entering MODE 4 from MODE 5 with an inoperable ECCS centrifugal charging subsystem and the provisions of LCO 3.0.4.b, which allow entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, should not be applied in this circumstance.

A.1

With no ECCS RHR subsystem OPERABLE, the plant is not prepared to respond to a loss of coolant accident or to continue a cooldown using the RHR pumps and heat exchangers. The Completion Time of immediately to initiate actions that would restore at least one ECCS RHR subsystem to OPERABLE status ensures that prompt action is taken to restore the required cooling capacity.

BASES

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ACTIONS (continued)

With both RHR subsystems inoperable, it would be unwise to require the plant to go to MODE 5, where the only available heat removal system is the RHR. Therefore, the appropriate action is to initiate measures to restore one ECCS RHR subsystem and to continue the actions until the subsystem is restored to OPERABLE status.

B.1

With no ECCS centrifugal charging subsystem OPERABLE, the plant is not prepared to provide high pressure response to Design Basis Events requiring SI. The 1 hour Completion Time to restore at least one ECCS centrifugal charging subsystem to OPERABLE status ensures that prompt action is taken to provide the required cooling capacity.

C.1

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

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SURVEILLANCE  
REQUIREMENTS

SR 3.5.3.1

The applicable Surveillance descriptions from Bases 3.5.2 apply.

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REFERENCES

The applicable references from Bases 3.5.2 apply.

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## B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

### B 3.5.4 Refueling Water Storage Tank (RWST)

#### BASES

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##### BACKGROUND

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

The RWST supplies both trains of the ECCS and the Containment Spray System through a common supply header during the injection phase of a loss of coolant accident (LOCA) recovery. Motor operated isolation valves are provided in the header to isolate the RWST from the ECCS once the system has been transferred to the recirculation mode. The recirculation mode is entered when ECCS pump suction is transferred to the containment sump following receipt of the RWST – Low Level signal coincident with Containment Sump Level – High signal. The transfer of the containment spray pump suction to the containment sump is manually initiated upon receipt of a high level in the containment sump or the RWST Low-Low (Level) alarm. Use of a single RWST to supply both trains of the ECCS and Containment Spray System is acceptable since the RWST is a passive component, and passive failures are not required to be assumed to occur coincidentally with Design Basis Events.

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

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

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

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

BASES

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## BACKGROUND (continued)

This LCO ensures that:

- a. The RWST contains sufficient borated water to support the ECCS during the injection phase;
- b. Sufficient water volume exists in the containment sump to support continued operation of the ECCS and Containment Spray System pumps at the time of transfer to the recirculation mode of cooling; and
- c. The reactor remains subcritical following a LOCA.

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

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APPLICABLE  
SAFETY  
ANALYSES

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

The RWST must also meet volume, boron concentration, and temperature requirements for non-LOCA events. The volume is not an explicit assumption in non-LOCA events since the required volume is a small fraction of the available volume. The deliverable volume limit is set by the LOCA and containment analyses. For the RWST, the deliverable volume is different from the total volume contained since, due to the design of the tank, more water can be contained than can be delivered. The minimum boron concentration is an explicit assumption in the main steam line break (MSLB) analysis to ensure the required shutdown capability.

## BASES

## APPLICABLE SAFETY ANALYSES (continued)

The maximum RWST temperature ensures that the amount of cooling provided from the RWST during the heatup phase of a feedline break is consistent with safety analysis assumptions; the minimum RWST temperature is an assumption in the MSLB analysis.

The MSLB analysis has considered a delay associated with the interlock between the VCT and RWST isolation valves, and the results show that the departure from nucleate boiling design basis is met. The delay has been established as 28 seconds, with offsite power available, or 58 seconds without offsite power.

For a large break LOCA analysis, the minimum water volume limit of 370,000 gallons and the lower boron concentration limit of 2500 ppm are used to compute the post LOCA sump boron concentration necessary to assure subcriticality. The large break LOCA is the limiting case since the safety analysis assumes that all control rods are out of the core.

The upper limit on boron concentration of 2700 ppm is used to determine the maximum allowable time to switch to hot leg recirculation following a LOCA. The purpose of switching from cold leg to hot leg recirculation is to minimize the potential for boron precipitation in the core following the accident.

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

## LCO

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

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

## APPLICABILITY

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

BASES

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## ACTIONS

A.1

With RWST boron concentration or borated water temperature not within limits, they must be returned to within limits within 8 hours. Under these conditions neither the ECCS nor the Containment Spray System can perform its design function. Therefore, prompt action must be taken to restore the RWST to OPERABLE condition. The 8 hour limit to restore the RWST temperature or boron concentration to within limits was developed considering the time required to change either the boron concentration or temperature and the fact that the contents of the tank are still available for injection.

B.1

With the RWST inoperable for reasons other than Condition A (e.g., water volume), it must be restored to OPERABLE status within 1 hour. In this condition, neither the ECCS nor the Containment Spray System can perform its design function. Therefore, prompt action must be taken to restore the RWST to OPERABLE status. The short time limit of 1 hour to restore the RWST to OPERABLE status is based on this condition simultaneously affecting redundant trains.

C.1 and C.2

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

BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.5.4.1

The RWST borated water temperature should be verified to be within the limits assumed in the accident analyses band.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.5.4.2

The RWST water volume should be verified to be above the required minimum level in order to ensure that a sufficient initial supply is available for injection and to support continued ECCS and Containment Spray System pump operation on recirculation.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.5.4.3

The boron concentration of the RWST should be verified to be within the required limits. This SR ensures that the reactor will remain subcritical following a LOCA. Further, it assures that the resulting sump pH will be maintained in an acceptable range so that boron precipitation in the core will not occur and the effect of chloride and caustic stress corrosion on mechanical systems and components will be minimized.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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REFERENCES

1. UFSAR, Chapter 6 and Chapter 15.
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## B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

### B 3.5.5 Seal Injection Flow

#### BASES

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BACKGROUND	<p>The function of the seal injection throttle valves during an accident is similar to the function of the ECCS throttle valves in that each restricts flow from the centrifugal charging pump header to the Reactor Coolant System (RCS).</p> <p>The restriction on reactor coolant pump (RCP) seal injection flow limits the amount of ECCS flow that would be diverted from the injection path following an accident. This limit is based on safety analysis assumptions that are required because RCP seal injection flow is not isolated during safety injection (SI).</p> <p>The RCP seal injection flow is restricted by the seal injection line flow which is adjusted through positioning of the manual RCP seal injection throttle valves. The RCP seal injection flow is determined by measuring the pressurizer pressure, the centrifugal charging pump discharge header pressure, and the RCP seal injection flow rate.</p> <p>The charging flow control valve throttles the centrifugal charging pump discharge header flow as necessary to maintain the programmed level in the pressurizer. The charging flow control valve fails open to ensure that, in the event of either loss of air or loss of control signal to the valve, when the centrifugal charging pumps are supplying charging flow, seal injection flow to the RCP seals is maintained. Positioning of the charging flow control valve may vary during normal plant operating conditions, resulting in a proportional change to RCP seal injection flow. The flow provided by RCP seal injection throttle valves will remain fixed when the charging flow control valve is repositioned provided the throttle valve(s) position are not adjusted.</p>
APPLICABLE SAFETY ANALYSES	<p>All ECCS subsystems are taken credit for in the large break loss of coolant accident (LOCA) at full power (Ref. 1). The LOCA analysis establishes the minimum flow for the ECCS pumps. The centrifugal charging pumps are also credited in the small break LOCA analysis. This analysis establishes the flow and discharge head at the design point for the centrifugal charging pumps. The steam generator tube rupture and main steam line break event analyses also credit the centrifugal charging pumps, but are not limiting in their design. Reference to these analyses is made in assessing changes to the Seal Injection System for evaluation of their effects in relation to the acceptance limits in these analyses.</p>

BASES

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APPLICABLE SAFETY ANALYSES (continued)

This LCO ensures that seal injection flow will be sufficient for RCP seal integrity but limited so that the ECCS trains will be capable of delivering sufficient water to match boiloff rates soon enough to minimize uncovering of the core following a large LOCA. It also ensures that the centrifugal charging pumps will deliver sufficient water for a small LOCA and sufficient boron to maintain the core subcritical. For smaller LOCAs, the charging pumps alone deliver sufficient fluid to overcome the loss and maintain RCS inventory. Seal injection flow satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

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LCO

The intent of the LCO limit on seal injection flow is to make sure that flow through the RCP seal water injection line is low enough to ensure that sufficient centrifugal charging pump injection flow is directed to the RCS via the injection points (Ref. 2).

The LCO is not strictly a flow limit, but rather a flow limit based on a flow line resistance. In order to establish the proper flow line resistance, a pressure and flow must be known. The flow line resistance is established by adjusting the RCP seal injection throttle valves (needle valves) to provide a total seal injection flow in the acceptable region of Figure 3.5.5-1 at a given pressure differential between the charging header and the RCS. The centrifugal charging pump discharge header pressure remains essentially constant through all the applicable MODES of this LCO. A reduction in RCS pressure would result in more flow being diverted to the RCP seal injection line than at normal operating pressure. The valve settings established at the prescribed centrifugal charging pump discharge header pressure result in a conservative valve position should RCS pressure decrease. The flow limits established by Figure 3.5.5-1 ensures that the minimum ECCS flow assumed in the safety analyses is maintained.

The limit on seal injection flow must be met to render the ECCS OPERABLE. If these conditions are not met, the ECCS flow will not be as assumed in the accident analyses.

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APPLICABILITY

In MODES 1, 2, and 3, the seal injection flow limit is dictated by ECCS flow requirements, which are specified for MODES 1, 2, 3, and 4. The seal injection flow limit is not applicable for MODE 4 and lower, however, because high seal injection flow is less critical as a result of the lower initial RCS pressure and decay heat removal requirements in these MODES. Therefore, RCP seal injection flow must be limited in MODES 1, 2, and 3 to ensure adequate ECCS performance.

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BASES

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ACTIONS

A.1

With the seal injection flow not within its limit, the amount of charging flow available to the RCS may be reduced. Under this Condition, action must be taken to restore the flow to within its limit. The operator has 4 hours from the time the flow is known to not be within the limit to correctly position the manual valves and thus be in compliance with the accident analysis. The Completion Time minimizes the potential exposure of the plant to a LOCA with insufficient injection flow and provides a reasonable time to restore seal injection flow within limits. This time is conservative with respect to the Completion Times of other ECCS LCOs; it is based on operating experience and is sufficient for taking corrective actions by operations personnel.

B.1 and B.2

When the Required Actions cannot be completed within the required Completion Time, a controlled shutdown must be initiated. The Completion Time of 6 hours for reaching MODE 3 from MODE 1 is a reasonable time for a controlled shutdown, based on operating experience and normal cooldown rates, and does not challenge plant safety systems or operators. Continuing the plant shutdown begun in Required Action B.1, an additional 6 hours is a reasonable time, based on operating experience and normal cooldown rates, to reach MODE 4, where this LCO is no longer applicable.

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SURVEILLANCE  
REQUIREMENTS

SR 3.5.5.1

Verification that the manual seal injection throttle valves are adjusted to give a flow within the limit ensures that the ECCS injection flows stay within the safety analysis. A differential pressure is established between the charging header and the RCS, and the total seal injection flow is verified to within the limit determined in accordance with the ECCS safety analysis.

The flow shall be verified by confirming seal injection flow and differential pressure within the acceptable region of Figure 3.5.5-1.

Control valves in the flow path between the charging header and the RCS pressure sensing points must be in their post accident position (e.g., charging flow control valve open) during this Surveillance to correlate with the acceptance criteria.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

## BASES

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### SURVEILLANCE REQUIREMENTS (continued)

As noted, the Surveillance is not required to be performed until 4 hours after the RCS pressure has stabilized at  $\geq 2215$  psig and  $\leq 2255$  psig. The RCS pressure requirement is specified since this configuration will produce the required pressure conditions necessary to assure that the manual valves are set correctly. The exception is limited to 4 hours to ensure that the Surveillance is timely.

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### REFERENCES

1. UFSAR, Chapter 6 and Chapter 15.
  2. 10 CFR 50.46.
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## B 3.6 CONTAINMENT SYSTEMS

### B 3.6.1 Containment

#### BASES

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**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 the radioactive material that may be released from the reactor core following a design basis loss of coolant accident (LOCA). Additionally, the containment and shield building provide shielding from the fission products that may be present in the containment atmosphere following accident conditions.

The containment vessel is a vertical cylindrical steel pressure vessel with hemispherical dome and a concrete base mat with steel membrane. It is completely enclosed by a reinforced concrete shield building. An annular space exists between the walls and domes of the steel containment vessel and the concrete shield building to provide for the collection, mixing, holdup, and controlled release of containment out leakage. Ice condenser containments utilize an outer concrete building for shielding and an inner steel containment for leak tightness.

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

The inner steel containment and its penetrations establish the leakage limiting boundary of the containment. Maintaining the containment OPERABLE limits the leakage of fission product radioactivity from the containment to the environment. SR 3.6.1.1 leakage rate requirements comply with 10 CFR 50, Appendix J, Option B (Ref. 1), as modified by approved exemptions.

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

BASES

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BACKGROUND (continued)

- 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 sealing mechanism associated with each containment penetration (e.g., welds, bellows, or O-rings) is OPERABLE (i.e., OPERABLE such that the containment leakage limits are met).
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APPLICABLE  
SAFETY  
ANALYSES

The safety design basis for the containment is that the containment must withstand the pressures and temperatures of the limiting Design Basis Accident (DBA) without exceeding the design leakage rates.

The DBAs that result in a challenge to containment OPERABILITY from high pressures and temperatures are a LOCA, a steam line break, and a rod ejection accident (REA) (Ref. 2). In addition, release of significant fission product radioactivity within containment can occur from a LOCA or REA. In the DBA analyses, it is assumed that the containment is OPERABLE such that, for the DBAs involving release of fission product radioactivity, release to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of 0.25% of containment air weight per day (Ref. 3). This leakage rate, used in the evaluation of offsite doses resulting from accidents, is defined in 10 CFR 50, Appendix J, Option B (Ref. 1), as  $L_a$ : the maximum allowable containment leakage rate at the calculated peak containment internal pressure ( $P_a$ ) resulting from the limiting design basis LOCA. The allowable leakage rate represented by  $L_a$  forms the basis for the acceptance criteria imposed on all containment leakage rate testing.  $L_a$  is assumed to be 0.25% per day in the safety analysis at  $P_a$  (Ref. 3).

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

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

BASES

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LCO Containment OPERABILITY is maintained by limiting leakage to  $\leq 1.0 L_a$ , except prior to the first startup after performing a required Containment Leakage Rate Testing Program leakage test. At this time the applicable leakage limits must be met.

Compliance with this LCO will ensure a containment configuration, including equipment hatches, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analysis.

Individual leakage rates specified for the containment air lock (LCO 3.6.2), 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 overall acceptance criteria of the Containment Leakage Rate Testing Program.

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APPLICABILITY In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material into containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, containment is not required to be OPERABLE in MODE 5 to prevent leakage of radioactive material from containment. The requirements for containment during MODE 6 are addressed in LCO 3.9.4, "Containment Penetrations."

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ACTIONS

A.1

In the event containment is inoperable, containment must be restored to OPERABLE status within 1 hour. The 1 hour Completion Time provides a period of time to correct the problem commensurate with the importance of maintaining containment 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.

BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.6.1.1

Maintaining the containment OPERABLE requires compliance with the visual examinations and leakage rate test requirements of the Containment Leakage Rate Testing Program. The containment concrete visual examinations may be performed during either power operation or during a maintenance or refueling outage. The visual examinations of the steel liner plate inside containment are performed during maintenance or refueling outages since this is the only time the liner plate is fully accessible.

Failure to meet air lock, shield building bypass leakage path, 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 Containment Leakage Rate Testing Program leakage test is required to be  $\leq 0.6 L_a$  for combined Type B and C leakage, and  $\leq 0.75 L_a$  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 the Containment Leakage Rate Testing Program. These periodic testing requirements verify that the containment leakage rate does not exceed the leakage rate assumed in the safety analysis.

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REFERENCES

1. 10 CFR 50, Appendix J, Option B.
  2. UFSAR, Chapter 15.
  3. UFSAR, Section 6.2.
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## B 3.6 CONTAINMENT SYSTEMS

### B 3.6.2 Containment Air Locks

#### BASES

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##### BACKGROUND

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

Each air lock is nominally a right circular cylinder, approximately 8 feet 7 inches 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.

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 analyses.

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##### APPLICABLE SAFETY ANALYSES

The DBAs that result in a release of radioactive material within containment are a loss of coolant accident (LOCA) and a rod ejection accident (Ref. 2). In the analysis of each of these accidents, it is assumed that containment is OPERABLE such that release of fission products to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of 0.25% of containment air weight per day (Ref. 3). This leakage rate is defined in 10 CFR 50, Appendix J, Option B (Ref. 1), as  $L_a = 0.25\%$  of containment air weight per day, the maximum allowable containment leakage rate at the calculated peak containment internal pressure  $P_a$  following a design basis LOCA. This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air locks.

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

## BASES

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

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

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**APPLICABILITY** In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the containment air locks are not required in MODE 5 to prevent leakage of radioactive material from containment. The requirements for the containment air locks during MODE 6 are addressed in LCO 3.9.4, "Containment Penetrations."

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**ACTIONS** The ACTIONS are modified by a Note that allows entry and exit to perform repairs on the affected air lock component. If the outer door is inoperable, then it may be easily accessed for most repairs. It is preferred that the air lock be accessed from inside primary containment by entering through the other OPERABLE air lock. However, if this 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 due to the low probability of an event that could pressurize the containment during the short time in which the OPERABLE door is expected to be open. After each entry and exit, the OPERABLE door must be immediately closed. If ALARA conditions permit, entry and exit should be via an OPERABLE air lock.

## BASES

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### ACTIONS (continued)

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

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

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

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

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

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

## BASES

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### ACTIONS (continued)

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 on a periodic basis 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 is 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 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.

BASES

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ACTIONS (continued)

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 within the 1 hour Completion Time. This specified time period is consistent with the ACTIONS of LCO 3.6.1, which requires that containment be restored to OPERABLE status within 1 hour.

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

D.1 and D.2

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

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SURVEILLANCE  
REQUIREMENTS

SR 3.6.2.1

Maintaining containment air locks OPERABLE requires compliance with the leakage rate test requirements of the Containment Leakage Rate Testing Program. This SR reflects the leakage rate testing requirements with regard to air lock leakage (Type B leakage tests). The acceptance criteria were established during initial air lock and containment OPERABILITY testing. The periodic testing requirements verify that the

## BASES

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### SURVEILLANCE REQUIREMENTS (continued)

air lock leakage does not exceed the allowed fraction of the overall containment leakage rate. The Frequency is required by the Containment Leakage Rate Testing Program.

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

#### SR 3.6.2.2

The air lock interlock is designed to prevent simultaneous opening of both doors in a single air lock. Since both the inner and outer doors of an air lock are designed to withstand the maximum expected post accident containment pressure, closure of either door will support containment OPERABILITY. Thus, the door interlock feature supports containment OPERABILITY while the air lock is being used for personnel transit in and out of the containment. Periodic testing of this interlock demonstrates that the interlock will function as designed and that simultaneous opening of the inner and outer doors will not inadvertently occur.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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- REFERENCES
1. 10 CFR 50, Appendix J, Option B.
  2. UFSAR, Section 15.4.
  3. UFSAR, Section 6.2.4.1.
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## B 3.6 CONTAINMENT SYSTEMS

### B 3.6.3 Containment Isolation Valves

#### BASES

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**BACKGROUND** The containment isolation valves form part of the containment pressure boundary and provide a means for fluid penetrations not serving accident consequence limiting systems to be provided with two isolation barriers that are closed on a containment 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, or an approved exemption is provided, so that no single credible failure or malfunction of an active component can result in a loss of isolation or leakage that exceeds limits assumed in the safety analyses. One of these barriers may be a closed system. These barriers (typically containment isolation valves) make up the Containment Isolation System.

Automatic isolation signals are produced during accident conditions. Containment Phase "A" isolation occurs upon receipt of a safety injection signal. The Phase "A" isolation signal isolates nonessential process lines in order to minimize leakage of fission product radioactivity. Containment Phase "B" isolation occurs upon receipt of a containment pressure High-High signal and isolates the remaining process lines, except systems required for accident mitigation. In addition to the isolation signals listed above, the purge and exhaust valves receive an isolation signal on a containment high radiation condition. As a result, the containment isolation valves (and blind flanges) help ensure that the containment atmosphere will be isolated from the environment in the event of a release of fission product radioactivity to the containment atmosphere as a result of a Design Basis Accident (DBA).

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

BASES

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BACKGROUND (continued)

Reactor Building Purge Ventilation (RBPV) System

The RBPV System operates to supply outside air into the containment for ventilation and cooling or heating and may also be used to reduce the concentration of noble gases within containment prior to and during personnel access. The RBPV System provides for mechanical ventilation of the primary containment, the instrument room located within the containment, and the annulus secondary containment located between primary containment and the Shield Building.

The RBPV System includes one supply duct penetration through the Shield Building wall into the annulus area. There are four purge air supply penetrations through the containment vessel, two to the upper compartment and two to the lower containment. Two normally closed 24-inch purge supply isolation valves at each penetration through the containment vessel provide containment isolation.

The RBPV System includes one exhaust duct penetration through the Shield Building wall from the annulus area. There are three purge air exhaust penetrations through the containment vessel, two from the upper compartment and one from the lower containment. There is one pressure relief penetration through the containment vessel. Two normally closed 24-inch purge exhaust isolation valves at each penetration through the containment vessel provide containment isolation. Two normally closed 8-inch pressure relief isolation valves through the containment vessel provide containment isolation.

The RBPV System includes one supply and one exhaust duct penetration through the Shield Building wall and one supply and one exhaust duct penetration through the containment vessel wall for ventilation of the instrument room inside containment. Two normally closed 12-inch purge isolation valves at each supply and exhaust penetration through the containment vessel provide containment isolation.

Since the valves used in the RBPV System are designed to meet the requirements for automatic containment isolation valves, these valves may be opened as needed in MODES 1, 2, 3, and 4.

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APPLICABLE  
SAFETY  
ANALYSES

The containment isolation valve LCO was derived from the assumptions related to minimizing the loss of reactor coolant inventory and establishing 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 analyses of any event requiring isolation of containment is applicable to this LCO.

BASES

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APPLICABLE SAFETY ANALYSES (continued)

The DBAs that result in a release of radioactive material within containment are a loss of coolant accident (LOCA) and a rod ejection accident (Ref. 1). In the analyses 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 bounding safety analyses assume that one pair of containment purge system lines are open at event initiation. The open purge system lines include one set of supply valves (i.e., inboard and outboard) and one set of exhaust valves (i.e., inboard and outboard).

The DBA analysis assumes that, within 85 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 85 seconds includes signal delay, diesel generator startup (for loss of offsite power), and containment isolation valve stroke times.

The single failure criterion required to be imposed in the conduct of plant safety analyses was considered in the original design of the containment purge valves. Two valves (i.e., one set) 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, pneumatically operated to open and spring closed. This arrangement was designed to preclude common mode failures from disabling both valves on a purge line.

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

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LCO

Containment isolation valves form a part of the containment boundary. The containment isolation valves' 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 containment isolation purge valves have blocks installed to prevent full opening. Blocked purge valves also actuate on an automatic signal. The valves covered by this LCO are listed along with their associated stroke times in the UFSAR (Ref. 2).

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

BASES

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LCO (continued)

intact. These passive isolation valves/devices are those listed in Reference 2.

Purge valves with resilient seals and shield building bypass leakage paths 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.

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APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the containment isolation valves are not required to be OPERABLE in MODE 5. The requirements for containment isolation valves during MODE 6 are addressed in LCO 3.9.4, "Containment Penetrations."

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ACTIONS

The ACTIONS are modified by a Note allowing 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.

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

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

## BASES

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### ACTIONS (continued)

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

Note 5 limits the number of open containment purge lines to no more than one set of supply valves and one set of exhaust valves.

#### A.1 and A.2

Condition A is applicable to penetration flow paths with two containment isolation valves, and penetration flow paths with only one containment isolation valve and a closed system. The closed system must meet the requirements of Reference 3.

In the event one containment isolation valve in one or more penetration flow paths is inoperable, except for inoperable containment vacuum relief isolation valve(s), or shield building bypass or containment purge isolation valve leakage not within limit, the affected penetration flow path must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and deactivated automatic containment isolation valve, a closed manual valve, a blind flange, and a check valve with flow through the valve secured. For a penetration flow path 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 Completion Time specified for each Category of containment isolation valve identified in Table B 3.6.3-1, Containment Isolation Valve Completion Times. The Completion Time is justified in Reference 4.

For affected penetration flow paths that cannot be restored to OPERABLE status within the specified 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 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

## BASES

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### ACTIONS (continued)

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.

Required Action A.2 is modified by two Notes. Note 1 applies to isolation devices located in high radiation areas and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Note 2 applies to isolation devices that are locked, sealed, or otherwise secured in position and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since the function of locking, sealing, or securing components is to ensure that these devices are not inadvertently repositioned. 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 inoperable containment vacuum relief isolation valve(s), or shield building bypass or containment purge isolation valve leakage not within limit, the affected penetration flow path must be isolated within 1 hour. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, a closed manual valve, and a blind flange. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1. In the event the affected penetration is isolated in accordance with Required Action B.1, the affected penetration must be verified to be isolated on a periodic basis per Required Action A.2, which remains in effect. This periodic verification is necessary to assure leak tightness of containment and that penetrations requiring isolation following an accident are isolated. The Completion Time of once per 31 days for verifying each affected penetration flow path is isolated is appropriate considering the fact that the valves are operated under administrative control 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.

BASES

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ACTIONS (continued)

C.1

In the event one containment isolation valve in two or more penetration flow paths is inoperable except for inoperable containment vacuum relief isolation valve(s), or shield building bypass or containment purge isolation valve leakage not within limit, all but one of the affected penetration flow path(s) must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic containment isolation valve, a closed manual valve, a blind flange, and a check valve with flow through the valve secured. For a penetration flow path isolated in accordance with Required Action C.1, the device used to isolate the penetration should be the closest available one to containment. Required Action C.1 must be completed within 4 hours. For subsequent containment isolation valve inoperabilities, the Required Action and Completion Time continue to apply to each additional containment isolation valve inoperability, with the Completion Time based on each subsequent entry into the Condition consistent with Note 2 to the ACTIONS Table (e.g., for each entry into the Condition). Each containment isolation valve(s) that is (are) declared inoperable for subsequent Condition C entries shall meet the Required Action and Completion Time. For the penetration flow paths isolated in accordance with Required Action C.1, the affected penetration(s) 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 that the penetrations requiring isolation following an accident are isolated. 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.

D.1 and D.2

In the event two or more pairs of containment purge lines are open, all but one penetration flow paths 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, and a blind flange. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1. The 1 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.

## BASES

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### ACTIONS (continued)

#### E.1

With one or more containment vacuum relief isolation valve(s) inoperable, the inoperable valve(s) must be restored to OPERABLE status or the affected penetration flow path must be isolated.

In the containment vacuum relief lines, containment vacuum relief valves 30-571, 30-572, and 30-573 are qualified to perform a containment isolation function. These valves are self-actuated, swing disk (check) valves with an elastomer seat. The valves are normally closed and are equipped with limit switches that provide fully open and fully closed indication in the main control room (MCR). Therefore, a 72 hour Completion Time is appropriate while actions are taken to return the containment vacuum relief isolation valves to service.

#### F.1

With the shield building bypass leakage rate (SR 3.6.3.8) not within limit, the assumptions of the safety analyses are not met. Therefore, the leakage must be restored to within limit. 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.

#### G.1, G.2, and G.3

In the event one or more containment purge supply or exhaust valves in one or more penetration flow paths are not within the purge supply and exhaust valve leakage limits, purge supply and exhaust valve leakage must be restored to within limits, or the affected penetration flow path must be isolated. The method of isolation must be by the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, closed manual valve, or blind flange. A purge valve with resilient seals utilized to satisfy Required Action G.1

## BASES

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### ACTIONS (continued)

must have been demonstrated to meet the leakage requirements of SR 3.6.3.5. 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 G.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 that those 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 a resilient seal that is isolated in accordance with Required Action G.1, SR 3.6.3.5 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. Since more reliance is placed on a single valve while in this Condition, a Frequency of once per 92 days was chosen and has been shown to be acceptable based on operating experience.

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

BASES

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ACTIONS (continued)

H.1 and H.2

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

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SURVEILLANCE  
REQUIREMENTS

SR 3.6.3.1

This SR ensures that the containment purge 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 containment purge valves are open for the reasons stated. The valves may be opened for pressure control, ALARA or air quality considerations for personnel entry, or for Surveillances that require the valves to be open. The number of valves open during MODES 1, 2, 3, and 4, is limited to no more than one pair of containment purge lines, that includes one set of supply valves and one set of exhaust valves. The containment purge valves are capable of closing in the environment following a LOCA. Therefore, these valves are allowed to be open for limited periods of time.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.6.3.2

This SR requires verification that each containment isolation manual valve and blind flange located outside containment and not locked, sealed, or otherwise secured and required to be closed during accident conditions is closed. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside of the containment boundary is within design limits. This SR does not require any testing or valve manipulation. Rather, it involves verification that those containment isolation valves outside containment and capable of being mispositioned are in the correct position.

## BASES

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### SURVEILLANCE REQUIREMENTS (continued)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

The SR specifies that containment isolation valves that are open under administrative controls are not required to meet the SR during the time the valves are open. This SR does not apply to valves that are locked, sealed, or otherwise secured in the closed position, since these were verified to be in the correct position upon locking, sealing, or securing.

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

#### SR 3.6.3.3

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

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

BASES

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.3.4

Verifying that the isolation time of each automatic power operated containment isolation valve is within limits is required to demonstrate OPERABILITY. The isolation time test ensures the valve will isolate in a time period less than or equal to that assumed in the safety analyses.

The isolation time and Frequency of this SR is in accordance with the Inservice Testing Program.

SR 3.6.3.5

For containment purge valves with resilient seals, additional leakage rate testing beyond the test requirements of 10 CFR 50, Appendix J, Option B, is required to ensure OPERABILITY. Operating experience has demonstrated that this type of seal has the potential to degrade in a shorter time period than do other seal types.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.6.3.6

Automatic containment isolation valves close on a containment isolation signal to prevent leakage of radioactive material from containment following a DBA. This SR ensures that each automatic containment isolation valve will actuate to its isolation position on a containment isolation signal. This surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.6.3.7

Verifying that each containment purge valve is blocked to restrict opening to < 50 degrees is required to ensure that the valves can close under DBA conditions within the times assumed in the analyses of References 1 and 2. If a LOCA occurs, the purge valves must close to maintain

BASES

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SURVEILLANCE REQUIREMENTS (continued)

containment leakage within the values assumed in the accident analysis. At other times when purge valves are required to be capable of closing (e.g., during movement of irradiated fuel assemblies), pressurization concerns are not present, thus the purge valves can be fully open.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.6.3.8

This SR ensures that the combined leakage rate of all shield building bypass leakage paths (those paths that would potentially allow leakage from the primary containment to circumvent the annulus secondary containment enclosure and escape to the auxiliary building secondary enclosure) is less than or equal to the specified leakage rate. This provides assurance that the assumptions in the safety analysis are met. The as-left bypass leakage rate prior to the first startup after performing a leakage test requires calculation using maximum pathway leakage (leakage through the worse of the two isolation valves). If the penetration is isolated by use of one closed and de-activated automatic valve, closed manual valve, or blind flange, then 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. Following startup, the as-found leakage rate will be calculated using the minimum pathway leakage. The Frequency is required by the Containment Leakage Rate Testing Program. This SR simply imposes additional acceptance criteria.

Bypass leakage is considered part of  $L_a$ .

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- REFERENCES
1. UFSAR, Chapter 15.
  2. UFSAR, Section 6.2.4 and Table 6.2.4-1.
  3. Standard Review Plan 6.2.4.
  4. WCAP-15791-A, Rev. 2, "Risk-Informed Evaluation of Extensions to Containment Isolation Valve Completion Times," June 2008.
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Table B 3.6.3-1 (Page 1 of 18)  
Containment Isolation Valve Completion Times

UNID	Penetration	Pressure Boundary Maintained		Pressure Boundary Compromised	
		Category	Completion Time	Category	Completion Time
FCV-01-7	X-14D	2	8 hours	9	8 hours
FCV-01-14	X-14A	2	8 hours	9	8 hours
FCV-01-25	X-14C	2	8 hours	9	8 hours
FCV-01-32	X-14B	2	8 hours	9	8 hours
FCV-01-147	X-13A	2	8 hours	9	8 hours
FCV-01-148	X-13B	2	8 hours	9	8 hours
FCV-01-149	X-13C	2	8 hours	9	8 hours
FCV-01-150	X-13D	2	8 hours	9	8 hours
FCV-01-181	X-14D	2	8 hours	9	8 hours
FCV-01-182	X-14A	2	8 hours	9	8 hours
FCV-01-183	X-14C	2	8 hours	9	8 hours
FCV-01-184	X-14B	2	8 hours	9	8 hours
VLV-01-532	X-13C	2	8 hours	9	8 hours
VLV-01-534	X-13B	2	8 hours	9	8 hours
VLV-01-536	X-13A	2	8 hours	9	8 hours
VLV-01-538	X-13D	2	8 hours	9	8 hours
VLV-01-824	X-14D	2	8 hours	9	8 hours
VLV-01-825	X-14A	2	8 hours	9	8 hours
VLV-01-826	X-14C	2	8 hours	9	8 hours
VLV-01-827	X-14B	2	8 hours	9	8 hours
VLV-01-922	X-13A	2	8 hours	9	8 hours
VLV-01-923	X-13B	2	8 hours	9	8 hours
VLV-01-924	X-13C	2	8 hours	9	8 hours
VLV-01-925	X-13D	2	8 hours	9	8 hours
VLV-03-351C	X-104	2	8 hours	9	8 hours
VLV-03-352C	X-102	2	8 hours	9	8 hours
VLV-03-500	X-12C	2	8 hours	9	8 hours
VLV-03-502	X-12B	2	8 hours	9	8 hours
VLV-03-504	X-12A	2	8 hours	9	8 hours
VLV-03-506	X-12D	2	8 hours	9	8 hours
VLV-03-842	X-40B	1	4 hours	8	4 Hours
VLV-03-847	X-40B	1	4 hours	8	4 Hours

Table B 3.6.3-1 (Page 2 of 18)  
Containment Isolation Valve Completion Times

UNID	Penetration	Pressure Boundary Maintained		Pressure Boundary Compromised	
		Category	Completion Time	Category	Completion Time
VLV-03-848	X-40A	1	4 hours	8	4 Hours
VLV-03-849	X-12A	2	8 hours	9	8 hours
VLV-03-850	X-12D	2	8 hours	9	8 hours
VLV-03-851	X-40B	1	4 hours	8	4 Hours
VLV-03-852	X-40A	1	4 hours	8	4 Hours
VLV-03-853	X-12A	2	8 hours	9	8 hours
VLV-03-854	X-12D	2	8 hours	9	8 hours
VLV-03-855	X-40B	1	4 hours	8	4 Hours
VLV-03-857	X-12A	2	8 hours	9	8 hours
VLV-03-858	X-12D	2	8 hours	9	8 hours
VLV-03-859	X-40B	1	4 hours	8	4 Hours
VLV-03-860	X-40A	1	4 hours	8	4 Hours
VLV-03-887	X-40B	1	4 hours	8	4 Hours
VLV-03-888	X-40A	1	4 hours	8	4 Hours
VLV-03-889	X-12A	2	8 hours	9	8 hours
VLV-03-890	X-12D	2	8 hours	9	8 hours
VLV-03-896	X-40B	1	4 hours	8	4 Hours
VLV-03-897	X-40B	1	4 hours	8	4 Hours
VLV-03-899	X-40A	1	4 hours	8	4 Hours
VLV-03-900	X-40A	1	4 hours	8	4 Hours
VLV-03-901	X-40A	1	4 hours	8	4 Hours
VLV-03-903	X-12A	2	8 hours	9	8 hours
VLV-03-904	X-12A	2	8 hours	9	8 hours
VLV-03-906	X-12D	2	8 hours	9	8 hours
VLV-03-907	X-12D	2	8 hours	9	8 hours
VLV-03-970	X-104	7	7 days	14	7 days
VLV-03-971	X-104	7	7 days	14	7 days
VLV-03-972	X-102	7	7 days	14	7 days
FCV-26-240	X-51	7	7 days	14	7 days
FCV-26-243	X-78	7	7 days	14	7 days
VLV-26-1258	X-51	7	7 days	14	7 days
VLV-26-1260	X-51	7	7 days	14	7 days

Table B 3.6.3-1 (Page 3 of 18)  
Containment Isolation Valve Completion Times

UNID	Penetration	Pressure Boundary Maintained		Pressure Boundary Compromised	
		Category	Completion Time	Category	Completion Time
VLV-26-1293	X-78	7	7 days	14	7 days
VLV-26-1296	X-78	7	7 days	14	7 days
DRIV-30-30CX	X-27A	7	7 days	14	7 days
DRIV-30-30CY	X-27A	7	7 days	14	7 days
DRIV-30-42X	X-27B	7	7 days	14	7 days
DRIV-30-42Y	X-27B	7	7 days	14	7 days
DRIV-30-43X	X-26A	7	7 days	14	7 days
DRIV-30-43Y	X-26A	7	7 days	14	7 days
DRIV30-44X	X-25B	7	7 days	14	7 days
DRIV-30-44Y	X-25B	7	7 days	14	7 days
DRIV-30-45X	X-85B	7	7 days	14	7 days
DRIV-30-45Y	X-85B	7	7 days	14	7 days
DRIV-30-46AX	X-111	7	7 days	14	7 days
DRIV-30-46AY	X-111	7	7 days	14	7 days
DRIV-30-46BY	X-111	7	7 days	14	7 days
DRIV-30-47AX	X-112	7	7 days	14	7 days
DRIV-30-47AY	X-112	7	7 days	14	7 days
DRIV-30-47BY	X-112	7	7 days	14	7 days
DRIV-30-48AX	X-113	7	7 days	14	7 days
DRIV-30-48AY	X-113	7	7 days	14	7 days
DRIV-30-48BY	X-113	7	7 days	14	7 days
DRIV-30-310X	X-26A	7	7 days	14	7 days
DRIV-30-310Y	X-26A	7	7 days	14	7 days
DRIV-30-311X	X-25B	7	7 days	14	7 days
DRIV-30-311Y	X-25B	7	7 days	14	7 days
FCV-30-7	X-9A	7	7 days	14	7 days
FCV-30-8	X-9A	7	7 days	14	7 days
FCV-30-9	X-9B	7	7 days	14	7 days
FCV-30-10	X-9B	7	7 days	14	7 days
FCV-30-14	X-10A	7	7 days	14	7 days
FCV-30-15	X-10A	7	7 days	14	7 days
FCV-30-16	X-10B	7	7 days	14	7 days

Table B 3.6.3-1 (Page 4 of 18)  
Containment Isolation Valve Completion Times

UNID	Penetration	Pressure Boundary Maintained		Pressure Boundary Compromised	
		Category	Completion Time	Category	Completion Time
FCV-30-17	X-10B	7	7 days	14	7 days
FCV-30-19	X-11	7	7 days	14	7 days
FCV-30-20	X-11	7	7 days	14	7 days
FCV-30-37	X-80	7	7 days	14	7 days
FCV-30-40	X-80	7	7 days	14	7 days
FCV-30-46	X-111	7	7 days	14	7 days
FCV-30-47	X-112	7	7 days	14	7 days
FCV-30-48	X-113	7	7 days	14	7 days
FCV-30-50	X-6	7	7 days	14	7 days
FCV-30-51	X-6	7	7 days	14	7 days
FCV-30-52	X-7	7	7 days	14	7 days
FCV-30-53	X-7	7	7 days	14	7 days
FCV-30-56	X-4	7	7 days	14	7 days
FCV-30-57	X-4	7	7 days	14	7 days
FCV-30-58	X-5	7	7 days	14	7 days
FCV-30-59	X-5	7	7 days	14	7 days
FSV-30-134	X-97	7	7 days	14	7 days
FSV-30-135	X-97	7	7 days	14	7 days
PDT-30-30C	X-27A	7	7 days	14	7 days
PDT-30-42	X-27B	7	7 days	14	7 days
PDT-30-43	X-26A	7	7 days	14	7 days
PDT-30-44	X-25B	7	7 days	14	7 days
PDT-30-45	X-85B	7	7 days	14	7 days
PDT-30-310	X-26A	7	7 days	14	7 days
PDT-30-311	X-25B	7	7 days	14	7 days
VLV-30-554TP	X-5	7	7 days	14	7 days
VLV-30-555TP	X-4	7	7 days	14	7 days
VLV-30-556TP	X-80	7	7 days	14	7 days
VLV-30-557TP	X-7	7	7 days	14	7 days
VLV-30-558TP	X-6	7	7 days	14	7 days
VLV-30-559TP	X-11	7	7 days	14	7 days
VLV-30-560TP	X-10B	7	7 days	14	7 days

Table B 3.6.3-1 (Page 5 of 18)  
Containment Isolation Valve Completion Times

UNID	Penetration	Pressure Boundary Maintained		Pressure Boundary Compromised	
		Category	Completion Time	Category	Completion Time
VLV-30-561TP	X-10A	7	7 days	14	7 days
VLV-30-562TP	X-9B	7	7 days	14	7 days
VLV-30-563TP	X-9A	7	7 days	14	7 days
VLV-30-571	X-111	7	7 days	14	7 days
VLV-30-572	X-112	7	7 days	14	7 days
VLV-30-573	X-113	7	7 days	14	7 days
FCV-31C-222	X-64	7	7 days	14	7 days
FCV-31C-223	X-64	7	7 days	14	7 days
FCV-31C-224	X-65	7	7 days	14	7 days
FCV-31C-225	X-65	7	7 days	14	7 days
FCV-31C-229	X-66	7	7 days	14	7 days
FCV-31C-230	X-66	7	7 days	14	7 days
FCV-31C-231	X-67	7	7 days	14	7 days
FCV-31C-232	X-67	7	7 days	14	7 days
VLV-31C-697	X-67	7	7 days	14	7 days
VLV-31C-715	X-66	7	7 days	14	7 days
VLV-31C-734	X-65	7	7 days	14	7 days
VLV-31C-752	X-64	7	7 days	14	7 days
FCV-32-81	X-90	7	7 days	14	7 days
FCV-32-103	X-26B	7	7 days	14	7 days
FCV-32-110	X-34	7	7 days	14	7 days
FCV-32-111	X-34	7	7 days	14	7 days
VLV-32-341	X-26B	7	7 days	14	7 days
VLV-32-345	X-26B	7	7 days	14	7 days
VLV-32-348	X-26B	7	7 days	14	7 days
VLV-32-353	X-90	7	7 days	14	7 days
VLV-32-354	X-90	7	7 days	14	7 days
VLV-32-358	X-90	7	7 days	14	7 days
VLV-32-373	X-34	7	7 days	14	7 days
VLV-32-375	X-34	7	7 days	14	7 days
VLV-32-377	X-34	7	7 days	14	7 days
VLV-32-383	X-34	7	7 days	14	7 days

Table B 3.6.3-1 (Page 6 of 18)  
Containment Isolation Valve Completion Times

UNID	Penetration	Pressure Boundary Maintained		Pressure Boundary Compromised	
		Category	Completion Time	Category	Completion Time
VLV-32-385	X-34	7	7 days	14	7 days
VLV-32-387	X-34	7	7 days	14	7 days
BLF - Sys 33	X-40D	7	7 days	14	7 days
VLV-33-211	X-76	7	7 days	14	7 days
VLV-33-722	X-76	7	7 days	14	7 days
VLV-33-739	X-76	7	7 days	14	7 days
CKV-43-460	X-106	7	7 days	14	7 days
CKV-43-461	X-103	7	7 days	14	7 days
FCV-43-3	X-25A	7	7 days	14	7 days
FCV-43-12	X-25D	7	7 days	14	7 days
FCV-43-23	X-96C	7	7 days	14	7 days
FCV-43-35	X-93	7	7 days	14	7 days
FSV-43-2	X-25A	7	7 days	14	7 days
FSV-43-11	X-25D	7	7 days	14	7 days
FSV-43-22	X-96C	7	7 days	14	7 days
FSV-43-34	X-93	7	7 days	14	7 days
FCV-43-55	X-14D	2	8 hours	9	8 hours
FCV-43-58	X-14A	2	8 hours	9	8 hours
FCV-43-61	X-14C	2	8 hours	9	8 hours
FCV-43-64	X-14B	2	8 hours	9	8 hours
FSV-43-200A	X-99, X-100	7	7 days	14	7 days
FSV-43-200I	X-99, X-100	7	7 days	14	7 days
FSV-43-201	X-99, X-100	7	7 days	14	7 days
FSV-43-202	X-99, X-100	7	7 days	14	7 days
FSV-43-207	X-92A, X-92B	7	7 days	14	7 days
FSV-43-208	X-92A, X-92B	7	7 days	14	7 days
FSV-43-210A	X-92A, X-92B	7	7 days	14	7 days
FSV-43-210I	X-92A, X-92B	7	7 days	14	7 days

Table B 3.6.3-1 (Page 7 of 18)  
Containment Isolation Valve Completion Times

UNID	Penetration	Pressure Boundary Maintained		Pressure Boundary Compromised	
		Category	Completion Time	Category	Completion Time
FSV-43-250	X-91	7	7 days	14	7 days
FSV-43-251	X-91	7	7 days	14	7 days
FSV-43-287	X-116A	7	7 days	14	7 days
FSV-43-288	X-116A	7	7 days	14	7 days
FSV-43-307	X-106	7	7 days	14	7 days
FSV-43-309	X-23	7	7 days	14	7 days
FSV-43-310	X-23	7	7 days	14	7 days
FSV-43-317	X-103	7	7 days	14	7 days
FSV-43-318	X-101	7	7 days	14	7 days
FSV-43-319	X-101	7	7 days	14	7 days
FSV-43-325	X-106	7	7 days	14	7 days
FSV-43-341	X-103	7	7 days	14	7 days
TV-43-464	X-103	7	7 days	14	7 days
TV-43-469	X-106	7	7 days	14	7 days
TV-43-474	X-101	7	7 days	14	7 days
TV-43-477	X-116A	7	7 days	14	7 days
VLV-43-417	X-92A, X-92B	7	7 days	14	7 days
VLV-43-419	X-99, X-100	7	7 days	14	7 days
VLV-43-421	X-92A, X-92B	7	7 days	14	7 days
VLV-43-423	X-99, X-100	7	7 days	14	7 days
VLV-43-424	X-92A, X-92B	7	7 days	14	7 days
VLV-43-426	X-99, X-100	7	7 days	14	7 days
VLV-43-427	X-99, X-100	7	7 days	14	7 days
VLV-43-492	X-23	7	7 days	14	7 days
VLV-43-497	X-91	7	7 days	14	7 days

Table B 3.6.3-1 (Page 8 of 18)  
Containment Isolation Valve Completion Times

UNID	Penetration	Pressure Boundary Maintained		Pressure Boundary Compromised	
		Category	Completion Time	Category	Completion Time
VLV-43-525	X-92A, X-92B	7	7 days	14	7 days
TTIV-52-508	X-98	7	7 days	14	7 days
TTIV-52-510	X-27C	7	7 days	14	7 days
VLV-52-500	X-87D	7	7 days	14	7 days
VLV-52-501	X-87D	7	7 days	14	7 days
VLV-52-502	X-87B	7	7 days	14	7 days
VLV-52-503	X-87B	7	7 days	14	7 days
VLV-52-504	X-27C	7	7 days	14	7 days
VLV-52-505	X-27C	7	7 days	14	7 days
VLV-52-506	X-98	7	7 days	14	7 days
VLV-52-507	X-98	7	7 days	14	7 days
VLV-59-522	X-77	7	7 days	14	7 days
VLV-59-529	X-77	7	7 days	14	7 days
VLV-59-633	X-77	7	7 days	14	7 days
VLV-59-651	X-77	7	7 days	14	7 days
VLV-59-704	X-77	7	7 days	14	7 days
BLF - Sys 61	X-79A	7	7 days	13	72 hours
BLF - Sys 61	X-79B	7	7 days	13	72 hours
FCV-61-96	X-115	7	7 days	14	7 days
FCV-61-97	X-115	7	7 days	14	7 days
FCV-61-110	X-114	7	7 days	14	7 days
FCV-61-122	X-114	7	7 days	14	7 days
FCV-61-191	X-47A	7	7 days	14	7 days
FCV-61-192	X-47A	7	7 days	14	7 days
FCV-61-193	X-47B	7	7 days	14	7 days
FCV-61-194	X-47B	7	7 days	14	7 days
VLV-61-532	X-47A	7	7 days	14	7 days
VLV-61-533	X-47A	7	7 days	14	7 days
VLV-61-680	X-47B	7	7 days	14	7 days
VLV-61-681	X-47B	7	7 days	14	7 days
VLV-61-691	X-115	7	7 days	14	7 days

Table B 3.6.3-1 (Page 9 of 18)  
Containment Isolation Valve Completion Times

UNID	Penetration	Pressure Boundary Maintained		Pressure Boundary Compromised	
		Category	Completion Time	Category	Completion Time
VLV-61-692	X-115	7	7 days	14	7 days
VLV-61-745	X-114	7	7 days	14	7 days
VLV-61-746	X-114	7	7 days	14	7 days
FCV-62-61	X-44	7	7 days	14	7 days
FCV-62-63	X-44	7	7 days	14	7 days
FCV-62-72	X-15	7	7 days	14	7 days
FCV-62-73	X-15	7	7 days	14	7 days
FCV-62-74	X-15	7	7 days	14	7 days
FCV-62-77	X-15	7	7 days	14	7 days
FCV-62-90	X-16	7	7 days	14	7 days
VLV-62-505	X-24	7	7 days	14	7 days
VLV-62-543	X-16	7	7 days	14	7 days
VLV-62-544	X-16	7	7 days	14	7 days
VLV-62-546	X-43A, X-43B, X-43C, X-43D	7	7 days	14	7 days
VLV-62-549	X-43A, X-43B, X-43C, X-43D	7	7 days	14	7 days
VLV-62-550	X-43A, X-43B, X-43C, X-43D	7	7 days	14	7 days
VLV-62-555	X-43A, X-43B, X-43C, X-43D	7	7 days	14	7 days
VLV-62-560	X-43D	7	7 days	14	7 days
VLV-62-561	X-43B	7	7 days	14	7 days
VLV-62-562	X-43C	7	7 days	14	7 days
VLV-62-563	X-43A	7	7 days	14	7 days
VLV-62-568	X-43D	7	7 days	14	7 days
VLV-62-569	X-43B	7	7 days	14	7 days

Table B 3.6.3-1 (Page 10 of 18)  
Containment Isolation Valve Completion Times

UNID	Penetration	Pressure Boundary Maintained		Pressure Boundary Compromised	
		Category	Completion Time	Category	Completion Time
VLV-62-570	X-43C	7	7 days	14	7 days
VLV-62-571	X-43A	7	7 days	14	7 days
VLV-62-572	X-43D	7	7 days	14	7 days
VLV-62-573	X-43B	7	7 days	14	7 days
VLV-62-574	X-43C	7	7 days	14	7 days
VLV-62-575	X-43A	7	7 days	14	7 days
VLV-62-576	X-43D	7	7 days	14	7 days
VLV-62-577	X-43B	7	7 days	14	7 days
VLV-62-578	X-43A	7	7 days	14	7 days
VLV-62-579	X-43C	7	7 days	14	7 days
VLV-62-639	X-44	7	7 days	14	7 days
VLV-62-662	X-15	7	7 days	14	7 days
VLV-62-707	X-15	7	7 days	14	7 days
VLV-62-709	X-16	7	7 days	14	7 days
FCV-63-21	X-32	7	7 days	14	7 days
FCV-63-22	X-33	1	4 hours	8	4 hours
FCV-63-23	X-30	7	7 days	14	7 days
FCV-63-25	X-22	1	4 hours	8	4 hours
FCV-63-26	X-22	1	4 hours	8	4 hours
FCV-63-64	X-39A	7	7 days	14	7 days
FCV-63-71	X-30	7	7 days	14	7 days
FCV-63-72	X-19A	1	4 hours	8	4 hours
FCV-63-73	X-19B	1	4 hours	8	4 hours
FCV-63-84	X-30	7	7 days	14	7 days
FCV-63-93	X-20B	7	7 days	14	7 days
FCV-63-94	X-20A	7	7 days	14	7 days
FCV-63-111	X-20B	7	7 days	14	7 days
FCV-63-112	X-20A	7	7 days	14	7 days
FCV-63-121	X-33	7	7 days	14	7 days
FCV-63-156	X-32	7	7 days	14	4 hours
FCV-63-157	X-21	7	7 days	14	4 hours
FCV-63-158	X-17	7	7 days	14	7 days

Table B 3.6.3-1 (Page 11 of 18)  
Containment Isolation Valve Completion Times

UNID	Penetration	Pressure Boundary Maintained		Pressure Boundary Compromised	
		Category	Completion Time	Category	Completion Time
FCV-63-167	X-21	7	7 days	14	7 days
FCV-63-172	X-17	1	4 hours	8	4 hours
FCV-63-174	X-22	7	7 days	14	7 days
FSV-63-25	X-22	7	7 days	14	7 days
FSV-63-26	X-22	7	7 days	14	7 days
VLV-63-311A	X-32	7	7 days	14	7 days
VLV-63-313A	X-21	7	7 days	14	7 days
VLV-63-314A	X-21	7	7 days	14	7 days
VLV-63-315A	X-32	7	7 days	14	7 days
VLV-63-316A	X-32	7	7 days	14	7 days
VLV-63-317A	X-21	7	7 days	14	7 days
VLV-63-318A	X-21	7	7 days	14	7 days
VLV-63-319A	X-33	7	7 days	14	7 days
VLV-63-320A	X-33	7	7 days	14	7 days
VLV-63-321A	X-33	7	7 days	14	7 days
VLV-63-322A	X-33	7	7 days	14	7 days
VLV-63-323A	X-33	7	7 days	14	7 days
VLV-63-324A	X-33	7	7 days	14	7 days
VLV-63-325A	X-33	7	7 days	14	7 days
VLV-63-326A	X-33	7	7 days	14	7 days
VLV-63-344A	X-30	7	7 days	14	7 days
VLV-63-413	X-20B	7	7 days	14	7 days
VLV-63-511	X-24	7	7 days	14	7 days
VLV-63-534	X-24	7	7 days	14	7 days
VLV-63-535	X-24	7	7 days	14	7 days
VLV-63-536	X-24	7	7 days	14	7 days
VLV-63-537	X-30	7	7 days	14	7 days
VLV-63-541	X-32	7	7 days	14	7 days

Table B 3.6.3-1 (Page 12 of 18)  
Containment Isolation Valve Completion Times

UNID	Penetration	Pressure Boundary Maintained		Pressure Boundary Compromised	
		Category	Completion Time	Category	Completion Time
VLV-63-543	X-32	7	7 days	8	4 Hours
VLV-63-545	X-32	7	7 days	8	4 Hours
VLV-63-547	X-21	7	7 days	8	4 Hours
VLV-63-549	X-21	7	7 days	8	4 Hours
VLV-63-551	X-33	1	4 hours	8	4 Hours
VLV-63-553	X-33	1	4 hours	8	4 Hours
VLV-63-555	X-33	1	4 hours	8	4 Hours
VLV-63-557	X-33	1	4 hours	8	4 Hours
VLV-63-581	X-22	1	4 hours	8	4 Hours
VLV-63-590	X-17	7	7 days	14	7 days
VLV-63-591	X-17	7	7 days	14	7 days
VLV-63-592	X-17	7	7 days	14	7 days
VLV-63-593	X-17	7	7 days	14	7 days
VLV-63-612A	X-32	7	7 days	14	7 days
VLV-63-626	X-24	7	7 days	14	7 days
VLV-63-627	X-24	7	7 days	14	7 days
VLV-63-630	X-20B	7	7 days	14	7 days
VLV-63-631	X-20A	7	7 days	14	7 days
VLV-63-632	X-20B	7	7 days	14	7 days
VLV-63-633	X-20A	7	7 days	14	7 days
VLV-63-634	X-20B	7	7 days	14	7 days
VLV-63-635	X-20A	7	7 days	14	7 days
VLV-63-636	X-17	7	7 days	14	7 days
VLV-63-637	X-17	7	7 days	14	7 days
VLV-63-638	X-24	7	7 days	14	7 days
VLV-63-640	X-17	1	4 hours	8	4 hours
VLV-63-642	X-17	7	7 days	14	7 days
VLV-63-643	X-17	1	4 hours	8	4 hours
VLV-63-648	X-21	7	7 days	14	7 days
VLV-63-649	X-21	7	7 days	14	7 days
VLV-63-650	X-21	7	7 days	14	7 days
VLV-63-653	X-33	7	7 days	14	7 days

Table B 3.6.3-1 (Page 13 of 18)  
Containment Isolation Valve Completion Times

UNID	Penetration	Pressure Boundary Maintained		Pressure Boundary Compromised	
		Category	Completion Time	Category	Completion Time
VLV-63-654	X-33	7	7 days	14	7 days
VLV-63-655	X-33	7	7 days	14	7 days
VLV-63-656	X-33	7	7 days	14	7 days
VLV-63-657	X-32	7	7 days	14	7 days
VLV-63-658	X-32	7	7 days	14	7 days
VLV-63-659	X-20B	7	7 days	14	7 days
VLV-63-660	X-20B	7	7 days	14	7 days
VLV-63-661	X-20A	7	7 days	14	7 days
VLV-63-667	X-20A	7	7 days	14	7 days
VLV-63-816	X-22	7	7 days	14	7 days
VLV-63-823	X-32	7	7 days	14	7 days
VLV-63-831	X-33	7	7 days	14	7 days
VLV-63-833	X-20A	7	7 days	14	7 days
VLV-63-836	X-33	7	7 days	14	7 days
VLV-63-862	X-21	7	7 days	14	7 days
VLV-63-864	X-32	7	7 days	14	7 days
VLV-63-870	X-17	7	7 days	14	7 days
CKV-67-1523A	X-58	7	7 days	14	7 days
CKV-67-1523B	X-60	7	7 days	14	7 days
CKV-67-1523C	X-62	7	7 days	14	7 days
CKV-67-1523D	X-56	7	7 days	14	7 days
FCV-67-83	X-56	7	7 days	14	7 days
FCV-67-87	X-59	7	7 days	14	7 days
FCV-67-88	X-59	7	7 days	14	7 days
FCV-67-89	X-56	7	7 days	14	7 days
FCV-67-90	X-60	7	7 days	14	7 days
FCV-67-91	X-60	7	7 days	14	7 days
FCV-67-95	X-63	7	7 days	14	7 days
FCV-67-96	X-63	7	7 days	14	7 days
FCV-67-99	X-62	7	7 days	14	7 days
FCV-67-103	X-61	7	7 days	14	7 days
FCV-67-104	X-61	7	7 days	14	7 days

Table B 3.6.3-1 (Page 14 of 18)  
Containment Isolation Valve Completion Times

UNID	Penetration	Pressure Boundary Maintained		Pressure Boundary Compromised	
		Category	Completion Time	Category	Completion Time
FCV-67-105	X-62	7	7 days	14	7 days
FCV-67-106	X-58	7	7 days	14	7 days
FCV-67-107	X-58	7	7 days	14	7 days
FCV-67-111	X-57	7	7 days	14	7 days
FCV-67-112	X-57	7	7 days	14	7 days
FCV-67-130	X-69	7	7 days	14	7 days
FCV-67-131	X-73	7	7 days	14	7 days
FCV-67-133	X-75	7	7 days	14	7 days
FCV-67-134	X-71	7	7 days	14	7 days
FCV-67-138	X-74	7	7 days	14	7 days
FCV-67-139	X-70	7	7 days	14	7 days
FCV-67-141	X-68	7	7 days	14	7 days
FCV-67-142	X-72	7	7 days	14	7 days
FCV-67-295	X-73	7	7 days	14	7 days
FCV-67-296	X-71	7	7 days	14	7 days
FCV-67-297	X-70	7	7 days	14	7 days
FCV-67-298	X-72	7	7 days	14	7 days
VLV-67-561A	X-58	7	7 days	14	7 days
VLV-67-561B	X-60	7	7 days	14	7 days
VLV-67-561C	X-62	7	7 days	14	7 days
VLV-67-561D	X-56	7	7 days	14	7 days
VLV-67-575A	X-59	7	7 days	14	7 days
VLV-67-575B	X-61	7	7 days	14	7 days
VLV-67-575C	X-63	7	7 days	14	7 days
VLV-67-575D	X-57	7	7 days	14	7 days
VLV-67-579A	X-69	7	7 days	14	7 days
VLV-67-579B	X-74	7	7 days	14	7 days
VLV-67-579C	X-75	7	7 days	14	7 days
VLV-67-579D	X-68	7	7 days	14	7 days
VLV-67-580A	X-69	7	7 days	14	7 days
VLV-67-580B	X-74	7	7 days	14	7 days
VLV-67-580C	X-75	7	7 days	14	7 days

Table B 3.6.3-1 (Page 15 of 18)  
Containment Isolation Valve Completion Times

UNID	Penetration	Pressure Boundary Maintained		Pressure Boundary Compromised	
		Category	Completion Time	Category	Completion Time
VLV-67-580D	X-68	7	7 days	14	7 days
VLV-67-585A	X-73	7	7 days	14	7 days
VLV-67-585B	X-70	7	7 days	14	7 days
VLV-67-585C	X-71	7	7 days	14	7 days
VLV-67-585D	X-72	7	7 days	14	7 days
VLV-67-772	X-56	7	7 days	14	7 days
VLV-67-774	X-60	7	7 days	14	7 days
VLV-67-776	X-62	7	7 days	14	7 days
VLV-67-778	X-58	7	7 days	14	7 days
FCV-68-305	X-39B	7	7 days	14	7 days
FCV-68-307	X-84A	7	7 days	14	7 days
FCV-68-308	X-84A	7	7 days	14	7 days
RVLIS - Sys 68	X-85C	7	7 days	14	7 days
RVLIS - Sys 68	X-86A	7	7 days	14	7 days
RVLIS - Sys 68	X-86B	7	7 days	14	7 days
RVLIS - Sys 68	X-86C	7	7 days	14	7 days
RVLIS - Sys 68	X-25C	7	7 days	14	7 days
RVLIS - Sys 68	X-27D	7	7 days	14	7 days
RVLIS - Sys 68	X-26C	7	7 days	14	7 days
VLV-68-559	X-24	7	7 days	14	7 days
VLV-68-560	X-24	7	7 days	14	7 days
VLV-68-561	X-24	7	7 days	14	7 days
FCV-70-85	X-35	7	7 days	14	7 days
FCV-70-87	X-50A	7	7 days	14	7 days
FCV-70-89	X-29	7	7 days	14	7 days
FCV-70-90	X-50A	7	7 days	14	7 days
FCV-70-92	X-29	7	7 days	14	7 days
FCV-70-134	X-50B	7	7 days	14	7 days
FCV-70-140	X-52	7	7 days	14	7 days
FCV-70-141	X-52	7	7 days	14	7 days
FCV-70-143	X-53	7	7 days	14	7 days
VLV-70-678B	X-50B	7	7 days	14	7 days

Table B 3.6.3-1 (Page 16 of 18)  
Containment Isolation Valve Completion Times

UNID	Penetration	Pressure Boundary Maintained		Pressure Boundary Compromised	
		Category	Completion Time	Category	Completion Time
VLV-70-679	X-50B	7	7 days	14	7 days
VLV-70-687	X-50A	7	7 days	14	7 days
VLV-70-691B	X-52	7	7 days	14	7 days
VLV-70-698	X-29	7	7 days	14	7 days
VLV-70-702B	X-53	7	7 days	14	7 days
VLV-70-702C	X-35	7	7 days	14	7 days
VLV-70-702E	X-53	7	7 days	14	7 days
VLV-70-702F	X-35	7	7 days	14	7 days
VLV-70-703	X-35, X-53	7	7 days	14	7 days
VLV-70-735	X-29	7	7 days	14	7 days
VLV-70-737	X-50A	7	7 days	14	7 days
VLV-70-759	X-35	7	7 days	14	7 days
VLV-70-760	X-53	7	7 days	14	7 days
VLV-70-762	X-35	7	7 days	14	7 days
VLV-70-764	X-35	7	7 days	14	7 days
VLV-70-765	X-53	7	7 days	14	7 days
VLV-70-791	X-52	7	7 days	14	7 days
DRIV-72-215F	X-49A	7	7 days	14	7 days
DRIV-72-216F	X-49A	7	7 days	14	7 days
DRIV-72-217F	X-49B	7	7 days	14	7 days
DRIV-72-218F	X-49B	7	7 days	14	7 days
FCV-72-2	X-48B	7	7 days	14	7 days
FCV-72-39	X-48A	7	7 days	14	7 days
FCV-72-40	X-49A	7	7 days	14	7 days
FCV-72-41	X-49B	7	7 days	14	7 days
RFV-72-40	X-49A	7	7 days	14	7 days
RFV-72-41	X-49B	7	7 days	14	7 days
TTIV-72-215E	X-49A	7	7 days	14	7 days
TTIV-72-216E	X-49A	7	7 days	14	7 days
TTIV-72-217E	X-49B	7	7 days	14	7 days
TTIV-72-218E	X-49B	7	7 days	14	7 days
VLV-72-512	X-24	7	7 days	14	7 days

Table B 3.6.3-1 (Page 17 of 18)  
Containment Isolation Valve Completion Times

UNID	Penetration	Pressure Boundary Maintained		Pressure Boundary Compromised	
		Category	Completion Time	Category	Completion Time
VLV-72-513	X-24	7	7 days	14	7 days
VLV-72-517	X-24	7	7 days	14	7 days
VLV-72-518	X-24	7	7 days	14	7 days
VLV-72-543	X-48A	7	7 days	14	7 days
VLV-72-544	X-48B	7	7 days	14	7 days
VLV-72-545	X-48A	7	7 days	14	7 days
VLV-72-546	X-48B	7	7 days	14	7 days
VLV-72-547	X-48A	7	7 days	14	7 days
VLV-72-548	X-48B	7	7 days	14	7 days
VLV-72-551	X-49B	7	7 days	14	7 days
VLV-72-552	X-49A	7	7 days	14	7 days
VLV-72-555	X-49B	7	7 days	14	7 days
VLV-72-556	X-49A	7	7 days	14	7 days
FCV-74-1	X-107	1	4 hours	8	4 hours
FCV-74-2	X-107	1	4 hours	8	4 hours
VLV-74-503	X-107	7	7 days	14	7 days
VLV-74-504	X-107	7	7 days	14	7 days
VLV-74-505	X-107	1	4 hours	8	4 hours
VLV-74-549	X-107	7	7 days	14	7 days
FCV-77-9	X-46	7	7 days	14	7 days
FCV-77-10	X-46	7	7 days	14	7 days
FCV-77-18	X-45	7	7 days	14	7 days
FCV-77-19	X-45	7	7 days	14	7 days
FCV-77-20	X-45	7	7 days	14	7 days
FCV-77-127	X-41	7	7 days	14	7 days
FCV-77-128	X-41	7	7 days	14	7 days
VLV-77-848	X-39B	7	7 days	14	7 days
VLV-77-849	X-39B	7	7 days	14	7 days
VLV-77-867	X-39A	7	7 days	14	7 days
VLV-77-868	X-39A	7	7 days	14	7 days
VLV-77-984	X-45	7	7 days	14	7 days
VLV-78-226A	X-83	7	7 days	14	7 days

Table B 3.6.3-1 (Page 18 of 18)  
Containment Isolation Valve Completion Times

UNID	Penetration	Pressure Boundary Maintained		Pressure Boundary Compromised	
		Category	Completion Time	Category	Completion Time
VLV-78-228A	X-82	7	7 days	14	7 days
VLV-78-557	X-83	6	72 hours	13	72 hours
VLV-78-558	X-83	6	72 hours	13	72 hours
VLV-78-560	X-82	6	72 hours	13	72 hours
VLV-78-561	X-82	6	72 hours	13	72 hours
FCV-81-12	X-42	7	7 days	14	7 days
VLV-81-502	X-42	7	7 days	14	7 days
VLV-81-529	X-42	7	7 days	14	7 days
VLV-84-511	X-46	7	7 days	14	7 days
BLF - Sys 88	X-54	7	7 days	13	72 hours
BLF - Sys 88	X-88	7	7 days	14	7 days
BLF - Sys 88	X-108	1	4 hours	8	4 hours
BLF - Sys 88	X-109	1	4 hours	8	4 hours
BLF - Sys 88	X-117	6	72 hours	13	72 hours
BLF - Sys 88	X-118	6	72 hours	13	72 hours
FCV-90-107	X-94A	7	7 days	14	7 days
FCV-90-107	X-94B	7	7 days	14	7 days
FCV-90-108	X-94B	7	7 days	14	7 days
FCV-90-109	X-94A	7	7 days	14	7 days
FCV-90-110	X-94C	7	7 days	14	7 days
FCV-90-111	X-94C	7	7 days	14	7 days
FCV-90-113	X-95A	7	7 days	14	7 days
FCV-90-113	X-95B	7	7 days	14	7 days
FCV-90-114	X-95B	7	7 days	14	7 days
FCV-90-115	X-95A	7	7 days	14	7 days
FCV-90-116	X-95C	7	7 days	14	7 days
FCV-90-117	X-95C	7	7 days	14	7 days

## B 3.6 CONTAINMENT SYSTEMS

### B 3.6.4 Containment Pressure

#### BASES

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**BACKGROUND** The containment pressure is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a loss of coolant accident (LOCA) or steam line break (SLB). These limits also prevent the containment pressure from exceeding the containment design negative pressure differential with respect to the annulus pressure 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. Should operation occur outside these limits coincident with a Design Basis Accident (DBA), post accident containment pressures could exceed calculated values.

---

**APPLICABLE SAFETY ANALYSES** Containment internal pressure is an initial condition used in the DBA analyses to establish the maximum peak containment internal pressure. The limiting DBAs considered, relative to containment pressure, are the LOCA and SLB, which are analyzed using computer pressure transients. The worst case LOCA generates larger mass and energy release than the worst case SLB. Thus, the LOCA event bounds the SLB event from the containment peak pressure standpoint (Ref. 1).

The initial pressure condition used in the containment analysis was 0.3 psig. This resulted in a maximum peak compression pressure of 7.18 psig in the upper containment from a LOCA. The containment analysis (Ref. 1) shows that the maximum peak calculated containment pressure,  $P_a$ , results from the limiting LOCA. The maximum containment pressure resulting from the worst case LOCA, 11.44 psig, does not exceed the containment design pressure, 12 psig.

The containment was also designed for an external pressure load equivalent to 0.5 psig. The inadvertent actuation of the Containment Spray System and Air Return System was analyzed to determine the resulting reduction in containment pressure. The initial pressure condition used in this analysis was 0.1 psi less than annulus pressure. This resulted in a minimum pressure inside containment of 0.49 psi less than annulus pressure, which is less than the design load.

BASES

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APPLICABLE SAFETY ANALYSES (continued)

For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the cooling effectiveness of the Emergency Core Cooling System during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. Therefore, for the reflood phase, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the containment pressure response in accordance with 10 CFR 50, Appendix K (Ref. 2).

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

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LCO

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

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APPLICABILITY

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

---

ACTIONS

A.1

When containment pressure is not within the limits of the LCO, it 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.

BASES

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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.

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SURVEILLANCE  
REQUIREMENTS

SR 3.6.4.1

Verifying that containment pressure is within limits ensures that unit operation remains within the limits assumed in the containment analysis.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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REFERENCES

1. UFSAR, Section 6.2.
  2. 10 CFR 50, Appendix K.
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## B 3.6 CONTAINMENT SYSTEMS

### B 3.6.5 Containment Air Temperature

#### BASES

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**BACKGROUND** The containment structure serves to contain radioactive material 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 steam line break (SLB).

The containment average air temperature limit is derived from the input conditions used in the containment functional analyses and the containment structure external pressure analyses. This LCO ensures that initial conditions assumed in the analysis of containment response to a DBA are not violated during 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 upon the energy released to the containment due to the event, as well as the initial containment temperature and pressure. The higher the initial temperature, the more energy that must be removed, resulting in a higher peak containment pressure and temperature. Exceeding containment design pressure may result in leakage greater than that assumed in the accident analysis. Operation with containment temperature in excess of the LCO limit violates an initial condition assumed in the accident analysis.

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**APPLICABLE SAFETY ANALYSES** Containment average air temperature is an initial condition used in the DBA analyses that establishes the containment environmental qualification operating envelope for both pressure and temperature. The limit for containment average air temperature ensures that operation is maintained within the assumptions used in the DBA analyses for containment (Ref. 1).

The limiting DBAs considered relative to containment OPERABILITY are the LOCA and SLB. The DBA LOCA and SLB are analyzed using computer codes designed to predict the resultant containment pressure transients. No two DBAs are assumed to occur simultaneously or consecutively. The postulated DBAs are analyzed with regard to Engineered Safety Feature (ESF) systems, assuming the loss of one ESF bus, which is the worst case single active failure, resulting in one train each of Containment Spray System, Residual Heat Removal System, and Air Return System being rendered inoperable.

The limiting DBA for the maximum peak containment air temperature is an SLB. For the upper compartment, the initial containment average air temperature assumed in the design basis analyses (Ref. 1) is 85°F.

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BASES

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APPLICABLE SAFETY ANALYSES (continued)

For the lower compartment, the initial average containment air temperature assumed in the design basis analyses is 120°F. However, a sensitivity analysis performed on the steam line break containment response analysis (Ref. 2) indicates that an increase of 5°F in the initial lower containment temperature to 125°F would net a 0.1°F increase in the calculated peak temperature in the lower containment. This resulted in a maximum containment air temperature of 325.6°F. The design temperature is 327°F.

The temperature upper limits are used to establish the environmental qualification operating envelope for both containment compartments. The maximum peak containment air temperature for both containment compartments was calculated to not exceed the containment design temperature. The basis of the containment design temperature, however, is to ensure the performance of safety related equipment inside containment (Ref. 3). Therefore, it is concluded that the calculated transient containment air temperatures are acceptable for the DBA SLB.

The temperature upper limits are also used in the depressurization analyses to ensure that the minimum pressure limit is maintained following an inadvertent actuation of the Containment Spray System for both containment compartments.

The containment pressure transient is sensitive to the initial air mass in containment and, therefore, to the initial containment air temperature. The limiting DBA for establishing the maximum peak containment internal pressure is a LOCA. The temperature lower limits, 85°F for the upper compartment and 100°F for the lower compartment, are used in this analysis to ensure that, in the event of an accident, the maximum containment internal pressure will not be exceeded in either containment compartment.

Containment average air temperature satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

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LCO

During a DBA, with an initial containment average air temperature within the LCO temperature limits, the resultant accident temperature profile assures that the containment structural temperature is maintained below its design temperature and that required safety related equipment will continue to perform its function. In MODES 2, 3 and 4, containment air temperature may be as low as 60°F because the resultant calculated peak containment accident pressure would not exceed the design pressure due to a lesser amount of energy released from the pipe break in these MODES.

BASES

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

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ACTIONS

A.1

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

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SURVEILLANCE  
REQUIREMENTS

SR 3.6.5.1 and SR 3.6.5.2

Verifying that containment average air temperature is within the LCO limits ensures that containment operation remains within the limits assumed for the containment analyses. In order to determine the containment average air temperature, the primary containment upper and lower compartment average air temperatures are the weighted average of the ambient air temperature monitoring stations located in the upper and lower compartment, respectively. The weighted average is the sum of each temperature multiplied by its respective containment volume fraction. In the event of inoperable temperature sensor(s), the weighted average shall be taken as the reduced total divided by one minus the volume fraction represented by the sensor(s) out of service. As a minimum, the temperature readings for the upper compartment average air temperature shall be obtained from Elevation 743 feet (ft), Elevation 786 ft, and either Elevation 786 or 845 ft. Additionally, as a minimum, the temperature readings for the lower compartment average air temperature shall be obtained from Elevation 722 ft, Elevation 700 ft, and either Elevation 685 or 703 ft.

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BASES

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SURVEILLANCE REQUIREMENTS (continued)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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REFERENCES

1. UFSAR, Section 6.2.
  2. LTR-CRA-11-201, Westinghouse Memo – Sequoyah Units 1 and 2 Steamline Break Containment Response Sensitivity Analysis Addressing an Increase in the Initial Temperature in the Lower Containment, August 5, 2011.
  3. 10 CFR 50.49.
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## B 3.6 CONTAINMENT SYSTEMS

### B 3.6.6 Containment Spray System

#### BASES

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**BACKGROUND** The Containment Spray System provides 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 System is designed to meet the requirements of 10 CFR 50, Appendix A, GDC 38, "Containment Heat Removal," GDC 39, "Inspection of Containment Heat Removal Systems," and GDC 40, "Testing of Containment Heat Removal Systems" (Ref. 1).

The Containment Spray System consists of two separate subsystems of equal capacity, each capable of meeting the system design basis spray coverage. A containment spray subsystem contains one containment spray train and one RHR spray train. Each containment spray train includes a containment spray pump, one containment spray heat exchanger, spray headers, nozzles, valves, and piping. Each containment spray train is powered from a separate Engineered Safety Feature (ESF) bus. The refueling water storage tank (RWST) supplies borated water to the Containment Spray System during the injection phase of operation. In the recirculation mode of operation, containment spray pump suction is transferred from the RWST to the containment recirculation sump(s).

The diversion of a portion of the recirculation flow from each train of the Residual Heat Removal (RHR) System to additional redundant spray headers completes the Containment Spray System heat removal capability. Each RHR spray train is capable of supplying spray coverage, if required, to supplement the Containment Spray System.

The containment spray train and RHR spray train provide a spray of cold or subcooled borated water into the upper region of containment to limit the containment pressure and temperature during a DBA. The RWST solution temperature is an important factor in determining the heat removal capability of the Containment Spray System during the injection phase. In the recirculation mode of operation, heat is removed from the containment sump water by the Containment Spray System heat exchangers. Each containment spray subsystem provides adequate spray coverage to meet the system design requirements for containment heat removal.

## BASES

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### BACKGROUND (continued)

The containment spray trains are actuated either automatically by a High – High containment pressure signal or manually. An automatic actuation opens the containment spray pump discharge valves, starts the two containment spray pumps, and begins the injection phase. A manual actuation of the containment spray trains requires the operator to actuate two separate switches on the main control board to begin the same sequence. The Low-Low alarm for the RWST or a high level in the containment sump signals the operator to manually align the system to the recirculation mode. Operation of the Containment Spray System in the recirculation mode is controlled by the operator in accordance with the emergency operation procedures.

The RHR spray trains are initiated manually, when required by the emergency operating procedures, after the Emergency Core Cooling System (ECCS) is operating in the recirculation mode. The RHR spray trains are available to supplement the containment spray trains, if required, in limiting containment pressure. This additional spray capacity would typically be used after the ice bed has been depleted and in the event that containment pressure rises above a predetermined limit. The Containment Spray System is an ESF system. It is designed to ensure that the heat removal capability required during the post accident period can be attained.

The operation of the Containment Spray System, together with the ice condenser, is adequate to assure pressure suppression during the initial blowdown of steam and water from a DBA. During the post blowdown period, the Air Return System (ARS) is automatically started. The ARS returns upper compartment air through the divider barrier to the lower compartment. This serves to continue circulating heated air and steam through the ice condenser, where heat is removed by the remaining ice.

The Containment Spray System limits the temperature and pressure that could be expected following a DBA. Protection of containment integrity limits leakage of fission product radioactivity from containment to the environment.

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### APPLICABLE SAFETY ANALYSES

The limiting DBAs considered relative to containment OPERABILITY are the loss of coolant accident (LOCA) and the steam line break (SLB). The DBA LOCA and SLB are analyzed using computer codes designed to predict the resultant containment pressure and temperature transients. No two DBAs are assumed to occur simultaneously or consecutively. The postulated DBAs are analyzed, in regard to containment ESF systems, assuming the loss of one ESF bus, which is the worst case single active failure, resulting in one train of containment spray, RHR, and an ARS fan being rendered inoperable (Ref. 2).

## BASES

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### APPLICABLE SAFETY ANALYSES (continued)

The DBA analyses show that the maximum peak containment pressure of 11.44 psig results from the LOCA analysis, and is calculated to be less than the containment design pressure. The basis of the containment design temperature (327°F) is to ensure the OPERABILITY of safety related equipment inside containment (Ref. 3). The maximum peak containment atmosphere temperature of 325.6°F results from the SLB analysis. Therefore, the calculated peak containment atmosphere temperature is acceptable for the DBA SLB.

The modeled containment spray trains actuation from the containment analysis is based on a response time associated with exceeding the High—High containment pressure signal setpoint to achieving full flow through the containment spray train nozzles. A delayed response time initiation provides conservative analyses of peak calculated containment temperature and pressure responses. The containment spray trains total response time of 250 seconds is composed of signal delay, diesel generator startup, and system startup time to full flow through the containment spray train nozzles.

For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the ECCS cooling effectiveness during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. For these calculations, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the calculated transient containment pressures in accordance with 10 CFR 50, Appendix K (Ref. 4).

Inadvertent actuation of the Containment Spray System is evaluated in the analysis, and the resultant reduction in containment pressure is calculated. The maximum calculated reduction in containment pressure resulted in a containment external pressure load of 0.49 psid, which is below the containment design external pressure load.

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

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### LCO

During a DBA, one subsystem of Containment Spray System is required to provide the heat removal capability assumed in the safety analyses. To ensure that these requirements are met, two containment spray subsystems must be OPERABLE with power from two safety related, independent power supplies.

A containment spray subsystem shall be compromised of one containment spray train and one RHR spray train.

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BASES

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LCO (continued)

Each containment spray train includes a containment spray pump, header, valves, heat exchanger, nozzles, piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the RWST upon an ESF actuation signal and manually transferring suction to the containment sump.

Each RHR spray train includes an RHR pump, header, valves, heat exchanger, nozzles, piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the containment sump.

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APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment and an increase in containment pressure and temperature requiring the operation of the Containment Spray System. Furthermore, as stated in the Applicability Note, the RHR spray trains are not required to be OPERABLE in MODE 4.

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

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ACTIONS

A.1

With one containment spray subsystem inoperable, the affected subsystem must be restored to OPERABLE status within 72 hours. The components in this degraded condition are capable of providing 100% of the heat removal needs after an accident. The 72 hour Completion Time was developed taking into account the redundant heat removal afforded by the OPERABLE train and the low probability of a DBA occurring during this period.

B.1 and B.2

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

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BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.6.6.1

Verifying the correct alignment of manual, power operated, and automatic valves, excluding check valves, in the containment spray train provides assurance that the proper flow path exists for containment spray train operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position since they were verified in the correct position prior to being secured. This SR does not require any testing or valve manipulation. Rather, it involves verification that those valves outside containment and capable of potentially being mispositioned, are in the correct position.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.6.6.2

Verifying that each containment spray train pump's developed head at the flow test point is greater than or equal to the required developed head ensures that spray pump performance has not degraded during the cycle. Flow and differential head are normal tests of centrifugal pump performance required by the ASME Code (Ref. 5). Since the containment spray train pumps cannot be tested with flow through the spray headers, they are tested on bypass flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice inspections confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of this SR is in accordance with the Inservice Testing Program.

SR 3.6.6.3 and SR 3.6.6.4

These SRs require verification that each automatic containment spray train valve actuates to its correct position and each containment spray train pump starts upon receipt of an actual or simulated containment spray actuation signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

The surveillance of containment sump isolation valves is also required by SR 3.6.6.3. A single surveillance may be used to satisfy both requirements.

BASES

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.6.5

With the containment spray train inlet valves closed and the containment spray header drained of any solution, low pressure air or smoke can be blown through test connections. This SR ensures that each spray nozzle is unobstructed and that spray coverage of the containment during an accident is not degraded.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.6.6.6

Verifying the correct alignment of manual, power operated, and automatic valves, excluding check valves, in the RHR spray train provides assurance that the proper flow path exists for system operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position since they were verified in the correct position prior to locking, sealing, or securing. A valve is also allowed to be in the nonaccident position provided it can be aligned to the accident position within the time assumed in the accident analysis. This is acceptable since the RHR spray mode is manually initiated. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned, are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.6.6.7

Verifying that each RHR spray train pump's developed head at the flow test point is greater than or equal to the required developed head ensures that spray pump performance has not degraded during the cycle. Flow and differential head are normal tests of centrifugal pump performance required by the ASME Code (Ref. 5). Since the RHR spray train pumps cannot be tested with flow through the spray headers, they are tested on bypass flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice inspections confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of this SR is in accordance with the Inservice Testing Program.

BASES

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.6.8

With the RHR spray train inlet valves closed and the RHR spray header drained of any solution, low pressure air or smoke can be blown through test connections. This SR ensures that each RHR spray nozzle is unobstructed and that spray coverage of the containment during an accident is not degraded.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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REFERENCES

1. 10 CFR 50, Appendix A, GDC 38, GDC 39, and GDC 40.
  2. UFSAR, Section 6.2.
  3. 10 CFR 50.49.
  4. 10 CFR 50, Appendix K.
  5. ASME Code for Operation and Maintenance of Nuclear Power Plants.
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## B 3.6 CONTAINMENT SYSTEMS

### B 3.6.7 Shield Building

#### BASES

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##### BACKGROUND

The shield building is a concrete structure that surrounds the steel containment vessel. Between the containment vessel and the shield building inner wall is an annular space that collects containment leakage that may occur following a loss of coolant accident (LOCA). This space also allows for periodic inspection of the outer surface of the steel containment vessel.

The Emergency Gas Treatment System (EGTS) is a system consisting of two subsystems:

- a. annulus vacuum control subsystem, and
- b. air cleanup subsystem.

The annulus vacuum control subsystem is used during normal operation to establish and maintain a negative pressure in the annulus space. The annulus vacuum control subsystem does not perform any safety function.

The air cleanup subsystem operates during a LOCA to establish and maintain a negative annulus pressure of at least 0.5 inches water gauge. Filters in the subsystem then control the release of radioactive contaminants to the environment. The EGTS air cleanup subsystem OPERABILITY requirements are specified in LCO 3.6.10, "Emergency Gas Treatment System (EGTS) Air Cleanup Subsystem." The shield building is required to be OPERABLE to ensure retention of containment leakage and proper operation of the EGTS Air Cleanup Subsystem.

The isolation devices for the penetrations in the shield building boundary are a part of the shield building leak tight barrier. To maintain the shield building boundary leak tight, the sealing mechanism associated with each penetration (e.g., welds, bellows, or O-rings) are required to be OPERABLE. Access to the annulus area of the shield building is provided via the reactor building access room door and the water tight annulus access door located on 690 ft. elevation. During normal operation, these doors provide personnel and equipment access to the shield building annulus area and are equipped with electrical interlocks to assure that one door is always closed.

BASES

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APPLICABLE SAFETY ANALYSES      The design basis for shield building OPERABILITY is a LOCA. Maintaining shield building OPERABILITY ensures that the release of radioactive material from the containment atmosphere is restricted to those leakage paths and associated leakage rates assumed in the accident analyses.

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

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LCO      Shield building OPERABILITY must be maintained to ensure proper operation of the EGTS Air Cleanup Subsystem and to limit radioactive leakage from the containment to those paths and leakage rates assumed in the accident analyses.

The LCO is modified by a Note to allow the annulus access door to be opened to allow normal transit entry and exit. The basis of this exception is the assumption that, for normal transit, the time which the door is open will be short (i.e., shorter than the Completion Time for Condition A).

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APPLICABILITY      Maintaining shield building OPERABILITY prevents leakage of radioactive material from the shield building. Radioactive material may enter the shield building from the containment following a LOCA. Therefore, shield building OPERABILITY is required in MODES 1, 2, 3, and 4 when a steam line break, LOCA, or rod ejection accident could release radioactive material to the 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.

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ACTIONS      A.1

In the event shield building OPERABILITY is not maintained, shield building OPERABILITY must be restored within 1 hour. One hour is a reasonable Completion Time considering the limited leakage design of containment and the low probability of a Design Basis Accident occurring during this time period. This specified time period is also consistent with the ACTIONS of LCO 3.6.1, "Containment," which requires the containment be restored to OPERABLE status within 1 hour.

BASES

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ACTIONS (continued)

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.

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SURVEILLANCE  
REQUIREMENTS

SR 3.6.7.1

Maintaining shield building OPERABILITY requires verifying the door in the access opening closed. The access opening contains one door. The annulus access door is normally kept closed, except when the access opening is being used for entry and exit.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.6.7.2

A visual inspection of the accessible shield building interior and exterior surfaces and verification that no apparent changes in the concrete surface appearance or other abnormal degradation will 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.

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REFERENCES

None.

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## B 3.6 CONTAINMENT SYSTEMS

### B 3.6.8 Hydrogen Mitigation System (HMS)

#### BASES

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**BACKGROUND** The HMS reduces the potential for breach of primary containment due to a hydrogen oxygen reaction in post accident environments. The HMS is required by 10 CFR 50.44, "Standards for Combustible Gas Control Systems in Light-Water-Cooled Reactors" (Ref. 1), to reduce the hydrogen concentration in the primary containment following a degraded core accident. The HMS must be capable of handling an amount of hydrogen equivalent to that generated from a metal water reaction involving 75% of the fuel cladding surrounding the active fuel region (excluding the plenum volume).

10 CFR 50.44 (Ref. 1) requires units with ice condenser containments to install suitable hydrogen control systems that would accommodate an amount of hydrogen equivalent to that generated from the reaction of 75% of the fuel cladding with water. The HMS provides this required capability. This requirement was placed on ice condenser units because of their small containment volume and low design pressure (compared with pressurized water reactor dry containments). Calculations indicate that if hydrogen equivalent to that generated from the reaction of 75% of the fuel cladding with water were to collect in the primary containment, the resulting hydrogen concentration would be far above the lower flammability limit such that, if ignited from a random ignition source, the resulting hydrogen burn would seriously challenge the containment and safety systems in the containment.

The HMS is based on the concept of controlled ignition using thermal ignitors, designed to be capable of functioning in a post accident environment, seismically supported, and capable of actuation from the control room. A total of 68 ignitors are distributed throughout the various regions of containment in which hydrogen could be released or to which it could flow in significant quantities. The ignitors are arranged in two independent trains such that each containment region has at least two ignitors, one from each train, controlled and powered redundantly so that ignition would occur in each region even if one train failed to energize. Additional information regarding containment regions and the distribution of hydrogen ignitors within each region is contained in UFSAR Section 6.2.5A.

When the HMS is initiated, the ignitor elements are energized and heat up to a surface temperature  $\geq 1700^{\circ}\text{F}$ . At this temperature, they ignite the hydrogen gas that is present in the airspace in the vicinity of the ignitor. The HMS depends on the dispersed location of the ignitors so that local pockets of hydrogen at increased concentrations would burn before

BASES

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BACKGROUND (continued)

reaching a hydrogen concentration significantly higher than the lower flammability limit. Hydrogen ignition in the vicinity of the ignitors is assumed to occur when the local hydrogen concentration reaches 8.0 volume percent and results in 85% of the hydrogen present being consumed.

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APPLICABLE  
SAFETY  
ANALYSES

The HMS causes hydrogen in containment to burn in a controlled manner as it accumulates following a degraded core accident (Ref. 2). Burning occurs at the lower flammability concentration, where the resulting temperatures and pressures are relatively benign. Without the system, hydrogen could build up to higher concentrations that could result in a violent reaction if ignited by a random ignition source after such a buildup.

The hydrogen ignitors are not included for mitigation of a Design Basis Accident (DBA) because an amount of hydrogen equivalent to that generated from the reaction of 75% of the fuel cladding with water is far in excess of the hydrogen calculated for the limiting DBA loss of coolant accident (LOCA). The hydrogen ignitors have been shown by probabilistic risk analysis to be a significant contributor to limiting the severity of accident sequences that are commonly found to dominate risk for units with ice condenser containments. The Hydrogen Mitigation System satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

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LCO

Two HMS trains must be OPERABLE with power from two independent, safety related power supplies.

For this unit, an OPERABLE HMS train consists of 33 of 34 ignitors energized on the train.

Operation with at least one HMS train ensures that the hydrogen in containment can be burned in a controlled manner. Unavailability of both HMS trains could lead to hydrogen buildup to higher concentrations, which could result in a violent reaction if ignited. The reaction could take place fast enough to lead to high temperatures and overpressurization of containment and, as a result, breach containment or cause containment leakage rates above those assumed in the safety analyses. Damage to safety related equipment located in containment could also occur.

Each containment region must contain at least one OPERABLE hydrogen ignitor. This ensures that, assuming a single failure, there is still ignition capability in an adjacent region.

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APPLICABILITY

Requiring OPERABILITY in MODES 1 and 2 for the HMS ensures its immediate availability after safety injection and scram actuated on a LOCA initiation. In the post accident environment, the two HMS trains are required to control the hydrogen concentration within containment to near

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## BASES

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### APPLICABILITY (continued)

its flammability limit of 4.1 volume percent assuming a worst case single failure. This prevents overpressurization of containment and damage to safety related equipment and instruments located within containment.

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

In MODES 5 and 6, the probability and consequences of a LOCA are reduced due to the pressure and temperature limitations of these MODES. Therefore, the HMS is not required to be OPERABLE in MODES 5 and 6.

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### ACTIONS

#### A.1 and A.2

With one HMS train inoperable, the inoperable train must be restored to OPERABLE status within 7 days or the OPERABLE train must be verified OPERABLE frequently by performance of SR 3.6.8.1. The 7 day Completion Time is based on the low probability of the occurrence of a degraded core event that would generate hydrogen in amounts equivalent to a metal water reaction of 75% of the core cladding, the length of time after the event that operator action would be required to prevent hydrogen accumulation from exceeding this limit, and the low probability of failure of the OPERABLE HMS train. Alternative Required Action A.2, by frequent surveillances, provides assurance that the OPERABLE train continues to be OPERABLE.

#### B.1

Condition B is one containment region with no OPERABLE hydrogen ignitor. Thus, while in Condition B, or in Conditions A and B simultaneously, there would always be ignition capability in the adjacent containment regions that would provide redundant capability by flame propagation to the region with no OPERABLE ignitors.

Required Action B.1 calls for the restoration of one hydrogen ignitor in each region to OPERABLE status within 7 days. The 7 day Completion Time is based on the same reasons given under Required Action A.1.

BASES

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## ACTIONS (continued)

C.1

The unit must be placed in a MODE in which the LCO does not apply if the HMS subsystem(s) cannot be restored to OPERABLE status within the associated Completion Time. This is done by placing the unit in 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.

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SURVEILLANCE  
REQUIREMENTSSR 3.6.8.1

This SR confirms that  $\geq 33$  of 34 hydrogen ignitors can be successfully energized in each train. The ignitors are simple resistance elements. Therefore, energizing provides assurance of OPERABILITY. The allowance of one inoperable hydrogen ignitor is acceptable because, although one inoperable hydrogen ignitor in a region would compromise redundancy in that region, the containment regions are interconnected so that ignition in one region would cause burning to progress to the others (i.e., there is overlap in each hydrogen ignitor's effectiveness between regions).

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.6.8.2

This SR confirms that the two inoperable hydrogen ignitors allowed by SR 3.6.8.1 (i.e., one in each train) are not in the same containment region which ensures that each containment region contains at least one OPERABLE hydrogen ignitor.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.6.8.3

A more detailed functional test is performed to verify system OPERABILITY. All ignitors (glow plugs), including normally inaccessible ignitors, are visually checked for a glow to verify that they are energized. Additionally, the surface temperature of each glow plug is measured to be  $\geq 1700^{\circ}\text{F}$  to demonstrate that a temperature sufficient for ignition is achieved.

BASES

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SURVEILLANCE REQUIREMENTS (continued)

The Surveillance Frequency is controlled under the Surveillance  
Frequency Control Program

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REFERENCES

1. 10 CFR 50.44.
  2. UFSAR, Section 6.2.
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## B 3.6 CONTAINMENT SYSTEMS

### B 3.6.9 Vacuum Relief Valves

#### BASES

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**BACKGROUND** The purpose of the vacuum relief lines is to 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 containment cooling features, such as the Containment Spray System, the Air Return System, or both. Multiple equipment failures or human errors are necessary to cause inadvertent actuation of these systems.

The containment pressure vessel contains three vacuum relief lines that protect the containment from excessive external loading.

The vacuum relief system has three identical lines located on the dome, at the same elevation, and 120° apart. Each line contains a containment vessel vacuum relief valve in series with a containment vessel vacuum relief isolation valve, the vacuum relief valve being outside of the isolation valve. The lines are installed such that there is sufficient space between the vacuum relief system and the Shield Building to prevent contact during seismic or pressure transient motion and to allow for an adequate airflow path.

Each containment vessel vacuum relief valve is a 24 inch, self-actuated, horizontally installed, swing-disc valve, with an elastomer seat. The seat material will withstand post-LOCA temperature, pressure, and radiation conditions. Each line has a design airflow rate of 28 pounds per second at a pressure differential of 0.5 psid across the entire line. Each normally closed vacuum relief valve is equipped with limit switches so that open and closed positions of the valve are indicated in the main control room. The opening of any of these valves is indicated in the main control room. The valves begin opening at a containment external pressure differential of 0.1 psid and will be fully open in 2.2 seconds for a vacuum relief system design basis event.

Each containment vessel vacuum relief isolation valve is a pneumatically operated butterfly valve with an elastomer seat. The valve, including seat material, will withstand post-LOCA temperature, pressure, and radiation conditions. Two separate trains of control air supplies are available to the two independent solenoid valves which power the isolation valve. The isolation valve, which is normally open, fails open, and will close when containment high pressure reaches the set pressure of 1.5 psid. The high pressure signal is developed from either of two independent sets of three pressure sensors and is completely independent of other containment isolation signals for other systems. Each isolation valve is equipped with

BASES

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BACKGROUND (continued)

a limit switch so that open and closed positions are indicated in the main control room.

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APPLICABLE  
SAFETY  
ANALYSES

Design of the vacuum relief lines involves calculating the effect of inadvertent actuation of containment cooling features, which can reduce the atmospheric temperature (and hence pressure) inside containment (Ref. 1). Conservative assumptions are used for all the relevant parameters in the calculation; for example, for the Containment Spray System, the minimum spray water temperature, maximum initial containment temperature, maximum spray flow, all spray trains operating, etc. The resulting containment pressure versus time is calculated, including the effect of the opening of the vacuum relief lines when their negative pressure setpoint is reached. It is also assumed that one valve fails to open.

The containment was designed for an external pressure load equivalent to 0.5 psig. The inadvertent actuation of the containment cooling features was analyzed to determine the resulting reduction in containment pressure. The initial pressure condition used in this analysis was 0.1 psi less than annulus pressure. This resulted in a minimum pressure inside containment of 0.49 psi less than annulus pressure, which is less than the design load.

The vacuum relief valves must also perform the containment isolation function in a containment high pressure event. For this reason, the system is designed to take the full containment positive design pressure and the environmental conditions (temperature, pressure, humidity, radiation, chemical attack, etc.) associated with the containment DBA.

The vacuum relief valves satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

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LCO

The LCO establishes the minimum equipment required to accomplish the vacuum relief function following the inadvertent actuation of containment cooling features. Three vacuum relief lines are required to be OPERABLE to ensure that at least two are available, assuming one vacuum relief valve fails to open.

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APPLICABILITY

In MODES 1, 2, 3, and 4, the containment cooling features, such as the Containment Spray System and the Air Return System, are required to be OPERABLE to mitigate the effects of a DBA. Excessive negative pressure inside containment could occur 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, Air Return System, or both.

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BASES

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APPLICABILITY (continued)

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 Air Return System are not required to be OPERABLE in MODES 5 and 6. Therefore, maintaining OPERABLE vacuum relief valves is not required in MODE 5 or 6.

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ACTIONS

A.1

When one of the required vacuum relief lines is 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.

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SURVEILLANCE  
REQUIREMENTS

SR 3.6.9.1

This SR cites 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 the ASME Code (Ref. 2). Therefore, SR Frequency is governed by the Inservice Testing Program.

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REFERENCES

1. UFSAR, Section 6.2.
  2. ASME Code for Operation and Maintenance of Nuclear Power Plants.
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## B 3.6 CONTAINMENT SYSTEMS

### B 3.6.10 Emergency Gas Treatment System (EGTS) Air Cleanup Subsystem

#### BASES

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**BACKGROUND** The EGTS Air Cleanup Subsystem is required by 10 CFR 50, Appendix A, GDC 41, "Containment Atmosphere Cleanup" (Ref. 1), to ensure that radioactive materials that leak from the primary containment into the shield building (secondary containment) following a Design Basis Accident (DBA) are filtered and adsorbed prior to exhausting to the environment.

The containment has a secondary containment called the shield building, which is a concrete structure that surrounds the steel primary containment vessel. Between the containment vessel and the shield building inner wall is an annular space that collects any containment leakage that may occur following a loss of coolant accident (LOCA). This space also allows for periodic inspection of the outer surface of the steel containment vessel.

The EGTS design consists of two subsystems. The annulus vacuum control subsystem is used to establish and maintain a negative pressure within the secondary containment annulus during normal plant operation. The annulus vacuum control subsystem does not perform any safety function. The air cleanup subsystem is actuated following a LOCA to maintain a negative pressure in the annulus area between the shield building and the steel containment. Filters in the air cleanup subsystem then control the release of radioactive contaminants to the environment. The air cleanup subsystem is the portion of EGTS that performs a safety function and is required to be OPERABLE. OPERABILITY requirements associated with the shield building are specified in LCO 3.6.7, "Shield Building." Shield building OPERABILITY is required to ensure retention of primary containment leakage and proper operation of the EGTS Air Cleanup Subsystem.

The EGTS Air Cleanup Subsystem consists of two separate and redundant trains. Each train includes a heater, a prefilter, moisture separators, 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

BASES

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BACKGROUND (continued)

Phase A containment isolation signal. The system is described in Reference 2.

The prefilters remove large particles in the air, and the moisture separators remove entrained water droplets present, to prevent excessive loading of the HEPA filters and charcoal absorbers. Heaters may be included to reduce the relative humidity of the airstream on systems that operate in high humidity.

The EGTS Air Cleanup Subsystem reduces the radioactive content in the shield building atmosphere following a DBA. Loss of the EGTS Air Cleanup Subsystem could cause site boundary doses, in the event of a DBA, to exceed the values given in licensing basis.

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APPLICABLE  
SAFETY  
ANALYSES

The EGTS Air Cleanup Subsystem 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 EGTS Air Cleanup Subsystem 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 EGTS Air Cleanup Subsystem actuation in the safety analyses is based upon a worst case response time following a Phase A containment isolation initiated at the limiting setpoint. The total response time, from exceeding the signal setpoint to attaining the negative pressure of 0.5 inch water gauge in the shield building, is 60 seconds. This response time is composed of signal delay, diesel generator startup and sequencing time, system startup time, and time for the system to attain the required pressure after starting.

The EGTS Air Cleanup Subsystem satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

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LCO

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

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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

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BASES

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APPLICABILITY (continued)

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 (although one or more trains may be operating for other reasons, such as habitability during maintenance in the shield building annulus).

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ACTIONS

A.1

With one EGTS Air Cleanup Subsystem 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 EGTS Air Cleanup Subsystem train and the low probability of a DBA occurring during this period. The Completion Time is adequate to make most repairs.

B.1 and B.2

If the EGTS Air Cleanup Subsystem 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.

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SURVEILLANCE  
REQUIREMENTS

SR 3.6.10.1

Operating each EGTS Air Cleanup Subsystem train from the Control Room with flow through the HEPA filters and charcoal adsorbers 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 Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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BASES

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.10.2

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

SR 3.6.10.3

The automatic startup ensures that each EGTS Air Cleanup Subsystem train responds properly.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.6.10.4

The EGTS Air Cleanup Subsystem filter cooling bypass valves are tested to verify OPERABILITY. The ability to cool the filters and adsorbers in an inactive air cleanup unit is accomplished with two crossover flow ducts that draw a small stream of air from the active air cleanup unit through the inactive air cleanup unit. The valves in the inactive train automatically receive a signal to open. The capability to manually open the suction valve for the inactive train and align to the affected unit is provided in the main control room to complete the flow path through the inactive unit.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.6.10.5

The proper functioning of the fans, dampers, filters, adsorbers, etc., as a system is verified by the ability of each train to produce the required system flow rate.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.10.6

The EGTS Air Cleanup Subsystem produces a negative pressure to prevent leakage from the shield building. This Surveillance verifies that the shield building can be rapidly drawn down to - 0.5 inch water gauge in the annulus. This test is used to ensure shield building boundary integrity. Establishment of this pressure is confirmed by this SR, which demonstrates that the shield building can be drawn down to a negative pressure of  $\geq 0.5$  inches of water gauge in the annulus in  $\leq 60$  seconds using one EGTS Air Cleanup Subsystem train. The time limit ensures that no significant quantity of radioactive material leaks from the shield building prior to developing the negative pressure. Since this Surveillance is a shield building boundary integrity test, it does not need to be performed with each EGTS Air Cleanup Subsystem train; thus, this Surveillance is performed on a STAGGERED TEST BASIS. The primary purpose of this SR is to ensure shield building integrity. The secondary purpose of this SR is to ensure that the EGTS Air Cleanup Subsystem train being tested functions as designed. Upon failure to meet this SR, the leak tightness of the shield building must be immediately assessed to determine the impact on the OPERABILITY of the shield building. If a negative pressure of  $\geq 0.5$  inch water gauge cannot be maintained in the annulus by either EGTS Air Cleanup Subsystem train (i.e., loss of shield building safety function), the shield building must be declared inoperable and ACTIONS of LCO 3.6.7 performed in accordance with LCO 3.0.6 and Specification 5.5.13, "Safety Function Determination Program (SFDP)."

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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REFERENCES

1. 10 CFR 50, Appendix A, GDC 41.
  2. UFSAR, Section 6.2.
  3. UFSAR, Chapter 15.
  4. Regulatory Guide 1.52, Revision 2.
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## B 3.6 CONTAINMENT SYSTEMS

### B 3.6.11 Air Return System (ARS)

#### BASES

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**BACKGROUND** The ARS is designed to assure the rapid return of air from the upper to the lower containment compartment after the initial blowdown following a Design Basis Accident (DBA). The return of this air to the lower compartment and subsequent recirculation back up through the ice condenser assists in cooling the containment atmosphere and limiting post accident pressure and temperature in containment to less than design values. Limiting pressure and temperature reduces the release of fission product radioactivity from containment to the environment in the event of a DBA.

The ARS provides post accident hydrogen mixing in selected areas of containment. Hydrogen collection headers are routed to potential hydrogen pockets in containment, terminating on the suction side of either of the two ARS fans at the header isolation valves. The minimum design flow from each potential hydrogen pocket is sufficient to limit the local concentration of hydrogen.

The ARS consists of two separate trains of equal capacity, each capable of meeting the design bases. Each train includes a 100% capacity air return fan, associated damper, and hydrogen collection headers with isolation valves. Each train is powered from a separate Engineered Safety Features (ESF) bus.

The ARS fans are automatically started by the Phase B containment isolation signal approximately 10 minutes after the containment pressure reaches the pressure setpoint. The fan backdraft dampers ensure that no energy released during the initial phase of a DBA will bypass the ice bed through the ARS fans.

After starting, the fans displace air from the upper compartment to the lower compartment, thereby returning the air that was displaced by the high energy line break blowdown from the lower compartment. After discharge into the lower compartment, air flows with steam produced by residual heat through the ice condenser doors into the ice condenser compartment where the steam portion of the flow is condensed. The air flow returns to the upper compartment through the top deck doors in the upper portion of the ice condenser compartment. The ARS fans operate continuously after actuation, circulating air through the containment volume and purging all potential hydrogen pockets in containment.

The ARS also functions, after all the ice has melted, to circulate any steam still entering the lower compartment to the upper compartment where the Containment Spray System can cool it.

BASES

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BACKGROUND (continued)

The ARS is an ESF system. It is designed to ensure that the heat removal capability required during the post accident period can be attained. The operation of the ARS, in conjunction with the ice bed, the Containment Spray System, and the Residual Heat Removal (RHR) System spray, provides the required heat removal capability to limit post accident conditions to less than the containment design values.

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APPLICABLE  
SAFETY  
ANALYSES

The limiting DBAs considered relative to containment temperature and pressure are the loss of coolant accident (LOCA) and the steam line break (SLB). The LOCA and SLB are analyzed using computer codes designed to predict the resultant containment pressure and temperature transients. DBAs are assumed not to occur simultaneously or consecutively. The postulated DBAs are analyzed, in regard to ESF systems, assuming the loss of one ESF bus, which is the worst case single active failure and results in one train each of the Containment Spray System, RHR System, and ARS being inoperable (Ref. 1). The DBA analyses show that the maximum peak containment pressure results from the LOCA analysis and is calculated to be less than the containment design pressure.

For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the cooling effectiveness of the Emergency Core Cooling System during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. For these calculations, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the calculated transient containment pressures, in accordance with 10 CFR 50, Appendix K (Ref. 2).

The analysis for minimum internal containment pressure (i.e., maximum external differential containment pressure) assumes inadvertent simultaneous actuation of both the ARS and the Containment Spray System. The containment vacuum relief valves are designed to accommodate inadvertent actuation of either or both systems.

The modeled ARS actuation from the containment analysis is based upon a response time associated with exceeding the containment pressure High-High signal setpoint to achieving full ARS air flow. A delayed response time initiation ensures that no energy released during the initial phase of a DBA will bypass the ice bed through the ARS fans. The ARS total response time of 600 seconds consists of the built in signal delay.

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

BASES

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LCO In the event of a DBA, one train of the ARS is required to provide the minimum air recirculation for heat removal and hydrogen mixing assumed in the safety analyses. To ensure this requirement is met, two trains of the ARS must be OPERABLE. This will ensure that at least one train will operate, assuming the worst case single failure occurs, which is in the loss of ESF power supply.

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APPLICABILITY In MODES 1, 2, 3, and 4, a DBA could cause an increase in containment pressure and temperature requiring the operation of the ARS. Therefore, the LCO is applicable in MODES 1, 2, 3, and 4.

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

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ACTIONS A.1

If one of the required trains of the ARS is inoperable, it must be restored to OPERABLE status within 72 hours. The components in this degraded condition are capable of providing 100% of the flow needs after an accident. The 72 hour Completion Time was developed taking into account the redundant flow capability of the OPERABLE ARS train and the low probability of a DBA occurring in this period.

B.1 and B.2

If the ARS 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.

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SURVEILLANCE REQUIREMENTS SR 3.6.11.1

Verifying that each ARS fan starts on an actual or simulated actuation signal, after a delay  $\geq 9.0$  minutes and  $\leq 11.0$  minutes, and operates for  $\geq 15$  minutes is sufficient to ensure that the fans are OPERABLE and that the associated controls and time delays are functioning properly. It also ensures that blockage, fan and/or motor failure, or excessive vibration can be detected for corrective action.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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BASES

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.11.2

Verifying ARS fan motor current with the return air dampers closed confirms one operating condition of the fan. This test is indicative of overall fan motor performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.6.11.3

Verifying the OPERABILITY of the return air damper provides assurance that the proper flow path will exist when the fan is started. By applying the correct counterweight, the damper operation can be confirmed.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program

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REFERENCES

1. UFSAR, Section 6.2.
  2. 10 CFR 50, Appendix K.
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## B 3.6 CONTAINMENT SYSTEMS

### B 3.6.12 Ice Bed

#### BASES

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**BACKGROUND** The ice bed consists of a minimum of 1,916,000 lb of ice stored within the ice condenser. The primary purpose of the ice bed is to provide a large heat sink in the event of a release of energy from a Design Basis Accident (DBA) in containment. The ice would absorb energy and limit containment peak pressure and temperature during the accident transient. Limiting the pressure and temperature reduces the release of fission product radioactivity from containment to the environment in the event of a DBA.

The ice condenser is an annular compartment enclosing approximately 300 degrees of the perimeter of the upper containment compartment, but penetrating the operating deck so that a portion extends into the lower containment compartment. The lower portion has a series of hinged doors exposed to the atmosphere of the lower containment compartment, which, for normal unit operation, are designed to remain closed. At the top of the ice condenser is another set of doors exposed to the atmosphere of the upper compartment, which also remain closed during normal unit operation. Intermediate deck doors, located below the top deck doors, form the floor of a plenum at the upper part of the ice condenser. These doors also remain closed during normal unit operation. The upper plenum area is used to facilitate surveillance and maintenance of the ice bed.

1944 ice baskets contain the ice within the ice condenser. The ice bed is considered to consist of the total volume from the bottom elevation of the ice baskets to the top elevation of the ice baskets. The ice baskets position the ice within the ice bed in an arrangement to promote heat transfer from steam to ice. This arrangement enhances the ice condenser's primary function of condensing steam and absorbing heat energy released to the containment during a DBA.

In the event of a DBA, the ice condenser inlet doors (located below the operating deck) open due to the pressure rise in the lower compartment. This allows air and steam to flow from the lower compartment into the ice condenser. The resulting pressure increase within the ice condenser causes the intermediate deck doors and the top deck doors to open, which allows the air to flow out of the ice condenser into the upper compartment. Steam condensation within the ice condenser limits the pressure and temperature buildup in containment. A divider barrier (i.e., operating deck and extensions thereof) separates the upper and lower compartments and ensures that the steam is directed into the ice condenser.

## BASES

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### BACKGROUND (continued)

The ice, together with the containment spray, is adequate to absorb the initial blowdown of steam and water from a DBA and the additional heat loads that would enter containment during several hours following the initial blowdown. The additional heat loads would come from the residual heat in the reactor core, the hot piping and components, and the secondary system, including the steam generators. During the post blowdown period, the Air Return System (ARS) returns upper compartment air through the divider barrier to the lower compartment. This serves to continue circulating heated air and steam from the lower compartment through the ice condenser where the heat is removed by the remaining ice.

As ice melts, the water passes through the ice condenser floor drains into the lower compartment. Thus, a second function of the ice bed is to be a large source of borated water (via the containment sump) for long term Emergency Core Cooling System (ECCS) and Containment Spray System heat removal functions in the recirculation mode.

A third function of the ice bed and melted ice is to remove fission product iodine that may be released from the core during a DBA. Iodine removal occurs during the ice melt phase of the accident and continues as the melted ice is sprayed into the containment atmosphere by the Containment Spray System. The ice is adjusted to an alkaline pH that facilitates removal of radioactive iodine from the containment atmosphere.

It is important for ice to exist in the ice baskets, the ice to be appropriately distributed around the 24 ice condenser bays, and for open flow paths to exist around ice baskets. This is especially important during the initial blowdown so that the steam and water mixture entering the lower compartment do not pass through only part of the ice condenser, depleting the ice there while bypassing the ice in other bays.

Two phenomena that can degrade the ice bed during the long service period are:

- a. Loss of ice by melting or sublimation; and
- b. Obstruction of flow passages through the ice bed due to buildup of ice.

## BASES

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### BACKGROUND (continued)

Both of these degrading phenomena are reduced by minimizing air leakage into and out of the ice condenser.

The ice bed limits the temperature and pressure that could be expected following a DBA, thus limiting leakage of fission product radioactivity from containment to the environment.

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### APPLICABLE SAFETY ANALYSES

The limiting DBAs considered relative to containment temperature and pressure are the loss of coolant accident (LOCA) and the steam line break (SLB). The LOCA and SLB are analyzed using computer codes designed to predict the resultant containment pressure and temperature transients. DBAs are not assumed to occur simultaneously or consecutively.

Although the ice condenser is a passive system that requires no electrical power to perform its function, the Containment Spray System and the ARS also function to assist the ice bed in limiting pressures and temperatures. Therefore, the postulated DBAs are analyzed in regards to containment Engineered Safety Feature (ESF) systems, assuming the loss of one ESF bus, which is the worst case single active failure and results in one train of the Containment Spray System and one ARS fan being inoperable.

The limiting DBA analyses (Ref. 1) show that the maximum peak containment pressure results from the LOCA analysis and is calculated to be less than the containment design pressure. For certain aspects of the transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the cooling effectiveness of the ECCS during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. For these calculations, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the calculated transient containment pressures, in accordance with 10 CFR 50, Appendix K (Ref. 2).

The maximum peak containment atmosphere temperature results from the SLB analysis and is discussed in the Bases for LCO 3.6.5, "Containment Air Temperature."

In addition to calculating the overall peak containment pressures, the DBA analyses include calculation of the transient differential pressures that occur across subcompartment walls during the initial blowdown phase of the accident transient. The internal containment walls and structures are designed to withstand these local transient pressure differentials for the limiting DBAs.

BASES

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APPLICABLE SAFETY ANALYSES (continued)

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

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LCO The ice bed LCO requires the existence of the required quantity of stored ice, appropriate distribution of the ice within the ice bed, open flow paths through the ice bed, and appropriate chemical content and pH of the stored ice. The stored ice functions to absorb heat during the blowdown phase and long term phase of a DBA, thereby limiting containment air temperature and pressure. The chemical content and pH of the stored ice provide core SDM (boron content) and remove radioactive iodine from the containment atmosphere when the melted ice is recirculated through the ECCS and the Containment Spray System, respectively.

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APPLICABILITY In MODES 1, 2, 3, and 4, a DBA could cause an increase in containment pressure and temperature requiring the operation of the ice bed. Therefore, the LCO is applicable in MODES 1, 2, 3, and 4.

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the ice bed is not required to be OPERABLE in these MODES.

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ACTIONS A.1

If the ice bed is inoperable, it must be restored to OPERABLE status within 48 hours. The Completion Time was developed based on operating experience, which confirms that due to the very large mass of stored ice, the parameters comprising OPERABILITY do not change appreciably in this time period. If a degraded condition is identified, even for temperature, with such a large mass of ice it is not possible for the degraded condition to significantly degrade further in a 48 hour period. Therefore, 48 hours is a reasonable amount of time to correct a degraded condition before initiating a shutdown.

B.1 and B.2

If the ice bed 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.

BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.6.12.1

Verifying that the maximum temperature of the ice bed is  $\leq 27^{\circ}\text{F}$  ensures that the ice is kept well below the melting point. The 12 hour Frequency was based on operating experience, which confirmed that, due to the large mass of stored ice, it is not possible for the ice bed temperature to degrade significantly within a 12 hour period and was also based on assessing the proximity of the LCO limit to the melting temperature.

Furthermore, the 12 hour Frequency is considered adequate in view of indications in the control room, including the alarm, to alert the operator to an abnormal ice bed temperature condition. This SR may be satisfied by use of the Ice Bed Temperature Monitoring System.

SR 3.6.12.2

The minimum weight figure of 1145 pounds of ice per basket contains a 15% conservative allowance for ice loss through sublimation which is a factor of 15 higher than assumed for the ice condenser design. The minimum weight figure of 2,225,880 pounds of ice also contains an additional 1% conservative allowance to account for systematic error in weighing instruments.

The Frequency of 18 months was based on ice storage tests and the allowance built into the required ice mass over and above the mass assumed in the safety analyses. Operating experience has verified that, with the 18-month Frequency, the weight requirements are maintained with no significant degradation between surveillances.

SR 3.6.12.3

This SR ensures that the flow channels through the ice bed have not accumulated ice blockage that exceeds 15 percent of the total flow area through the ice bed region. The allowable 15 percent buildup of ice is based on the analysis of the sub-compartment response to a design basis LOCA with partial blockage of the ice condenser flow channels. The analysis did not perform detailed flow area modeling, but lumped the ice condenser bays into six sections ranging from 2.75 bays to 6.5 bays. Individual bays are acceptable with greater than 15 percent blockage, as long as 15 percent blockage is not exceeded for any analysis section.

To provide a 95 percent confidence that flow blockage does not exceed the allowed 15 percent, the visual inspection must be made for at least 54 (33 percent) of the 162 flow channels per ice condenser bay. The visual inspection of the ice bed flow channels is to inspect the flow area, by looking down from the top of the ice bed, and where view is achievable up

## BASES

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### SURVEILLANCE REQUIREMENTS (continued)

from the bottom of the ice bed. Flow channels to be inspected are determined by random sample. As the most restrictive ice bed flow passage is found at a lattice frame elevation, the 15 percent blockage criteria only applies to "flow channels" that comprise the area:

- a. between ice baskets, and
- b. past lattice frames and wall panels.

Due to a significantly larger flow area in the regions of the upper deck grating and the lower inlet plenum support structures and turning vanes, a gross buildup of ice on these structures would be required to degrade air and steam flow. Therefore, these structures are excluded as part of a flow channel for application of the 15 percent blockage criteria. Industry experience has shown that removal of ice from the excluded structures during the refueling outage is sufficient to ensure they remain OPERABLE throughout the operating cycle. Removal of any gross ice buildup on the excluded structures is performed following outage maintenance activities.

Operating experience has demonstrated that the ice bed is the region that is the most flow restrictive, due to the normal presence of ice accumulation on lattice frames and wall panels. The flow area through the ice basket support platform is not a more restrictive flow area because it is easily accessible from the lower plenum and is maintained clear of ice accumulation. There is no mechanistically credible method for ice to accumulate on the ice basket support platform during plant operation. Plant and industry experience has shown that the vertical flow area through the ice basket support platform remains clear of ice accumulation that could produce blockage. Normally only a glaze may develop or exist on the ice basket support platform which is not significant to blockage of flow area. Additionally, outage maintenance practices provide measures to clear the ice basket support platform following maintenance activities of any accumulation of ice that could block flow areas.

Frost buildup or loose ice is not to be considered as flow channel blockage, whereas attached ice is considered blockage of a flow channel. Frost is the solid form of water that is loosely adherent, and can be brushed off with the open hand.

#### SR 3.6.12.4

Verifying the chemical composition of the stored ice ensures that the stored ice has a boron concentration  $\geq 1800$  ppm and  $\leq 2500$  ppm as sodium tetraborate and a high pH,  $\geq 9.0$  and  $\leq 9.5$ , in order to meet the requirement for borated water when the melted ice is used in the ECCS

BASES

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## SURVEILLANCE REQUIREMENTS (continued)

recirculation mode of operation. Additionally, the minimum boron concentration value is used to assure reactor subcriticality in a post LOCA environment, while the maximum boron concentration is used as the bounding value in the hot leg switchover timing calculation (Ref. 3). This is accomplished by obtaining at least 24 ice samples. Each sample is taken approximately one foot from the top of the ice of each randomly selected ice basket in each ice condenser bay. The SR is modified by a Note that allows the boron concentration and pH value obtained from averaging the individual samples' analysis results to satisfy the requirements of the SR. If either the average boron concentration or average pH value is outside their prescribed limit, then entry into Condition A is required. Sodium tetraborate has been proven effective in maintaining the boron content for long storage periods, and it also enhances the ability of the solution to remove and retain fission product iodine. The high pH is required to enhance the effectiveness of the ice and the melted ice in removing iodine from the containment atmosphere. This pH range also minimizes the occurrence of chloride and caustic stress corrosion on mechanical systems and components exposed to ECCS and Containment Spray System fluids in the recirculation mode of operation.

The Frequency of 54 months is intended to be consistent with the expected length of three fuel cycles, and was developed considering these facts:

- a. Long term ice storage tests have determined that the chemical composition of the stored ice is extremely stable,
- b. There are no normal operating mechanisms that decrease the boron concentration of the stored ice, and pH remains within a 9.0-9.5 range when boron concentrations are above approximately 1200 ppm.
- c. Operating experience has demonstrated that meeting the boron concentration and pH requirements has never been a problem, and
- d. Someone would have to enter the containment to take the sample, and, if the unit is at power, that person would receive a radiation dose.

SR 3.6.12.5

This SR ensures that a representative sampling of ice baskets, which are relatively thin walled, perforated cylinders, have not been degraded by wear, cracks, corrosion, or other damage. The Frequency of 40 months for a visual inspection of the structural soundness of the ice baskets is based on engineering judgment and considers such factors as the

## BASES

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### SURVEILLANCE REQUIREMENTS (continued)

thickness of the basket walls relative to corrosion rates expected in their services environment and the results of the long term ice storage testing.

#### SR 3.6.12.6

This SR ensures that initial ice fill and any subsequent ice additions meet the boron concentration and pH requirements of SR 3.6.12.4. The SR is modified by a Note that allows the chemical analysis to be performed on either the liquid or resulting ice of each sodium tetraborate solution prepared. If ice is obtained from offsite sources, then chemical analysis data must be obtained for the ice supplied.

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### REFERENCES

1. UFSAR, Section 6.2.
  2. 10 CFR 50, Appendix K.
  3. Sequoyah Nuclear Plant Units 1 and 2 – Nuclear Steam Supply System Engineering Support Services – Contract 99NAN-251787 – Letter N9873, Contract Work Authorization N20000 020 – Tritium Production Core – Post Loss of Coolant Accident (LOCA) Long Term Core Cooling Analysis – N2N 058, dated August 13, 2001.
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## B 3.6 CONTAINMENT SYSTEMS

### B 3.6.13 Ice Condenser Doors

#### BASES

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- BACKGROUND** The ice condenser doors consist of the inlet doors, the intermediate deck doors, and the top deck doors. The functions of the doors are to:
- a. Seal the ice condenser from air leakage during the lifetime of the unit; and
  - b. Open in the event of a Design Basis Accident (DBA) to direct the hot steam air mixture from the DBA into the ice bed, where the ice would absorb energy and limit containment peak pressure and temperature during the accident transient.

Limiting the pressure and temperature following a DBA reduces the release of fission product radioactivity from containment to the environment.

The ice condenser is an annular compartment enclosing approximately 300 degrees of the perimeter of the upper containment compartment, but penetrating the operating deck so that a portion extends into the lower containment compartment. The inlet doors separate the atmosphere of the lower compartment from the ice bed inside the ice condenser. The top deck doors are above the ice bed and are exposed to the atmosphere of the upper compartment. The intermediate deck doors, located below the top deck doors, form the floor of a plenum at the upper part of the ice condenser. This plenum area is used to facilitate surveillance and maintenance of the ice bed.

The ice baskets held in the ice bed within the ice condenser are arranged to promote heat transfer from steam to ice. This arrangement enhances the ice condenser's primary function of condensing steam and absorbing heat energy released to the containment during a DBA.

In the event of a DBA, the ice condenser inlet doors (located below the operating deck) open due to the pressure rise in the lower compartment. This allows air and steam to flow from the lower compartment into the ice condenser. The resulting pressure increase within the ice condenser causes the intermediate deck doors and the top deck doors to open, which allows the air to flow out of the ice condenser into the upper compartment. Steam condensation within the ice condensers limits the pressure and temperature buildup in containment. A divider barrier separates the upper and lower compartments and ensures that the steam is directed into the ice condenser.

## BASES

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### BACKGROUND (continued)

The ice, together with the containment spray, serves as a containment heat removal system and is adequate to absorb the initial blowdown of steam and water from a DBA as well as the additional heat loads that would enter containment during the several hours following the initial blowdown. The additional heat loads would come from the residual heat in the reactor core, the hot piping and components, and the secondary system, including the steam generators. During the post blowdown period, the Air Return System (ARS) returns upper compartment air through the divider barrier to the lower compartment. This serves to continue circulating heated air and steam from the lower compartment through the ice condenser, where the heat is removed by the remaining ice.

The water from the melted ice drains into the lower compartment where it serves as a source of borated water (via the containment sump) for the Emergency Core Cooling System (ECCS) and the Containment Spray System heat removal functions in the recirculation mode. The ice (via the Containment Spray System) and the recirculated ice melt also serve to clean up the containment atmosphere.

The ice condenser doors ensure that the ice stored in the ice bed is preserved during normal operation (doors closed) and that the ice condenser functions as designed if called upon to act as a passive heat sink following a DBA.

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### APPLICABLE SAFETY ANALYSES

The limiting DBAs considered relative to containment pressure and temperature are the loss of coolant accident (LOCA) and the steam line break (SLB). The LOCA and SLB are analyzed using computer codes designed to predict the resultant containment pressure and temperature transients. DBAs are assumed not to occur simultaneously or consecutively.

Although the ice condenser is a passive system that requires no electrical power to perform its function, the Containment Spray System and ARS also function to assist the ice bed in limiting pressures and temperatures. Therefore, the postulated DBAs are analyzed with respect to Engineered Safety Feature (ESF) systems, assuming the loss of one ESF bus, which is the worst case single active failure and results in one train of the Containment Spray System and one ARS fan being rendered inoperable.

The limiting DBA analyses (Ref. 1) show that the maximum peak containment pressure results from the LOCA analysis and is calculated to be less than the containment design pressure. For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the cooling effectiveness of

## BASES

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### APPLICABLE SAFETY ANALYSES (continued)

the ECCS during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. For these calculations, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the calculated transient containment pressures, in accordance with 10 CFR 50, Appendix K (Ref. 2).

The maximum peak containment atmosphere temperature results from the SLB analysis and is discussed in the Bases for LCO 3.6.5, "Containment Air Temperature."

An additional design requirement was imposed on the ice condenser door design for a small break accident in which the flow of heated air and steam is not sufficient to fully open the doors.

For this situation, the doors are designed so that all of the doors would partially open by approximately the same amount. Thus, the partially opened doors would modulate the flow so that each ice bay would receive an approximately equal fraction of the total flow.

This design feature ensures that the heated air and steam will not flow preferentially to some ice bays and deplete the ice there without utilizing the ice in the other bays.

In addition to calculating the overall peak containment pressures, the DBA analyses include the calculation of the transient differential pressures that would occur across subcompartment walls during the initial blowdown phase of the accident transient. The internal containment walls and structures are designed to withstand the local transient pressure differentials for the limiting DBAs.

The ice condenser doors satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

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### LCO

This LCO establishes the minimum equipment requirements to assure that the ice condenser doors perform their safety function. The ice condenser inlet doors, intermediate deck doors, and top deck doors must be closed to minimize air leakage into and out of the ice condenser, with its attendant leakage of heat into the ice condenser and loss of ice through melting and sublimation. The doors must be OPERABLE to ensure the proper opening of the ice condenser in the event of a DBA. OPERABILITY includes being free of any obstructions that would limit their opening, and for the inlet doors, being adjusted such that the opening and closing torques are within limits. The ice condenser doors function with the ice condenser to limit the pressure and temperature that could be expected following a DBA.

## BASES

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**APPLICABILITY** In MODES 1, 2, 3, and 4, a DBA could cause an increase in containment pressure and temperature requiring the operation of the ice condenser doors. Therefore, the LCO is applicable in MODES 1, 2, 3, and 4.

The probability and consequences of these events in MODES 5 and 6 are reduced due to the pressure and temperature limitations of these MODES. Therefore, the ice condenser doors are not required to be OPERABLE in these MODES.

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**ACTIONS** Note 1 provides clarification that, for this LCO, separate Condition entry is allowed for each ice condenser door.

Note 2 has been added to allow an intermediate deck or top deck door to be inoperable for a short duration solely due to personnel standing on or opening the door to perform required Surveillances, minor preventative maintenance, or system walkdowns, and does not require entry into Condition B. This is acceptable since the ice bed temperature is normally continuously monitored using an alarm in the control room, which alarms on an increasing ice bed temperature.

### A.1

If one or more ice condenser inlet doors are inoperable due to being physically restrained from opening, the door(s) must be restored to OPERABLE status within 1 hour. The Required Action is necessary to return operation to within the bounds of the containment analysis. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1, "Containment," which requires containment to be restored to OPERABLE status within 1 hour.

### B.1 and B.2

If one or more ice condenser doors are determined to be partially open or otherwise inoperable for reasons other than Condition A or if a door is found that is not closed, it is acceptable to continue unit operation for up to 14 days, provided the ice bed temperature is monitored once per 4 hours to ensure that the open or inoperable door is not allowing enough air leakage to cause the maximum ice bed temperature to approach the melting point. The Completion Time of once per 4 hours is based on the fact that temperature changes cannot occur rapidly in the ice bed because of the large mass of ice involved. The 14 day Completion Time is based on long term ice storage tests that indicate that if the temperature is maintained at or below 27°F, there would not be a significant loss of ice from sublimation. If the maximum ice bed temperature is > 27°F at any time, the situation reverts to Condition C and a Completion Time of 48 hours is allowed to restore the inoperable door

## BASES

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### ACTIONS (continued)

to OPERABLE status or enter into Required Actions D.1 and D.2. Ice bed temperature must be verified to be within the specified Frequency as augmented by the provisions of SR 3.0.2. If this verification is not made, Required Actions D.1 and D.2, not Required Action C.1, must be taken. Entry into Condition B is not required due to personnel standing on or opening an intermediate deck or upper deck door for short durations to perform required surveillances, minor maintenance such as ice removal, or routine tasks such as system walkdowns.

#### C.1

If Required Actions B.1 or B.2 are not met, the doors must be restored to OPERABLE status and closed positions within 48 hours. The 48 hour Completion Time is based on the fact that, with the very large mass of ice involved, it would not be possible for the temperature to decrease to the melting point and a significant amount of ice to melt in a 48 hour period. Condition C is entered from Condition B only when the Completion Time of Required Action B.2 is not met or when the ice bed temperature has not been verified at the required frequency.

#### D.1 and D.2

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

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### SURVEILLANCE REQUIREMENTS

#### SR 3.6.13.1

Verifying, by means of the Inlet Door Position Monitoring System, that the inlet doors are in their closed positions makes the operator aware of an inadvertent opening of one or more doors.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

#### SR 3.6.13.2

Verifying, by visual inspection, that each intermediate deck door is closed and not impaired by ice, frost, or debris provides assurance that the

## BASES

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### SURVEILLANCE REQUIREMENTS (continued)

intermediate deck doors (which form the floor of the upper plenum where frequent maintenance on the ice bed is performed) have not been left open or obstructed.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

#### SR 3.6.13.3

Verifying, by visual inspection, that the ice condenser inlet doors are not impaired by ice, frost, or debris provides assurance that the doors are free to open in the event of a DBA.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

#### SR 3.6.13.4

Verifying the opening torque of the inlet doors provides assurance that no doors have become stuck in the closed position. The value of 675 in-lb is based on the design opening pressure on the doors of 1.0 lb/ft<sup>2</sup>.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

#### SR 3.6.13.5

The torque test Surveillance ensures that the inlet doors have not developed excessive friction and that the return springs are producing a door return torque within limits. The torque test consists of the following:

1. Verify that the torque, T(OPEN), required to cause opening motion at the 40° open position is < 195 in-lb,
2. Verify that the torque, T(CLOSE), required to hold the door stationary (i.e., keep it from closing) at the 40° open position is > 78 in-lb, and
3. Calculate the frictional torque,  $T(\text{FRICT}) = 0.5 \{T(\text{OPEN}) - T(\text{CLOSE})\}$ , and verify that the T(FRICT) is  $\leq 40$  in-lb.

The purpose of the friction and return torque Specifications is to ensure that, in the event of a small break LOCA or SLB, all of the 24 door pairs open uniformly. This assures that, during the initial blowdown phase, the

BASES

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SURVEILLANCE REQUIREMENTS (continued)

steam and water mixture entering the lower compartment does not pass through part of the ice condenser, depleting the ice there, while bypassing the ice in other bays.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.6.13.6

Verifying the OPERABILITY of the intermediate deck doors provides assurance that the intermediate deck doors are free to open in the event of a DBA. The verification consists of visually inspecting the intermediate doors for structural deterioration, verifying free movement of the vent assemblies, and ascertaining free movement of each door when lifted with the applicable force shown below:

<u>Door</u>	<u>Lifting Force</u>
a. 0-1, 0-5	≤ 37.4 lb
b. 0-2, 0-6	≤ 33.8 lb
c. 0-3, 0-7	≤ 31.0 lb
d. 0-4, 0-8	≤ 31.8 lb

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.6.13.7

Verifying, by visual inspection, that the top deck doors are in place, closed, and not obstructed provides assurance that the doors are performing their function of keeping warm air out of the ice condenser during normal operation, and would not be obstructed if called upon to open in response to a DBA.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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REFERENCES

1. UFSAR, Chapter 6.
  2. 10 CFR 50, Appendix K.
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## B 3.6 CONTAINMENT SYSTEMS

### B 3.6.14 Divider Barrier Integrity

#### BASES

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**BACKGROUND** The divider barrier consists of the operating deck and associated seals, personnel access doors, and equipment hatches that separate the upper and lower containment compartments. Divider barrier integrity is necessary to minimize bypassing of the ice condenser by the hot steam and air mixture released into the lower compartment during a Design Basis Accident (DBA). This ensures that most of the gases pass through the ice bed, which condenses the steam and limits pressure and temperature during the accident transient. Limiting the pressure and temperature reduces the release of fission product radioactivity from containment to the environment in the event of a DBA.

In the event of a DBA, the ice condenser inlet doors (located below the operating deck) open due to the pressure rise in the lower compartment. This allows air and steam to flow from the lower compartment into the ice condenser. The resulting pressure increase within the ice condenser causes the intermediate deck doors and the door panels at the top of the condenser to open, which allows the air to flow out of the ice condenser into the upper compartment. The ice condenses the steam as it enters, thus limiting the pressure and temperature buildup in containment. The divider barrier separates the upper and lower compartments and ensures that the steam is directed into the ice condenser. The ice, together with the containment spray, is adequate to absorb the initial blowdown of steam and water from a DBA as well as the additional heat loads that would enter containment over several hours following the initial blowdown. The additional heat loads would come from the residual heat in the reactor core, the hot piping and components, and the secondary system, including the steam generators. During the post blowdown period, the Air Return System (ARS) returns upper compartment air through the divider barrier to the lower compartment. This serves to continue circulating heated air and steam from the lower compartment through the ice condenser, where the heat is removed by the remaining ice.

Divider barrier integrity ensures that the high energy fluids released during a DBA would be directed through the ice condenser and that the ice condenser would function as designed if called upon to act as a passive heat sink following a DBA.

## BASES

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### APPLICABLE SAFETY ANALYSES

Divider barrier integrity ensures the functioning of the ice condenser to the limiting containment pressure and temperature that could be experienced following a DBA. The limiting DBAs considered relative to containment temperature and pressure are the loss of coolant accident (LOCA) and the steam line break (SLB). The LOCA and SLB are analyzed using computer codes designed to predict the resultant containment pressure and temperature transients. DBAs are assumed not to occur simultaneously or consecutively.

Although the ice condenser is a passive system that requires no electrical power to perform its function, the Containment Spray System and the ARS also function to assist the ice bed in limiting pressures and temperatures. Therefore, the postulated DBAs are analyzed, with respect to containment Engineered Safety Feature (ESF) systems, assuming the loss of one ESF bus, which is the worst case single active failure and results in the inoperability of one train in the Containment Spray System and one ARS fan.

The limiting DBA analyses (Ref. 1) show that the maximum peak containment pressure results from the LOCA analysis and is calculated to be less than the containment design pressure. The maximum peak containment temperature results from the SLB analysis and is discussed in the Bases for LCO 3.6.5, "Containment Air Temperature."

In addition to calculating the overall peak containment pressures, the DBA analyses include calculation of the transient differential pressures that occur across subcompartment walls during the initial blowdown phase of the accident transient. The internal containment walls and structures are designed to withstand these local transient pressure differentials for the limiting DBAs.

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

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### LCO

This LCO establishes the minimum equipment requirements to ensure that the divider barrier performs its safety function of ensuring that bypass leakage, in the event of a DBA, does not exceed the bypass leakage assumed in the accident analysis. Included are the requirements that the personnel access doors and equipment hatches in the divider barrier are OPERABLE and closed and that the divider barrier seal is properly installed and has not degraded with time. An exception to the requirement that the doors be closed is made to allow personnel transit through the divider barrier. The basis of this exception is the assumption that, for personnel transit, the time during which a door is open will be short (i.e., shorter than the Completion Time of 1 hour for Condition A). The divider barrier functions with the ice condenser to limit the pressure and temperature that could be expected following a DBA.

BASES

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APPLICABILITY In MODES 1, 2, 3, and 4, a DBA could cause an increase in containment pressure and temperature requiring the integrity of the divider barrier. Therefore, the LCO is applicable in MODES 1, 2, 3, and 4.

The probability and consequences of these events in MODES 5 and 6 are low due to the pressure and temperature limitations of these MODES. As such, divider barrier integrity is not required in these MODES.

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ACTIONS

A.1

If one or more personnel access doors or equipment hatches are inoperable or open, 1 hour is allowed to restore the door(s) and equipment hatches to OPERABLE status and the closed position. The 1 hour Completion Time is consistent with LCO 3.6.1, "Containment," which requires that containment be restored to OPERABLE status within 1 hour.

Condition A has been modified by a Note to provide clarification that separate Condition entry is allowed for each personnel access door or equipment hatch.

B.1

If the divider barrier seal is inoperable, 1 hour is allowed to restore the seal to OPERABLE status. The 1 hour Completion Time is consistent with LCO 3.6.1, which requires that containment be restored to OPERABLE status within 1 hour.

C.1 and C.2

If divider barrier integrity 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.

BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.6.14.1

Verification, by visual inspection, that the personnel access doors and equipment hatches between the upper and lower containment compartments are closed provides assurance that divider barrier integrity is maintained prior to the reactor being taken from MODE 5 to MODE 4. This SR is necessary because many of the doors and hatches may have been opened for maintenance during the shutdown.

SR 3.6.14.2

Verification, by visual inspection, that the personnel access door and equipment hatch seals, sealing surfaces, and alignments are acceptable provides assurance that divider barrier integrity is maintained. This inspection cannot be made when the door or hatch is closed. Therefore, SR 3.6.14.2 is required for each door or hatch that has been opened, prior to the final closure. Some doors and hatches may not be opened for long periods of time. Those that use resilient materials in the seals must be opened and inspected periodically to provide assurance that the seal material has not aged to the point of degraded performance.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.6.14.3

Verification, by visual inspection, after each opening of a personnel access door or equipment hatch that it has been closed makes the operator aware of the importance of closing it and thereby provides additional assurance that divider barrier integrity is maintained while in applicable MODES.

SR 3.6.14.4

Conducting periodic physical property tests on divider barrier seal test coupons provides assurance that the seal material has not degraded in the containment environment, including the effects of irradiation with the reactor at power. The required test consists of a differential pressure test. The test sequence will be as follows: two coupons will be tested to 60 psid; with no failures, the results are acceptable. If a failure occurs at 60 psid, four coupons will be tested to 30 psid; with no failures, the results are acceptable. If a failure occurs at 30 psid, five coupons will be sent to the manufacturer for LOCA environment simulation (radiation, humidity, temperature) and testing to 15 psid.

BASES

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SURVEILLANCE REQUIREMENTS (continued)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.6.14.5

Visual inspection of the seal around the perimeter provides assurance that the seal is properly secured in place.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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REFERENCES

1. UFSAR, Section 6.2.
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## B 3.6 CONTAINMENT SYSTEMS

### B 3.6.15 Containment Recirculation Drains

#### BASES

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<b>BACKGROUND</b>	<p>The containment recirculation drains consist of the ice condenser drains and the refueling canal drains. The ice condenser is partitioned into 24 bays, each having a pair of inlet doors that open from the bottom plenum to allow the hot steam-air mixture from a Design Basis Accident (DBA) to enter the ice condenser. The drains shall provide a flow area out of the ice condenser of at least 15 square feet. No more than two adjacent bays shall be without drains. Each drain leads to a drain pipe that drops down several feet, then makes one or more 90° bends and exits into the lower compartment. A check (flapper) valve at the end of each pipe keeps warm air from entering during normal operation, but when the water exerts pressure, it opens to allow the water to spill into the lower compartment. This prevents water from backing up and interfering with the ice condenser inlet doors. The water delivered to the lower containment serves to cool the atmosphere as it falls through to the floor and provides a source of borated water at the containment sump for long term use by the Emergency Core Cooling System (ECCS) and the Containment Spray System during the recirculation mode of operation.</p> <p>The two refueling canal drains are at low points in the refueling canal. During a refueling, plugs are installed in the drains and the canal is flooded to facilitate the refueling process. The water acts to shield and cool the spent fuel as it is transferred from the reactor vessel to storage. After refueling, the canal is drained and the plugs removed. In the event of a DBA, the refueling canal drains are the main return path to the lower compartment for Containment Spray System water sprayed into the upper compartment.</p> <p>The ice condenser drains and the refueling canal drains function with the ice bed, the Containment Spray System, and the ECCS to limit the pressure and temperature that could be expected following a DBA.</p>
<b>APPLICABLE SAFETY ANALYSES</b>	<p>The limiting DBAs considered relative to containment temperature and pressure are the loss of coolant accident (LOCA) and the steam line break (SLB). The LOCA and SLB are analyzed using computer codes designed to predict the resultant containment pressure and temperature transients. DBAs are assumed not to occur simultaneously or consecutively. Although the ice condenser is a passive system that requires no electrical power to perform its function, the Containment Spray System and the Air Return System (ARS) also function to assist the ice bed in limiting pressures and temperatures. Therefore, the</p>

BASES

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APPLICABLE SAFETY ANALYSES (continued)

analysis of the postulated DBAs, with respect to Engineered Safety Feature (ESF) systems, assumes the loss of one ESF bus, which is the worst case single active failure and results in one train of the Containment Spray System and one ARS fan being rendered inoperable.

The limiting DBA analyses (Ref. 1) show that the maximum peak containment pressure results from the LOCA analysis and is calculated to be less than the containment design pressure. The maximum peak containment atmosphere temperature results from the SLB analysis and is discussed in the Bases for LCO 3.6.5, "Containment Air Temperature." In addition to calculating the overall peak containment pressures, the DBA analyses include calculation of the transient differential pressures that occur across subcompartment walls during the initial blowdown phase of the accident transient. The internal containment walls and structures are designed to withstand these local transient pressure differentials for the limiting DBAs.

The containment recirculation drains satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

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LCO

This LCO establishes the minimum requirements to ensure that the containment recirculation drains perform their safety functions. The ice condenser floor drain valve disks must be closed to minimize air leakage into and out of the ice condenser during normal operation and must open in the event of a DBA when water begins to drain out. The refueling canal drains must have their plugs removed and remain clear to ensure the return of Containment Spray System water to the lower containment in the event of a DBA. The containment recirculation drains function with the ice condenser, ECCS, and Containment Spray System to limit the pressure and temperature that could be expected following a DBA.

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APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause an increase in containment pressure and temperature, which would require the operation of the containment recirculation drains. Therefore, the LCO is applicable in MODES 1, 2, 3, and 4.

The probability and consequences of these events in MODES 5 and 6 are low due to the pressure and temperature limitations of these MODES. As such, the containment recirculation drains are not required to be OPERABLE in these MODES.

BASES

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ACTIONS

A.1

If one ice condenser floor drain is inoperable, 1 hour is allowed to restore the drain to OPERABLE status. The Required Action is necessary to return operation to within the bounds of the containment analysis. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1, "Containment," which requires that containment be restored to OPERABLE status within 1 hour.

B.1

If one refueling canal drain is inoperable, 1 hour is allowed to restore the drain to OPERABLE status. The Required Action is necessary to return operation to within the bounds of the containment analysis. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1, which requires that containment be restored to OPERABLE status in 1 hour.

C.1 and C.2

If the affected drain(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 and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

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SURVEILLANCE  
REQUIREMENTS

SR 3.6.15.1

Verifying the OPERABILITY of the refueling canal drains ensures that they will be able to perform their functions in the event of a DBA. This Surveillance confirms that the refueling canal drain plugs have been removed and that the drains are clear of any obstructions that could impair their functioning. In addition to debris near the drains, attention must be given to any debris that is located where it could be moved to the drains in the event that the Containment Spray System is in operation and water is flowing to the drains. SR 3.6.15.1 must be performed before entering MODE 4 from MODE 5 after every filling of the canal to ensure that the plugs have been removed and that no debris that could impair the drains was deposited during the time the canal was filled.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.15.2

Verifying the OPERABILITY of the ice condenser floor drains ensures that they will be able to perform their functions in the event of a DBA. Inspecting the drain valve disk ensures that the valve is performing its function of sealing the drain line from warm air leakage into the ice condenser during normal operation, yet will open if melted ice fills the line following a DBA. Verifying that the drain lines are not obstructed ensures their readiness to drain water from the ice condenser.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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REFERENCES      1. UFSAR, Sections 6.2 and 6.5.

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## B 3.7 PLANT SYSTEMS

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

#### BASES

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BACKGROUND	<p>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 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.</p> <p>Five MSSVs are located on each main steam header, outside containment, upstream of the main steam isolation valves, as described in UFSAR, Section 10.3.2 (Ref. 1). The MSSVs must have sufficient capacity to limit the secondary system pressure to <math>\leq 110\%</math> of the steam generator design pressure in order to meet the requirements of the ASME Code, Section III (Ref. 2). The MSSV design includes staggered setpoints, according to Table 3.7.1-2 in the accompanying LCO, so that only the needed valves will actuate. Staggered setpoints reduce the potential for valve chattering that is due to steam pressure insufficient to fully open all valves following a turbine reactor trip.</p>
APPLICABLE SAFETY ANALYSES	<p>The design basis for the MSSVs comes from Reference 2 and its purpose is to limit the secondary system pressure to <math>\leq 110\%</math> of design pressure for any anticipated operational occurrence (AOO) or accident considered in the Design Basis Accident (DBA) and transient analysis.</p> <p>The events that challenge the relieving capacity of the MSSVs, and thus RCS pressure, are those characterized as decreased heat removal events, which are presented in UFSAR, Section 15.2.7 (Ref. 3). Of these, the full power turbine trip without steam dump is the limiting AOO. This event also terminates normal feedwater flow to the steam generators.</p> <p>The safety analysis demonstrates that the transient response for turbine trip occurring from full power without a direct reactor trip presents no hazard to the integrity of the RCS or the Main Steam System. One turbine trip analysis is performed assuming primary system pressure control via operation of the pressurizer relief valves and spray. This analysis demonstrates that the DNB (Departure from Nucleate Boiling) design basis is met. Another analysis is performed assuming no primary system pressure control, but crediting reactor trip on high pressurizer pressure and operation of the pressurizer safety valves. This analysis demonstrates that RCS integrity is maintained by showing that the maximum RCS pressure does not exceed 110% of the design pressure. All cases analyzed demonstrate that the MSSVs maintain Main Steam System integrity by limiting the maximum steam pressure to less than or equal to 110% of the steam generator design pressure.</p>

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## BASES

## APPLICABLE SAFETY ANALYSES (continued)

In addition to the decreased heat removal events, reactivity insertion events may also challenge the relieving capacity of the MSSVs. The UFSAR, Section 15.2.2 (Ref. 7) safety analysis of the uncontrolled rod cluster control assembly (RCCA) bank withdrawal at power event is characterized by an increase in core power and steam generation rate until reactor trip occurs when the Overtemperature  $\Delta T$ , Overpower  $\Delta T$ , High Pressurizer Pressure, High Pressurizer Water Level, or Power Range Neutron Flux-High setpoint is reached. Steam flow to the turbine will not increase from its initial value for this event. The increased heat transfer to the secondary side causes an increase in steam pressure and may result in opening of the MSSVs prior to reactor trip, assuming no credit for operation of the atmospheric relief or condenser steam dump valves. The analysis of the RCCA bank withdrawal (Reference 8) at power slow event demonstrates that the MSSVs are capable of preventing secondary side overpressurization for this AOO.

The UFSAR safety analyses discussed above assume that all of the MSSVs for each steam generator are OPERABLE. If there are inoperable MSSV(s), it is necessary to limit the primary system power during steady-state operation and AOOs to a value that does not result in exceeding the combined steam flow capacity of the turbine (if available) and the remaining OPERABLE MSSVs. The required limitation on primary system power necessary to prevent secondary system overpressurization may be determined by system transient analyses or conservatively arrived at by a simple heat balance calculation. In some circumstances it is necessary to limit the primary side heat generation that can be achieved during an AOO by reducing the setpoint of the Power Range Neutron Flux-High reactor trip function.

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

## LCO

The accident analysis requires that five MSSVs per steam generator be OPERABLE to provide overpressure protection for design basis transients occurring at 102% RTP. The LCO requires that five MSSVs per steam generator be OPERABLE in compliance with Reference 2, and the DBA analysis.

The OPERABILITY of the MSSVs is defined as the ability to open upon demand within the setpoint tolerances, to relieve steam generator overpressure, and reseal when pressure has been reduced. The OPERABILITY of the MSSVs is determined by periodic surveillance testing in accordance with the Inservice Testing Program.

This LCO provides assurance that the MSSVs will perform their designed safety functions to mitigate the consequences of accidents that could result in a challenge to the RCPB, or Main Steam System integrity.

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**BASES**

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**APPLICABILITY** In MODES 1, 2, and 3, five MSSVs per steam generator are required to be OPERABLE to prevent Main Steam System overpressurization.

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.

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

With one or more MSSVs inoperable, action must be taken so that the available MSSV relieving capacity meets Reference 2 requirements. Operation with less than all five MSSVs OPERABLE for each steam generator is permissible, if THERMAL POWER is limited to the relief capacity of the remaining MSSVs. This is accomplished by restricting THERMAL POWER so that the energy transfer to the most limiting steam generator is not greater than the available relief capacity in that steam generator.

A.1 and A.2

With one or more inoperable MSSVs on one or more steam generators, Required Action A.1 requires an appropriate reduction in THERMAL POWER within 4 hours. Therefore, with a reactor power reduction alone there may be insufficient total steam flow capacity provided by the remaining OPERABLE MSSVs to preclude overpressurization in the event of a turbine trip without steam dump. Therefore, a Completion Time of 36 hours is allowed in Required Action A.2 to reduce the Power Range Neutron Flux – High reactor trip setpoints. The Completion Time of 36 hours is based on a reasonable time to correct the MSSV inoperability, the time required to perform the power reduction, operating experience in resetting all channels of a protective function, and on the low probability of the occurrence of a transient that could result in steam generator overpressure during this period.

The maximum THERMAL POWER corresponding to the heat removal capacity of the remaining OPERABLE MSSVs is determined via a conservative heat balance calculation as described in the attachment to Reference 6, with an appropriate allowance for Nuclear Instrumentation System trip channel uncertainties.

Required Action A.2 is modified by a Note, indicating that the Power Range Neutron Flux-High reactor trip setpoint reduction is only required in MODE 1. In MODES 2 and 3 the reactor protection system trips specified in LCO 3.3.1, "Reactor Trip System Instrumentation," provide sufficient protection.

BASES

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## ACTIONS (continued)

The allowed Completion Times are reasonable based on operating experience to accomplish the Required Actions in an orderly manner without challenging unit systems.

B.1 and B.2

If the Required Actions are not completed within the associated Completion Time, or if one or more steam generators have  $\geq 4$  inoperable MSSVs, 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.

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SURVEILLANCE  
REQUIREMENTSSR 3.7.1.1

This SR verifies the OPERABILITY of the MSSVs by the verification of each MSSV lift setpoint in accordance with the Inservice Testing Program. The ASME Code (Ref. 4), requires that safety and relief valve tests be performed in accordance with ANSI/ASME OM-1-2001 through 2003 Addenda (Ref. 5). According to Reference 5, the following tests are required:

- 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 be tested every 24 months. The ASME Code specifies the activities and frequencies necessary to satisfy the requirements. Table 3.7.1-2 allows a  $\pm 3\%$  setpoint tolerance for OPERABILITY; however, the valves are reset to  $\pm 1\%$  during the Surveillance to allow for drift. The lift settings, according to Table 3.7.1-2, correspond to ambient conditions of the valve at nominal operating temperature and pressure.

This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. The MSSVs may be either bench

## BASES

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### SURVEILLANCE REQUIREMENTS (continued)

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

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### REFERENCES

1. UFSAR, Section 10.3.2.
  2. ASME, Boiler and Pressure Vessel Code, Section III, dated 1968, and March 1970 Addenda.
  3. UFSAR, Section 15.2.7.
  4. ASME Code for Operation and Maintenance of Nuclear Power Plants.
  5. ANSI/ASME OM-1-2001 through 2003 Addenda.
  6. NRC Information Notice 94-60, "Potential Overpressurization of the Main Steam System," August 22, 1994.
  7. UFSAR, Section 15.2.2.
  8. AREVA Document 51-5006459-00, "SQN Uncontrolled RCCA Withdrawal Accident Analysis Profile."
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## B 3.7 PLANT SYSTEMS

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

#### BASES

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BACKGROUND	<p>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 generators.</p> <p>One MSIV is located in each main steam line outside, but close to, containment. The MSIVs are downstream from the main steam safety valves (MSSVs) and auxiliary feedwater (AFW) pump turbine steam supply, to prevent MSSV and AFW isolation from the steam generators by MSIV closure. Closing the MSIVs isolates each steam generator from the others, and isolates the turbine, Steam Dump System, and other auxiliary steam supplies from the steam generators.</p> <p>The MSIVs close on a main steam isolation signal generated by either low steam line pressure, high - high containment pressure, or high steam pressure rate. The MSIVs fail closed on loss of control power or control air.</p> <p>Each MSIV has an MSIV bypass valve. Although these bypass valves are normally closed, they receive the same emergency closure signal as do their associated MSIVs. The MSIVs may also be actuated manually.</p> <p>A description of the MSIVs is found in the UFSAR, Section 10.3 (Ref. 1).</p>
APPLICABLE SAFETY ANALYSES	<p>The design basis of the MSIVs is established by the containment analysis for the large steam line break (SLB) inside containment, discussed in the UFSAR, Section 6.2 (Ref. 2). It is also affected by the accident analysis of the SLB events presented in the UFSAR, Section 15.4.2 (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).</p> <p>The limiting case for the containment analysis is the SLB inside containment, with a loss of offsite power following turbine trip, and failure of the MSIV on the affected steam generator to close. At lower powers, the steam generator inventory and temperature are at their maximum, maximizing the analyzed mass and energy release to the containment. Due to reverse flow and failure of the MSIV to close, the additional mass and energy in the steam headers downstream from the other MSIVs contribute to the total release. With the most reactive rod cluster control assembly assumed stuck in the fully withdrawn position, there is an increased possibility that the core will become critical and return to power. The core is ultimately shut down by the boric acid injection delivered by the Emergency Core Cooling System.</p>

BASES

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## APPLICABLE SAFETY ANALYSES (continued)

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

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

- a. A HELB inside containment. In order to maximize the mass and energy release into containment, the analysis assumes that the MSIV for the affected steam generator remains open. For this accident scenario, steam is discharged into containment from all steam generators until the remaining MSIVs close. After MSIV closure, steam is discharged into containment only from the affected steam generator and from the residual steam in the main steam header downstream of the closed MSIVs in the unaffected loops. Closure of the MSIVs isolates the break from the unaffected steam generators.
- b. A break outside of containment and upstream from the MSIVs is not a containment pressurization concern. The uncontrolled blowdown of more than one steam generator must be prevented to limit the potential for uncontrolled RCS cooldown and positive reactivity addition. Closure of the MSIVs isolates the break and limits the blowdown to a single steam generator.
- c. A break downstream of the MSIVs will be isolated by the closure of the MSIVs.
- d. Following a steam generator tube rupture, closure of the MSIVs isolates the ruptured steam generator from the intact steam generators to minimize radiological releases.
- e. The MSIVs are also utilized during other events such as a feedwater line break. This event is less limiting so far as MSIV OPERABILITY is concerned.

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

BASES

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LCO This LCO requires that four MSIVs in the 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.

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

In MODE 4, 5 or 6, the steam generator energy is low; therefore, the MSIVs are not required for isolation of potential high energy secondary system pipe breaks in these MODES.

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ACTIONS A.1

With one MSIV inoperable in MODE 1, action must be taken to restore OPERABLE status within 4 hours. Some repairs to the MSIV can be made with the unit hot. The 4 hour Completion Time is reasonable, considering the low probability of an accident occurring during this time period that would require a closure of the MSIVs.

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 4 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 Times are reasonable, based on operating experience, to reach MODE 2 and to close the MSIVs in an orderly manner and without challenging unit systems.

BASES

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## ACTIONS (continued)

C.1 and C.2

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

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

The 4 hour Completion Time is consistent with that allowed in Condition A.

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

D.1 and D.2

If the MSIVs cannot be restored to OPERABLE status or are not 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 at least in MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from MODE 2 conditions in an orderly manner and without challenging unit systems.

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SURVEILLANCE  
REQUIREMENTSSR 3.7.2.1

This SR verifies that the closure time of each MSIV is within 5 seconds. The MSIV isolation time is within the limit assumed in the accident and containment analyses. This SR also verifies the valve closure time is in accordance with the Inservice Testing Program. This SR is normally performed upon returning the unit to operation following a refueling outage. The MSIVs should not be tested at power, since even a part stroke exercise increases the risk of a valve closure when the unit is generating power. As the MSIVs are not tested at power, they are exempt from the ASME Code (Ref. 5), requirements during operation in MODE 1 or 2.

## BASES

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### SURVEILLANCE REQUIREMENTS (continued)

The Frequency is in accordance with the Inservice Testing Program.

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

#### SR 3.7.2.2

This SR verifies that each MSIV can close on an actual or simulated actuation signal. This Surveillance is normally performed upon returning the plant to operation following a refueling outage.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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.

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### REFERENCES

1. UFSAR, Section 10.3.
  2. UFSAR, Section 6.2.
  3. UFSAR, Section 15.4.2.
  4. 10 CFR 100.11.
  5. ASME Code for Operation and Maintenance of Nuclear Power Plants.
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## B 3.7 PLANT SYSTEMS

### B 3.7.3 Main Feedwater Isolation Valves (MFIVs), Main Feedwater Regulating Valves (MFRVs) and MFRV Bypass Valves

#### BASES

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**BACKGROUND** The MFIVs isolate main feedwater (MFW) flow to the secondary side of the steam generators following a high energy line break (HELB). The safety related function of the MFRVs is to provide the second isolation of MFW flow to the secondary side of the steam generators following a HELB. Closure of the MFIVs, or MFRVs and MFRV bypass valves terminates flow to the steam generators, terminating the event for feedwater line breaks (FWLBs) occurring upstream of the MFIVs or MFRVs. The consequences of events occurring in the main steam lines or in the MFW lines downstream from the MFIVs will be mitigated by their closure. Closure of the MFIVs, or MFRVs and MFRV 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, or MFRVs and MFRV bypass valves, isolate the nonsafety related portions from the safety related portions of the system. In the event of a secondary side pipe rupture inside containment, the valves limit the quantity of high energy fluid that enters containment through the break, and provide a pressure boundary for the controlled addition of auxiliary feedwater (AFW) to the intact loops.

One MFIV, and one MFRV and its MFRV bypass valve, are located on each MFW line, outside but close to containment. The MFIVs, MFRVs, and MFRV bypass valves are located upstream of the AFW injection point so that AFW may be supplied to the steam generators following MFIV or MFRV closure. The piping volume from these valves to the steam generators must be accounted for in calculating mass and energy releases, and refilled prior to AFW reaching the steam generator following either an SLB or FWLB.

The MFIVs, MFRVs, and MFRV bypass valves, close on receipt of a  $T_{avg}$  - Low coincident with reactor trip (P-4), safety injection, or steam generator water level high-high signal. They may also be actuated manually. In addition to the MFIVs, and the MFRVs and MFRV bypass valves, a check valve outside containment is available. The check valve isolates the feedwater line, penetrating containment, and ensures that the consequences of events do not exceed the capacity of the containment heat removal systems.

BASES

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BACKGROUND (continued)

A description of the MFIVs and MFRVs is found in UFSAR, Section 10.4.7 (Ref. 1).

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APPLICABLE  
SAFETY  
ANALYSES

The design basis of the MFIVs and MFRVs is established by the analyses for the large SLB. It is also influenced by the accident analysis for the large FWLB. Closure of the MFIVs, or MFRVs and MFRV bypass valves, may also be relied on to terminate an SLB for core response analysis and excess feedwater event upon the receipt of a steam generator water level high-high signal or a feedwater isolation signal on high steam generator level.

Failure of an MFIV, MFRV, or the MFRV bypass valves to close following an SLB or FWLB can result in additional mass and energy being delivered to the steam generators, contributing to cooldown. This failure also results in additional mass and energy releases following an SLB or FWLB event.

The MFIVs, MFRVs, and MFRV bypass valves satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

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LCO

This LCO ensures that the MFIVs, MFRVs, and MFRV bypass valves will isolate MFW flow to the steam generators, following an FWLB or main steam line break. These valves will isolate the nonsafety related portions from the safety related portions of the system.

This LCO requires that four MFIVs, four MFRVs and four MFRV bypass valves be OPERABLE. The MFIVs, MFRVs and the MFRV bypass valves are considered OPERABLE when isolation times are within limits and they close on an isolation actuation signal.

Failure to meet the LCO requirements can result in additional mass and energy being released to containment following an SLB or FWLB inside containment. If a feedwater isolation signal on high steam generator level is relied on to terminate an excess feedwater flow event, failure to meet the LCO may result in the introduction of water into the main steam lines.

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APPLICABILITY

The MFIVs, MFRVs and the MFRV bypass valves must be OPERABLE whenever there is significant mass and energy in the Reactor Coolant System and steam generators. This ensures that, in the event of a HELB, a single failure cannot result in the blowdown of more than one steam generator. In MODES 1, 2, and 3, the MFIVs, MFRVs and the MFRV bypass valves are required to be OPERABLE 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 or isolated by a closed manual valve, they are already performing their safety function.

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BASES

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APPLICABILITY (continued)

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

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ACTIONS

The ACTIONS Table is modified by a Note indicating that separate Condition entry is allowed for each valve. This includes separate Condition entry for two valves in the same flow path being inoperable. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable valve. Complying with the Required Actions may allow for continued operation, and subsequent inoperable valves are governed by subsequent Condition entry and application of associated Required Actions.

A.1 and A.2

With one MFIV in one or more flow paths inoperable, action must be taken to restore the affected valves to OPERABLE status, or to close or isolate inoperable affected valves within 72 hours. When these valves are closed or isolated, they are performing their required safety function.

The 72 hour Completion Time takes into account the redundancy afforded by the remaining OPERABLE valves and the low probability of an event occurring during this time period that would require isolation of the MFW flow paths. The 72 hour Completion Time is reasonable, based on operating experience.

Inoperable MFIVs that are closed or isolated must be verified on a periodic basis that they are closed or isolated. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of valve status indications available in the control room, and other administrative controls, to ensure that these valves are closed or isolated.

B.1 and B.2

With one MFRV in one or more flow paths inoperable, action must be taken to restore the affected valves to OPERABLE status, or to close or isolate inoperable affected valves within 72 hours. When these valves are closed or isolated, they are performing their required safety function.

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

## BASES

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### ACTIONS (continued)

flow paths. The 72 hour Completion Time is reasonable, based on operating experience.

Inoperable MFRVs, that are closed or isolated, must be verified on a periodic basis that they are closed or isolated. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of valve status indications available in the control room, and other administrative controls to ensure that the valves are closed or isolated.

#### C.1 and C.2

With one MFRV bypass valve in one or more flow paths inoperable, action must be taken to restore the affected valves to OPERABLE status, or to close or isolate inoperable affected valves within 72 hours. When these valves are closed or isolated, they are performing their required safety function.

The 72 hour Completion Time takes into account the redundancy afforded by the remaining OPERABLE valves and the low probability of an event occurring during this time period that would require isolation of the MFW flow paths. The 72 hour Completion Time is reasonable, based on operating experience.

Inoperable MFRV bypass valves that are closed or isolated must be verified on a periodic basis that they are closed or isolated. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of valve status indications available in the control room, and other administrative controls, to ensure that these valves are closed or isolated.

#### D.1

With two valves in one or more flow paths inoperable, there may be no redundant system to operate automatically and perform the required safety function. Although the containment can be isolated with the failure of two valves in parallel in the same flow path, the double failure 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, at least one valve in each flow path must be restored to OPERABLE status, or the affected flow path isolated within 8 hours. This action returns the system to the condition where at least one valve in each flow path is performing the required

BASES

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ACTIONS (continued)

safety function. The 8 hour Completion Time is reasonable, based on operating experience, to complete the actions required to close the MFIV or MFRV, or otherwise isolate the affected flow path.

E.1 and E.2

If the MFIV(s) and MFRV(s) and the MFRV bypass valve(s) cannot be restored to OPERABLE status, or closed, or isolated 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 full power conditions in an orderly manner and without challenging unit systems.

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SURVEILLANCE  
REQUIREMENTS

SR 3.7.3.1

This SR verifies that the closure time of each MFIV, MFRV, and MFRV bypass valve is within the limit given in Reference 2 and is within that assumed in the accident and containment analyses. This SR also verifies the valve closure time is in accordance with the Inservice Testing Program. This SR is normally performed upon returning the unit to operation following a refueling outage. These valves should not be tested at power since even a part stroke exercise increases the risk of a valve closure with the unit generating power. This is consistent with the ASME Code (Ref. 3), quarterly stroke requirements during operation in MODES 1 and 2.

The Frequency for this SR is in accordance with the Inservice Testing Program.

SR 3.7.3.2

This SR verifies that each MFIV, MFRV, and MFRV bypass valves can close on an actual or simulated actuation signal. This Surveillance is normally performed upon returning the plant to operation following a refueling outage.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

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- REFERENCES
1. UFSAR, Section 10.4.7.
  2. UFSAR, Section 7.3.
  3. ASME Code for Operation and Maintenance of Nuclear Power Plants.
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## B 3.7 PLANT SYSTEMS

### B 3.7.4 Atmospheric Relief Valves (ARVs)

#### BASES

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**BACKGROUND** The ARVs provide a method for cooling the unit to residual heat removal (RHR) entry conditions should the preferred heat sink via the Steam Dump System to the condenser not be available, as discussed in the UFSAR, 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 ARVs may also be required to meet the cooldown when steam pressure drops too low for maintenance of a vacuum in the condenser to permit use of the Steam Dump System.

One ARV line for each of the four steam generators is provided. Each ARV line consists of one ARV.

The ARVs are equipped with pneumatic controllers to permit control of the cooldown rate. The air supplies to the ARVs are from two trains from the plant safety grade Auxiliary Control Air System (ACAS). ACAS train A supplies air to ARVs for steam generators 1 and 3 and train B supplies air to ARVs for steam generators 2 and 4. The ARVs receive the necessary electrical power from the 125 volt vital battery system.

A description of the ARVs is found in Reference 1. The ARVs are OPERABLE with a DC power source and plant safety grade air supply available. In addition, handwheels are provided for manual operation of ARVs for steam generators 1 and 4. Air cylinders connected at control stations outside containment provide an alternate means of operation of the ARVs for steam generators 2 and 3.

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**APPLICABLE SAFETY ANALYSES** The design basis of the ARVs is established by the capability to cool the unit to RHR entry conditions. The ARVs, since their set pressure is slightly lower than the safety valves, prevent excessive lifting of the safety valves. Only two ARVs are required for plant cool down following any credible event.

In the accident analysis presented in Reference 2, the ARVs are assumed to be used by the operator to cool down the unit to RHR entry conditions for accidents accompanied by a loss of offsite power. In Reference 3 (SGTR), the ARVs are assumed to be available following a steam generator tube rupture accompanied by a loss of offsite power. The ARVs allow the operator to establish sufficient subcooling in the RCS so that the primary system will remain subcooled after the RCS pressure is decreased to stop primary to secondary break flow into the ruptured steam generator. Four ARVs are required to be OPERABLE to allow operators to initiate the RCS cooldown, following a steam generator tube rupture, using the ARVs on the intact steam generators. This cooldown

BASES

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APPLICABLE SAFETY ANALYSES (continued)

supports the termination of break flow within the required time specified in the accident analysis to prevent steam generator overfill.

The time required to terminate the primary to secondary break flow for an SGTR is more critical than the time required to cool down to RHR conditions for this event and also for other accidents. Thus, the SGTR is the limiting event for the ARVs.

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

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LCO

Four ARV lines are required to be OPERABLE. One ARV line is required from each of four steam generators to ensure that at least three ARV lines are available to conduct a unit cooldown to establish sufficient subcooling in the RCS following an SGTR, in which one steam generator becomes unavailable.

Failure to meet the LCO can result in the inability to cooldown the RCS to establish sufficient subcooling and prevent steam generator overfill following the steam generator rupture when the condenser is unavailable for use with the Steam Dump System.

An ARV is considered OPERABLE when it is capable of providing controlled relief of the main steam flow and capable of fully opening and closing from the main control room.

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APPLICABILITY

In MODES 1, 2, and 3, and in MODE 4, when a steam generator is being relied upon for heat removal, the ARVs are required to be OPERABLE.

In MODE 5 or 6, an SGTR is not a credible event.

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ACTIONS

A.1

With one or more ARV lines inoperable due to one train of ACAS nonfunctional, action must be taken to restore the ACAS train to functional status within 72 hours. The 72 hour Completion Time is reasonable to repair the nonfunctional ACAS train, based on the availability of the remaining OPERABLE ARV lines, the alternate means to control the inoperable ARVs and the low probability of an event occurring during the time the ACAS train is nonfunctional. Alternate means of operation include valve reach rod handwheels and backup air bottles at the control stations outside containment.

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BASES

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ACTIONS (continued)

B.1

With one or more ARV lines inoperable for reasons other than Condition A, action must be taken to restore all ARV lines to OPERABLE status. The 24 hour Completion Time is reasonable to repair inoperable ARV lines, based on the availability of the Steam Dump System and MSSVs, and the low probability of an event occurring during this period that would require the ARV lines.

C.1 and C.2

If the ARV lines cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4, without reliance upon steam generator for heat removal, within 24 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

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SURVEILLANCE  
REQUIREMENTS

SR 3.7.4.1

To perform a controlled cooldown of the RCS, the ARVs must be able to be opened remotely and throttled through their full range. This SR ensures that the ARVs are tested through a full control cycle at least once per fuel cycle. Performance of inservice testing or use of an ARV during a unit cooldown may satisfy this requirement.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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REFERENCES

1. UFSAR, Section 10.3.
  2. UFSAR, Chapter 15.
  3. UFSAR, Section 15.4.3.
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## B 3.7 PLANT SYSTEMS

### B 3.7.5 Auxiliary Feedwater (AFW) System

#### BASES

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#### 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. Except for the common miniflow line to and supply line from the condensate storage tanks and some shared support facilities such as the condensate storage tanks and parts of the Control Air System, the two reactor units have separate AFW Systems. The normal suction for both units AFW pumps is through a common header connected to two condensate storage tanks (CSTs) (LCO 3.7.6, "Condensate Storage Tank (CST)"). The pumps are grouped into unit specific systems with each systems' pumps aligned to their respective units steam generators secondary side via connections to the main feedwater piping between the main feedwater isolation check valve and the steam generator. The nonessential condensate supply is isolated from the essential portion of the AFW System by check valves. A low AFW pump suction pressure automatically actuates valves from ERCW on a two-out-of-three signal to align the AFW pump suction to ERCW to ensure the AFW pumps have an adequate water supply. 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 relief valves (LCO 3.7.4, "Atmospheric Relief Valves (ARVs)"). If the main condenser is available, steam may be released via the steam dump valves and recirculated to the CST.

The AFW System consists of two motor driven AFW pumps and one steam turbine driven pump configured into three trains. Each motor driven pump provides 100% of AFW flow capacity, and the turbine driven pump provides 200% of the required capacity to the steam generators, as assumed in the accident analysis, except for the Feedwater Line Break (FWLB) and Small Break Loss-of-Coolant accident (SBLOCA). The pumps are equipped with 1½ inch recirculation lines to prevent pump operation against a closed system. Each motor driven AFW pump is powered from an independent Class 1E power supply and feeds two steam generators. The steam turbine driven AFW pump receives steam from two main steam lines upstream of the main steam isolation valves. Each of the steam feed lines will supply 100% of the requirements of the turbine driven AFW pump.

The AFW System is capable of supplying feedwater to the steam generators during normal unit startup, shutdown, and hot standby conditions.

BASES

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BACKGROUND (continued)

The turbine driven AFW pump supplies a common header capable of feeding all steam generators with DC powered pneumatic control valves actuated to open by the Engineered Safety Feature Actuation System (ESFAS). One pump at full flow is sufficient to remove decay heat and cool the unit to residual heat removal (RHR) entry conditions. Thus, the requirement for diversity in motive power sources for the AFW System is met.

The AFW System is designed to supply sufficient water to the steam generator(s) to remove decay heat with steam generator pressure at the setpoint of the MSSVs. Subsequently, the AFW System supplies sufficient water to cool the unit to RHR entry conditions, with steam released through the ARVs.

The AFW System actuates automatically on steam generator water level low-low by the ESFAS (LCO 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation"). The system also actuates on loss of offsite power, safety injection, initiation of Anticipated Transient Without Scram (ATWS) Mitigation Actuation Circuitry (AMSAC), trip of one MFW pump with turbine load above 77% Unit 2, and trip of all MFW pumps. The AFW System actuations on an AMSAC signal and on a MFW pump trip/power coincident signal are not required as part of this LCO.

The AFW System is discussed in the UFSAR, Section 10.4.7.2 (Ref. 1).

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APPLICABLE  
SAFETY  
ANALYSES

The AFW System mitigates the consequences of any event with loss of normal feedwater.

The design basis of the AFW System is to supply water to the steam generators to remove decay heat and other residual heat by delivering at least the minimum required flow rate to the steam generators at pressures corresponding to the lowest steam generator safety valve set pressure plus 3%.

In addition, the AFW System must supply enough makeup water to replace steam generator secondary inventory lost as the unit cools to MODE 4 conditions. Sufficient AFW flow must also be available to account for flow losses such as pump recirculation and line breaks.

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

- a. Feedwater Line Break (FWLB) and
- b. Loss of MFW.

BASES

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APPLICABLE SAFETY ANALYSES (continued)

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

The AFW System design is such that it can perform its function following an FWLB between the MFW isolation valves and containment, combined with a loss of offsite power following turbine trip, and a single active failure of the steam turbine driven AFW pump. In such a case, one motor driven AFW pump would deliver to the broken MFW header at the pump runout flow until the problem was detected, and flow terminated by the operator. Sufficient flow would be delivered to the intact steam generator by the redundant AFW pump.

The ESFAS automatically actuates the AFW turbine driven pump and associated power operated valves and controls when required to ensure an adequate feedwater supply to the steam generators during loss of power. Air operated valves are provided for each AFW line to control the AFW flow to each steam generator.

The AFW System satisfies the requirements of Criterion 3 of 10 CFR 50.36(c)(2)(ii).

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LCO

This LCO provides assurance that the AFW System will perform its design safety function to mitigate the consequences of accidents that could result in overpressurization of the reactor coolant pressure boundary. Three independent AFW pumps in three diverse trains are required to be OPERABLE to ensure the availability of decay heat removal for all events accompanied by a loss of offsite power and a single failure. This is accomplished by powering two of the pumps from independent emergency buses. The third AFW pump is powered by a different means, a steam driven turbine supplied with steam from a source that is not isolated by closure of the MSIVs.

The AFW System is configured into three trains. The AFW System is considered OPERABLE when the components and flow paths required to provide redundant AFW flow to the steam generators are OPERABLE. This requires that the two motor driven AFW pumps be OPERABLE in two diverse paths, each supplying AFW to separate steam generators. The turbine driven AFW pump is required to be OPERABLE with redundant steam supplies from each of two main steam lines upstream of the MSIVs, and shall be capable of supplying AFW to any of the steam generators. The piping, valves, instrumentation, and controls in the required flow paths are also required to be OPERABLE.

BASES

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## LCO (continued)

Each motor-driven auxiliary feedwater pump (one Train A and one Train B) supplies flow paths to two steam generators. Each flow path contains an automatic air-operated level control valve (LCV). The LCVs have the same train designation as the associated pump and are provided trained air. The turbine driven auxiliary feedwater pump supplies flow paths to all four steam generators. Each of these flow paths contains an automatic opening (non-modulating) air-operated LCV, two of which are designated as Train A, receive A-train air, and provide flow to the same steam generators that are supplied by the B-train motor-driven auxiliary feedwater pump. The remaining two LCVs are designated as Train B, receive B-train air, and provide flow to the same steam generators that are supplied by the A-train motor-driven pump. This design provides the required redundancy to ensure that at least two steam generators receive the necessary flow assuming any single failure. It can be seen from the description provided above that the loss of a single train of air (A or B) will not prevent the auxiliary feedwater system from performing its intended safety function and is no more severe than the loss of a single auxiliary feedwater pump. Therefore, the loss of a single train of auxiliary air only affects the capability of a single motor-driven auxiliary feedwater pump because the turbine-driven pump is still capable of providing flow to two steam generators that are separate from the other motor-driven pump.

Two redundant steam sources are required to be operable to ensure that at least one source is available for the steam-driven auxiliary feedwater (AFW) pump operation following a feedwater or main steam line break. This requirement ensures that the plant remains within its design basis (i.e., AFW to two intact steam generators) given the event of a loss of the No.1 steam generator because of a main steam line or feedwater line break and a single failure of the B-train motor driven AFW pump. The two redundant sources must be aligned such that No.1 steam generator source is open and operable and the No.4 steam generator source is closed and operable.

For instances where one train of emergency raw cooling water (ERCW) is declared inoperable in accordance with technical specifications, the AFW turbine-driven pump is considered operable since it is supplied by both trains of ERCW. Similarly, the AFW turbine-driven pump is considered operable when one train of the AFW loss of power start function is declared inoperable in accordance with Technical Specifications because both 6.9 kilovolt shutdown board logic trains supply this function. This position is consistent with American National Standards Institute/ANSI 58.9 requirements (i.e., postulation of the failure of the opposite train is not required while relying on the TS limiting condition for operation).

## BASES

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### LCO (continued)

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

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### APPLICABILITY

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

In MODE 4 the AFW System may be used for heat removal via the steam generators.

In MODE 5 or 6, the steam generators are not normally used for heat removal, and the AFW System is not required.

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### ACTIONS

A Note prohibits the application of LCO 3.0.4.b to an inoperable AFW train. There is an increased risk associated with entering a MODE or other specified condition in the Applicability with an AFW train inoperable and the provisions of LCO 3.0.4.b, which allow entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, should not be applied in this circumstance.

#### A.1

If the turbine driven AFW train is inoperable due to one inoperable steam supply, or if a turbine driven pump is inoperable for any reason while in MODE 3 immediately following refueling, action must be taken to restore the inoperable equipment to an OPERABLE status within 7 days. The 7 day Completion Time is reasonable, based on the following reasons:

- a. For the inoperability of the turbine driven AFW pump due to one inoperable steam supply, the 7 day Completion Time is reasonable since there is a redundant steam supply line for the turbine driven pump and the turbine driven train is still capable of performing its specified function for most postulated events.
- b. For the inoperability of a turbine driven AFW pump while in MODE 3 immediately subsequent to a refueling, the 7 day Completion Time is reasonable due to the minimal decay heat levels in this situation.

## BASES

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### ACTIONS (continued)

- c. For both the inoperability of the turbine driven pump due to one inoperable steam supply and an inoperable turbine driven AFW pump while in MODE 3 immediately following a refueling outage, the 7 day Completion Time is reasonable due to the availability of redundant OPERABLE motor driven AFW pumps, and due to the low probability of an event requiring the use of the turbine driven AFW pump.

Condition A is modified by a Note which limits the applicability of the Condition for an inoperable turbine driven AFW pump in MODE 3 to when the unit has not entered MODE 2 following refueling. Condition A allows one AFW train to be inoperable for 7 days vice the 72 hour Completion Time in Condition B. This longer Completion Time is based on the reduced decay heat following refueling and prior to the reactor being critical.

#### B.1

With one of the required AFW trains (pump or flow path) inoperable in MODE 1, 2, or 3 for reasons other than Condition A, 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 occurring during this time period.

#### C.1 and C.2

With one of the required motor driven AFW trains (pump or flow path) inoperable and the turbine driven AFW train inoperable due to one inoperable steam supply, action must be taken to restore the affected equipment to OPERABLE status within 48 hours. Assuming no single active failures when in this condition, the accident (a feedline break (FLB) or main steam line break (MSLB)) could result in the loss of the remaining steam supply to the turbine driven AFW pump due to the faulted steam generator (SG).

## BASES

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### ACTIONS (continued)

The 48 hour Completion Time is reasonable based on the fact that the remaining motor driven AFW train is capable of providing 100% of the AFW flow requirements, and the low probability of an event occurring that would challenge the AFW system.

#### D.1 and D.2

When Required Action A.1, B.1, C.1, or C.2 cannot be completed within the required Completion Time, or if two AFW trains are inoperable in MODE 1, 2, or 3 for reasons other than Condition C, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 18 hours.

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

In MODE 4 with two AFW trains inoperable, operation is allowed to continue because only one motor driven pump AFW train is required in accordance with the Note that modifies the LCO. Although not required, the unit may continue to cool down and initiate RHR.

#### E.1

If all three AFW trains are inoperable in MODE 1, 2, or 3, the unit is in a seriously degraded condition with no safety related means for conducting a cooldown, and only limited means for conducting a cooldown with nonsafety related 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.

Required Action E.1 is modified by a Note indicating that all required MODE changes are suspended until one AFW train is restored to OPERABLE status. In this case, LCO 3.0.3 is not applicable because it could force the unit into a less safe condition.

## BASES

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### ACTIONS (continued)

#### F.1

In MODE 4, either the reactor coolant pumps or the RHR loops can be used to provide forced circulation. This is addressed in LCO 3.4.6, "RCS Loops - MODE 4." With the required AFW train inoperable, action must be taken to immediately restore the inoperable train to OPERABLE status. The immediate Completion Time is consistent with LCO 3.4.6.

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### SURVEILLANCE REQUIREMENTS

#### SR 3.7.5.1

Verifying the correct alignment for manual, power operated, and automatic valves in the AFW System water and steam supply flow paths provides assurance that the proper flow paths will exist for AFW 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 being mispositioned are in the correct position.

The SR is modified by a Note that states one or more AFW trains may be considered OPERABLE during alignment and operation for steam generator level control, if it is capable of being manually (i.e., remotely or locally, as appropriate) realigned to the AFW mode of operation, provided it is not otherwise inoperable. This exception allows the system to be out of its normal standby alignment and temporarily incapable of automatic initiation without declaring the train(s) inoperable. Since AFW may be used during startup, shutdown, hot standby operations, and hot shutdown operations for steam generator level control, and these manual operations are an accepted function of the AFW System, OPERABILITY (i.e., the intended safety function) continues to be maintained.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

#### SR 3.7.5.2

Verifying that each AFW pump's developed head at the flow test point is greater than or equal to the required developed head ensures that AFW pump performance has not degraded during the cycle. Flow and differential head are normal tests of centrifugal pump performance required by the ASME Code (Ref 2). Because it is undesirable to introduce cold AFW into the steam generators while they are operating,

## BASES

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### SURVEILLANCE REQUIREMENTS (continued)

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, trend performance, and detect incipient failures by indicating abnormal performance. Performance of inservice testing discussed in the ASME Code (Ref. 2) (only required at 3 month intervals) satisfies this requirement.

This SR is modified by a Note indicating that the SR should be deferred until suitable test conditions are established. This deferral is required because there is insufficient steam pressure to perform the test.

#### SR 3.7.5.3

This SR verifies that AFW can be delivered to the appropriate steam generator in the event of any accident or transient that generates an ESFAS, or in the event the CSTs become depleted, 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 Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

The SR is modified by two Notes. Note 1 states that one or more AFW trains may be considered OPERABLE during alignment and operation for steam generator level control, if it is capable of being manually (i.e., remotely or locally, as appropriate) realigned to the AFW mode of operation, provided it is not otherwise inoperable. This exception allows the system to be out of its normal standby alignment and temporarily incapable of automatic initiation without declaring the train(s) inoperable. Since AFW may be used during startup, shutdown, hot standby operations, and hot shutdown operations for steam generator level control, and these manual operations are an accepted function of the AFW System, OPERABILITY (i.e., the intended safety function) continues to be maintained. Note 2 states that the SR is only required to be met in MODES 1, 2, and 3. It is not required to be met in MODE 4, since the AFW train is only required for the purposes of removing decay heat when the SG is relied upon for heat removal. The operation of the AFW train is by manual means and automatic startup of the AFW train is not required.

BASES

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.5.4

This SR verifies that the AFW pumps will start in the event of any accident or transient that generates an ESFAS by demonstrating that each AFW pump starts automatically on an actual or simulated actuation signal in MODES 1, 2, and 3. In MODE 4, the required pump is already operating and the autostart function is not required.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

This SR is modified by three Notes. Note 1 indicates that the SR may be deferred until suitable test conditions are established. This deferral is required because there is insufficient steam pressure to perform the test. Note 2 states that one or more AFW trains may be considered OPERABLE during alignment and operation for steam generator level control, if it is capable of being manually (i.e., remotely or locally, as appropriate) realigned to the AFW mode of operation, provided it is not otherwise inoperable. This exception allows the system to be out of its normal standby alignment and temporarily incapable of automatic initiation without declaring the train(s) inoperable. Since AFW may be used during startup, shutdown, hot standby operations, and hot shutdown operations for steam generator level control, and these manual operations are an accepted function of the AFW System. OPERABILITY (i.e., the intended safety function) continues to be maintained. Note 3 states that the SR is only required to be met in MODES 1, 2, and 3. It is not required to be met in MODE 4, since the AFW train is only required for the purposes of removing decay heat when the SG is relied upon for heat removal. The operation of the AFW train is by manual means and automatic startup of the AFW train is not required.

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REFERENCES

1. UFSAR, Section 10.4.7.2.
  2. ASME Code for Operation and Maintenance of Nuclear Power Plants.
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## B 3.7 PLANT SYSTEMS

### B 3.7.6 Condensate Storage Tank (CST)

#### BASES

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**BACKGROUND** The CST provides the preferred 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 3.7.5). The steam produced is released to the atmosphere by the main steam safety valves or the atmospheric relief 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 dump valves. The condensed steam is returned to the CST by the hotwell pump. This has the advantage of conserving condensate while minimizing releases to the environment.

The CST consists of a non-seismic qualified carbon steel tank with a capacity of 385,000 gallons. The CST is the preferred and primary source of clean water for the AFW System. The essential raw cooling water (ERCW) system is the backup source of water in addition to being the Safety Grade source of water. The ERCW supply can be manually aligned based on CST level or automatically aligned on a two-out-of-three low-pressure signal in the condensate suction line.

A description of the CST is found in the UFSAR, 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 events in the accident analysis as discussed in the UFSAR, Chapters 6 and 15 (Refs. 2 and 3, respectively). For anticipated operational occurrences and accidents that do not affect the OPERABILITY of the steam generators, the analysis assumption is generally 2 hours at MODE 3, steaming through the MSSVs, followed by a cooldown to residual heat removal (RHR) entry conditions at the design cooldown rate.

The limiting event for the condensate volume is the large feedwater line break coincident with a loss of offsite power. Single failures that also affect this event include the following:

- a. Failure of the diesel generator powering the motor driven AFW pump to the unaffected steam generator (requiring additional steam to drive the remaining AFW pump turbine) and
- b. Failure of the steam driven AFW pump (requiring a longer time for cooldown using only one motor driven AFW pump).

BASES

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APPLICABLE SAFETY ANALYSES (continued)

These are not usually the limiting failures in terms of consequences for these events.

A nonlimiting event is considered a break in either the main feedwater or AFW line near where the two join. This break has the potential for dumping condensate until terminated by operator action. This loss of condensate inventory is partially compensated for by the retention of steam generator inventory.

The CST satisfies Criteria 2 and 3 of 10 CFR 50.36(c)(2)(ii).

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LCO

The CST must contain sufficient cooling water to remove decay heat for 2 hours following a reactor trip from 100.7% RTP, and then to cool down the RCS to RHR entry conditions, assuming a coincident loss of offsite power. In doing this, it must retain sufficient water to ensure adequate net positive suction head for the AFW pumps during cooldown.

The CST level required is equivalent to a usable volume of  $\geq 240,000$  gallons, which is based on holding the unit in MODE 3 for 2 hours, followed by a cooldown to RHR entry conditions within the following 6 hours. This basis is established in Reference 4 and is the minimum volume required for a plant cooldown.

The OPERABILITY of the CST is determined by maintaining the tank level at or above the minimum required level.

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APPLICABILITY

In MODES 1, 2, and 3, and in MODE 4, when steam generator is being relied upon for heat removal, the CST is required to be OPERABLE.

In MODE 5 or 6, the CST is not required because the AFW System is not required.

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ACTIONS

A.1 and A.2

If the CST is not OPERABLE, the OPERABILITY of the backup supply should be verified by administrative means within 4 hours and once every 12 hours thereafter. OPERABILITY of the backup feedwater supply must include verification that the flow paths from the backup water supply to the AFW pumps are OPERABLE, and that the backup supply has the required volume of water available. The CST must be restored to OPERABLE status within 7 days, because the backup supply may be performing this function in addition to its normal functions. The 4 hour Completion Time is reasonable, based on operating experience, to verify the OPERABILITY of the backup water supply. Additionally, verifying the

BASES

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ACTIONS (continued)

backup water supply every 12 hours is adequate to ensure the backup water supply continues to be available. The 7 day Completion Time is reasonable, based on an OPERABLE backup water supply being available, and the low probability of an event occurring during this time period requiring the CST.

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 the steam generator for heat removal, within 18 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

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SURVEILLANCE  
REQUIREMENTS

SR 3.7.6.1

This SR verifies that the CST contains the required volume of cooling water.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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REFERENCES

1. UFSAR, Section 9.2.6.
  2. UFSAR, Chapter 6.
  3. UFSAR, Chapter 15.
  4. UFSAR, Section 10.4.7.
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## B 3.7 PLANT SYSTEMS

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

#### BASES

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<b>BACKGROUND</b>	<p>The CCS 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 CCS also provides this function for various nonessential components, as well as the spent fuel storage pool. The CCS serves as a barrier to the release of radioactive byproducts between potentially radioactive systems and the Essential Raw Cooling Water (ERCW) System, and thus to the environment.</p> <p>The CCS is arranged as two independent, full capacity cooling loops, and has isolatable nonsafety related components. Although each unit's trains are independent, the CCS B trains share components. Up to three of the five CCS pumps may be shared and the two B train component cooling heat exchangers are shared between the two units. Normally, only CCS pump C-S (common-spare) will be aligned to the train B headers of both units along with both 0B heat exchangers, however, either pump 1B-B (Unit 1) or 2B-B (Unit 2) can be realigned to the train B headers if necessary. 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 board, except for the C-S pump which is powered from shared boards. An OPERABLE C-S pump is powered from the Unit 2 "B" board. It can, however, be manually transferred to the Unit 1 "A" board. When the C-S pump is powered from the Unit 1 "A" board, it is considered inoperable because the configuration is not tested. An open surge tank in the system provides an automatic makeup function to ensure that sufficient net positive suction head is available. The pump in each train is automatically started on receipt of a safety injection signal (unit specific safety injection signals except for the C-S pump, which starts from either units safety injection signal).</p> <p>Additional information on the design and operation of the system, along with a list of the components served, is presented in the UFSAR, Section 9.2.1 (Ref. 1). The principal safety related function of the CCS is the removal of decay heat from the reactor via the Residual Heat Removal (RHR) System. This may be during a normal or post accident cooldown and shutdown.</p>
<b>APPLICABLE SAFETY ANALYSES</b>	<p>The design basis of the CCS is for one CCS train to remove the post loss of coolant accident (LOCA) heat load from the containment sump during the recirculation phase, with a maximum CCS temperature of 104.5°F (Ref. 1). The Emergency Core Cooling System (ECCS) LOCA and containment OPERABILITY LOCA each model the maximum and minimum performance of the CCS, respectively. The normal temperature of the CCS is 35 - 95°F, and, during unit cooldown to MODE 5</p>

BASES

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APPLICABLE SAFETY ANALYSES (continued)

( $T_{avg} < 200^{\circ}\text{F}$ ), a maximum temperature of  $120^{\circ}\text{F}$  can be approached. This prevents the containment sump fluid from increasing in temperature during the recirculation phase following a LOCA, and provides a gradual reduction in the temperature of this fluid as it is supplied to the Reactor Coolant System (RCS) by the ECCS pumps.

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

The CCS also functions to cool the unit from RHR entry conditions ( $T_{avg} < 350^{\circ}\text{F}$ ), to MODE 5 ( $T_{avg} < 200^{\circ}\text{F}$ ), during normal and post accident operations. The time required to cool from  $350^{\circ}\text{F}$  to  $200^{\circ}\text{F}$  is a function of the number of CCS and RHR trains operating. One CCS train is sufficient to remove decay heat during subsequent operations with  $T_{avg} < 200^{\circ}\text{F}$ . This assumes a maximum ERCW temperature of  $87^{\circ}\text{F}$  occurring simultaneously with the maximum heat loads on the system.

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

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LCO

The CCS 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 CCS 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 trains of CCS must be OPERABLE. At least one CCS train will operate assuming the worst case single active failure occurs coincident with a loss of offsite power.

A CCS train is considered OPERABLE when:

- a. The pump and associated 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 CCS from other components or systems not required for safety may render those components or systems inoperable but does not affect the OPERABILITY of the CCS.

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APPLICABILITY

In MODES 1, 2, 3, and 4, the CCS is a normally operating system, which must be prepared to perform its post accident safety functions, primarily RCS heat removal, which is achieved by cooling the RHR heat exchanger.

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BASES

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APPLICABILITY (continued)

In MODE 5 or 6, the OPERABILITY requirements of the CCS are determined by the systems it supports.

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ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that the applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops - MODE 4," be entered if an inoperable CCS train results in an inoperable RHR loop. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components.

If one CCS train is inoperable, action must be taken to restore OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE CCS train is adequate to perform the heat removal function. The 72 hour Completion Time is reasonable, based on the redundant capabilities afforded by the OPERABLE train, and the low probability of a DBA occurring during this period.

B.1 and B.2

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

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SURVEILLANCE  
REQUIREMENTS

SR 3.7.7.1

This SR is modified by a Note indicating that the isolation of the CCS flow to individual components may render those components inoperable but does not affect the OPERABILITY of the CCS.

Verifying the correct alignment for manual, power operated, and automatic valves in the CCS flow path provides assurance that the proper flow paths exist for CCS 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 being mispositioned are in the correct position.

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BASES

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SURVEILLANCE REQUIREMENTS (continued)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.7.7.2

This SR verifies proper automatic operation of the CCS pumps on an actual or simulated (i.e, Safety Injection) actuation signal. The CCS is a normally operating system that cannot be fully actuated as part of routine testing during normal operation.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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REFERENCES      1. UFSAR, Section 9.2.1.

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## B 3.7 PLANT SYSTEMS

### B 3.7.8 Essential Raw Cooling Water (ERCW) System

#### BASES

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**BACKGROUND** The ERCW 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, and a normal shutdown, the ERCW system also provides this function for various safety related and nonsafety related components. The safety related function is covered by this LCO.

The ERCW system consists of two separate and independent, 100% capacity, safety related, cooling water trains. The water supply and distribution system is essentially common to both units. Two common trains feed both units. Each train consists of two main supply headers, two strainers, four pumps, two traveling water screens, and associated piping, valving, and instrumentation. To meet the design requirements, with the Ultimate Heat Sink (UHS) temperature at its limit, the system requires two main supply headers, two strainers, and two pumps sharing one traveling screen. The pumps and valves are remote and manually aligned, except in the unlikely event of a loss of coolant accident (LOCA). The pumps aligned to the critical loops and selected by the selector switch are automatically started upon receipt of a safety injection signal, and the essential valves are aligned to their post accident positions. Additionally, each emergency diesel generator has two assigned ERCW pumps. The two assigned ERCW pumps are interlocked so that only the selected pump will start if offsite power is lost.

Additional information about the design and operation of the ERCW system, along with a list of the components served, is presented in the UFSAR, Section 9.2.2 (Ref. 1). The principal safety related function of the ERCW system is the removal of decay heat from the reactor via the CCS.

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**APPLICABLE SAFETY ANALYSES** The design basis of the ERCW system is for one ERCW system train, in conjunction with the CCS and a 100% capacity containment cooling system, to remove core decay heat following a design basis LOCA as discussed in the UFSAR, 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 ECCS pumps. The ERCW system is designed to perform its function with a single failure of any active component, assuming the loss of offsite power.

The ERCW system, in conjunction with the CCS, also cools the unit from residual heat removal (RHR), as discussed in the UFSAR, Section 5.5.7,

BASES

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APPLICABLE SAFETY ANALYSES (continued)

(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 component cooling water and RHR System trains that are operating. One ERCW system train is sufficient to remove decay heat during subsequent operations in MODES 5 and 6. This assumes a maximum ERCW system temperature of 87°F occurring simultaneously with maximum heat loads on the system.

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

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LCO

Two ERCW system trains are required to be OPERABLE to provide the required redundancy to ensure that the system functions to remove post accident heat loads, assuming that the worst case single active failure occurs coincident with the loss of offsite power.

An ERCW system train is considered OPERABLE during MODES 1, 2, 3, and 4 when the required ERCW pumps are operable and the associated piping, valves, heat exchanger, and instrumentation and controls required to perform the safety related function as described in UFSAR Section 9.2.2.

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APPLICABILITY

In MODES 1, 2, 3, and 4, the ERCW system is a normally operating system that is required to support the OPERABILITY of the equipment serviced by the ERCW system and required to be OPERABLE in these MODES.

In MODES 5 and 6, the OPERABILITY requirements of the ERCW system are determined by the systems it supports.

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ACTIONS

A.1 and A.2

Condition A is modified by two Notes that limit the conditions and parameters that allow entry into Condition A. The first Note limits the applicability of Condition A to the time period when the opposite unit is either defueled or in MODE 6 following defueled with refueling water cavity level  $\geq 23$  ft. above the top of the reactor vessel flange. The second Note requires a temperature limitation on UHS Temperature. In

## BASES

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### ACTIONS (continued)

order to credit the temperature limit, the effected ERCW train must be aligned in accordance with UFSAR 9.2.2.2. This will allow the plant configuration to be aligned (i.e., cross-ties exist and isolation of loads to facilitate maintenance activities) to minimize the heat load on the ERCW system to ensure the ERCW system continues to meet its design function.

The 7 day Completion Time is acceptable based on the following:

- The low probability of a DBA occurring during that time;
- The heat load on the ERCW System is substantially lower than assumed for the DBA with the opposite unit defueled or subsequent to defueled; and
- The redundant capabilities afforded by the OPERABLE train.

If one ERCW system train is inoperable for planned maintenance, action must be taken to restore OPERABLE status within 7 days. In this Condition, the remaining OPERABLE ERCW system train is adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure in the OPERABLE ERCW system train could result in loss of ERCW system function.

Required Action A.1 is modified by two Notes. The first Note indicates that the applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources – Operating," should be entered if an inoperable ERCW system 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 ERCW system train results in an inoperable residual heat removal loop. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components.

Required Action A.2 ensures the credited temperature limit for Ultimate Heat Sink is maintained.

### B.1

If one ERCW system train is inoperable for reasons other than Condition A, action must be taken to restore OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE ERCW system train is adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure in the OPERABLE ERCW system train could result in loss of ERCW system function.

Required Action B.1 is modified by two Notes.

## BASES

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### ACTIONS (continued)

The first Note indicates that the applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources - Operating," should be entered if an inoperable ERCW system 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 ERCW system train results in an inoperable residual heat removal loop. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components. 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.

#### C.1 and C.2

If the ERCW system train cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours.

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

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### SURVEILLANCE REQUIREMENTS

#### SR 3.7.8.1

This SR is modified by a Note indicating that the isolation of the ERCW system components or systems may render those components inoperable, but does not affect the OPERABILITY of the ERCW system.

Verifying the correct alignment for manual, power operated, and automatic valves in the ERCW system flow path provides assurance that the proper flow paths exist for ERCW system 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 being locked, sealed, or secured. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.8.2

This SR verifies proper automatic operation of the ERCW system valves on an actual or simulated actuation signal. The Safety Injection signal is the automatic actuation signal. The ERCW system is a normally operating system that cannot be fully actuated as part of normal testing. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.7.8.3

This SR verifies proper automatic operation of the ERCW system pumps on an actual or simulated (i.e., Safety Injection) actuation signal. The ERCW system is a normally operating system that cannot be fully actuated as part of normal testing during normal operation.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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REFERENCES

1. UFSAR, Section 9.2.2.
  2. UFSAR, Section 6.2.
  3. UFSAR, Section 5.5.7.
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## B 3.7 PLANT SYSTEMS

### B 3.7.9 Ultimate Heat Sink (UHS)

#### BASES

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<b>BACKGROUND</b>	<p>The UHS provides a heat sink for processing and operating heat from safety related components during a transient or accident, as well as during normal operation. This is done by utilizing the Essential Raw Cooling Water (ERCW) system and the Component Cooling Water System (CCS).</p> <p>The UHS has been defined as that complex of water sources, including necessary retaining structures, and the canals or conduits connecting the sources with, but not including, the cooling water system intake structures as discussed in the UFSAR, Section 9.2.5 (Ref. 1). The two principal functions of the UHS are the dissipation of residual heat after reactor shutdown, and dissipation of residual heat after an accident.</p> <p>Chickamauga Lake (Tennessee River system) is used to meet the requirements for a UHS.</p> <p>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.</p> <p>Additional information on the design and operation of the system, along with a list of components served, can be found in Reference 1.</p>
<b>APPLICABLE SAFETY ANALYSES</b>	<p>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 residual heat removal (RHR) operation. Its maximum post accident heat load occurs approximately 25 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 and RHR are required to remove the core decay heat.</p> <p>The operating limits are based on conservative heat transfer analyses for the worst case LOCA. Reference 1 provides the details of the assumptions used in the analysis, which include worst expected meteorological conditions, conservative uncertainties when calculating decay heat, and worst case single active failure (e.g., single failure of a 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.</p> <p>The UHS satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).</p>

BASES

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LCO The UHS is required to be OPERABLE and is considered OPERABLE if it contains a sufficient volume of water at or below the maximum temperature that would allow the ERCW system 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 ERCW system. To meet this condition, the average ERCW supply header water temperature should not exceed 87°F and the level of the UHS should not fall below 674 ft mean sea level USGS datum during normal unit operation.

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APPLICABILITY In MODES 1, 2, 3, and 4, the UHS is required to support the OPERABILITY of the equipment serviced by the UHS and required to be OPERABLE in these MODES.

In MODE 5 or 6, the OPERABILITY requirements of the UHS are determined by the systems it supports.

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ACTIONS A.1 and A.2

If the UHS is 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.

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SURVEILLANCE REQUIREMENTS SR 3.7.9.1

This SR verifies that adequate long term (30 day) cooling can be maintained. The specified level also ensures that sufficient NPSH is available to operate the ERCW pumps. This SR verifies that the UHS water level is  $\geq$  674 ft mean sea level USGS datum.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.9.2

This SR verifies that the ERCW System is available to cool the CCS to at least its maximum design temperature with the maximum accident or normal design heat loads for 30 days following a Design Basis Accident. This SR verifies that the average ERCW supply header water temperature is  $\leq 87^{\circ}\text{F}$ .

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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- REFERENCES
1. UFSAR, Section 9.2.5.
  2. Regulatory Guide 1.27.
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## B 3.7 PLANT SYSTEMS

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

#### BASES

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**BACKGROUND** The CREVS provides a protected environment from which occupants can control the unit following an uncontrolled release of radioactivity, hazardous chemicals, or smoke.

The CREVS consists of two independent, redundant trains that recirculate and filter the air in the control room envelope (CRE) and a CRE boundary that limits the inleakage of unfiltered air. Each CREVS train consists of 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, doors, barriers, and instrumentation also form part of the system.

The CRE is the area within the confines of the CRE boundary that contains the spaces that control room occupants inhabit to control the unit during normal and accident conditions. The CRE is the area within Elevation 732 of the Control Building which encompasses the Main Control Room, Technical Support Center, Men's and Women's Locker rooms, Men's and Women's Bathrooms, Kitchen, and Relay Room. The CRE is protected during normal operation, natural events, and accident conditions. The CRE boundary is the combination of walls, floor, roof, ducting, doors, penetrations and equipment that physically form the CRE. The OPERABILITY of the CRE boundary must be maintained to ensure that the inleakage of unfiltered air into the CRE will not exceed the inleakage assumed in the licensing basis analysis of design basis accident (DBA) consequences to CRE occupants. The CRE and its boundary are defined in the Control Room Envelope Habitability Program.

The CREVS is an emergency system, parts of which may also operate during normal unit operations in the standby mode of operation.

Actuation of the CREVS places the system in the emergency mode of operation. Actuation of the system to the emergency radiation state of the emergency mode of operation, closes the unfiltered outside air intake and unfiltered exhaust dampers, and aligns the system for recirculation of the air within the CRE through the redundant trains of HEPA filters and the charcoal adsorbers. The emergency radiation state also initiates pressurization and filtered ventilation of the air supply to the CRE.

Outside air is filtered, and added to the air being recirculated from the CRE. Pressurization of the CRE minimizes infiltration of unfiltered air through the CRE boundary from all the surrounding areas adjacent to the CRE boundary.

BASES

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BACKGROUND (continued)

The air entering the CRE is continuously monitored by radiation monitors. One detector output above the setpoint will cause actuation of the emergency radiation state.

A single CREVS train operating at a flow rate of 4000 cfm ( $\pm 10\%$ ) will pressurize the CRE to 0.125 inches water gauge relative to the outside atmosphere. Additionally, CREVS maintains a slightly positive pressure relative to external areas adjacent to the CRE boundary. The CREVS operation in maintaining the CRE habitable is discussed in the UFSAR, Sections 6.4 and 9.4 (Refs. 1 and 2).

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 CREVS is designed in accordance with Seismic Category I requirements.

The CREVS is designed to maintain a habitable environment in the CRE for 30 days of continuous occupancy after a DBA without exceeding a 5 rem whole body dose or its equivalent to any part of the body.

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APPLICABLE  
SAFETY  
ANALYSES

The CREVS components are arranged in redundant, safety related ventilation trains. The location of components and ducting within the CRE ensures an adequate supply of filtered air to all areas requiring access. The CREVS provides airborne radiological protection for the CRE occupants, as demonstrated by the CRE occupant dose analyses for the most limiting design basis accident, fission product release presented in the UFSAR, Chapter 15 (Ref. 3).

The CREVS provides protection from smoke and hazardous chemicals to the CRE occupants. The analysis of hazardous chemical releases demonstrates that the toxicity limits are not exceeded in the CRE following a hazardous chemical release (Refs. 4 and 5). The evaluation of a smoke challenge demonstrates that it will not result in the inability of the CRE occupants to control the reactor either from the control room or from the remote shutdown panels (Refs. 2 and 4).

The worst case single active failure of a component of the CREVS, assuming a loss of offsite power, does not impair the ability of the system to perform its design function.

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

BASES

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LCO

Two independent and redundant CREVS trains are required to be OPERABLE to ensure that at least one is available if a single active failure disables the other train. Total system failure, such as from a loss of both ventilation trains or from an inoperable CRE boundary, could result in exceeding a dose of 5 rem whole body or its equivalent to any part of the body to the CRE occupants in the event of a large radioactive release.

Each CREVS train is considered OPERABLE when the individual components necessary to limit CRE occupant exposure are OPERABLE. A CREVS train is OPERABLE when the associated:

- a. Fan is OPERABLE;
- b. HEPA filters and charcoal adsorbers are not excessively restricting flow, and are capable of performing their filtration functions;
- c. Ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained; and
- d. Tornado dampers are de-activated and in the open position.

In order for the CREVS trains to be considered OPERABLE, the CRE boundary must be maintained such that the CRE occupant dose from a large radioactive release does not exceed the calculated dose in the licensing basis consequence analyses for DBAs, and that CRE occupants are protected from hazardous chemicals and smoke.

The LCO is modified by a Note allowing the CRE boundary to be opened intermittently under administrative controls. This Note only applies to openings in the CRE boundary that can be rapidly restored to the design condition, such as doors, hatches, floor plugs, and access panels. For entry and exit through doors, the administrative control of the opening is performed by the person(s) entering or exiting the area. For other openings, these controls should be proceduralized and consist of stationing a dedicated individual at the opening who is in continuous communication with the operators in the CRE. This individual will have a method to rapidly close the opening and to restore the CRE boundary to a condition equivalent to the design condition when a need for CRE isolation is indicated.

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APPLICABILITY

In MODES 1, 2, 3, 4, 5, and 6, and during movement of irradiated fuel assemblies, the CREVS must be OPERABLE to ensure that the CRE will remain habitable during and following a DBA.

During movement of irradiated fuel assemblies, the CREVS must be OPERABLE to cope with the release from a fuel handling accident.

BASES

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ACTIONS

A.1

When one CREVS train is inoperable, for reasons other than an inoperable CRE boundary, action must be taken to restore OPERABLE status within 7 days. In this Condition, the remaining OPERABLE CREVS train is adequate to perform the CRE occupant protection function. However, the overall reliability is reduced because a failure in the OPERABLE CREVS train could result in loss of CREVS function. The 7 day Completion Time is based on the low probability of a DBA occurring during this time period, and ability of the remaining train to provide the required capability.

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

If the unfiltered inleakage of potentially contaminated air past the CRE boundary and into the CRE can result in CRE occupant radiological dose greater than the calculated dose of the licensing basis analyses of DBA consequences (allowed to be up to 5 rem whole body or its equivalent to any part of the body), or inadequate protection of CRE occupants from hazardous chemicals or smoke, the CRE boundary is inoperable. Actions must be taken to restore an OPERABLE CRE boundary within 90 days.

During the period that the CRE boundary is considered inoperable, action must be initiated to implement mitigating actions to lessen the effect on CRE occupants from the potential hazards of a radiological or chemical event or a challenge from smoke. Actions must be taken within 24 hours to verify that in the event of a DBA, the mitigating actions will ensure that CRE occupant radiological exposures will not exceed the calculated dose of the licensing basis analyses of DBA consequences, and that CRE occupants are protected from hazardous chemicals and smoke. These mitigating actions (i.e., actions that are taken to offset the consequences of the inoperable CRE boundary) should be preplanned for implementation upon entry into the condition, regardless of whether entry is intentional or unintentional. The 24 hour Completion Time is reasonable based on the low probability of a DBA occurring during this time period, and the use of mitigating actions. The 90 day Completion Time is reasonable based on the determination that the mitigating actions will ensure protection of CRE occupants within analyzed limits while limiting the probability that CRE occupants will have to implement protective measures that may adversely affect their ability to control the reactor and maintain it in a safe shutdown condition in the event of a DBA. In addition, the 90 day Completion Time is a reasonable time to diagnose, plan and possibly repair, and test most problems with the CRE boundary.

BASES

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## ACTIONS (continued)

C.1

When both CREVS trains are inoperable due to the tornado dampers not in the correct position (i.e., open and de-activated) as a result of a tornado warning, action must be taken to restore at least one train of CREVS to OPERABLE status within 8 hours. In this condition, the shutdown of the operating unit would not be reasonable in consideration that the actions that created the inoperable condition were for the protection of the operating unit and would not be expected to last for a significant duration. The 8 hour Completion Time is reasonable based on the low probability of a DBA occurring during this time period, and high probability that the CREVS trains can be returned to OPERABLE status within 8 hours following the tornado warning.

D.1 and D.2

In MODE 1, 2, 3, or 4, if the inoperable CREVS train or the CRE boundary cannot be restored to OPERABLE status within the required Completion Time, the unit must be placed in a MODE that minimizes 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.

E.1 and E.2

In MODE 5 or 6, or during movement of irradiated fuel assemblies, if the inoperable CREVS train cannot be restored to OPERABLE status within the required Completion Time, action must be taken to immediately place the OPERABLE CREVS train in the recirculation mode. This action ensures that the remaining train is OPERABLE, that no failures preventing automatic actuation will occur, and that any active failure would be readily detected.

An alternative to Required Action E.1 is to immediately suspend activities that could result in a release of radioactivity that might require isolation of the CRE. This places the unit in a condition that minimizes the accident risk. This does not preclude the movement of fuel to a safe position.

BASES

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ACTIONS (continued)

F.1

In MODE 5 or 6, or during movement of irradiated fuel assemblies, with two CREVS trains inoperable or with one or more CREVS trains inoperable due to an inoperable CRE boundary, action must be taken immediately to suspend activities that could result in a release of radioactivity that might require isolation of the CRE. This places the unit in a condition that minimizes the accident risk. This does not preclude the movement of fuel to a safe position.

G.1

If both CREVS trains are inoperable in MODE 1, 2, 3, or 4 for reasons other than an inoperable CRE boundary or tornado dampers not in the correct position (i.e., Condition B or C), the CREVS may not be capable of performing the intended function and the unit is in a condition outside the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.

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SURVEILLANCE  
REQUIREMENTS

SR 3.7.10.1

Verifying the correct position of the tornado dampers in the CREVS flow paths provides assurance that the proper flow paths will exist for CREVS operation. This SR does not apply to tornado dampers that are locked, sealed, or otherwise secured in position, since these were verified to be in the correct position prior to locking, sealing, or securing. This Surveillance does not require any testing or damper manipulation. Rather, it involves verification that the tornado dampers are in the correct position (open and de-activated). The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.7.10.2

Standby systems should be checked periodically to ensure that they function properly. As the environment and normal operating conditions on this system are not too severe, testing each train once every month provides an adequate check of this system. Operation for  $\geq 15$  continuous minutes demonstrates OPERABILITY of the system. Periodic operation ensures that blockage, fan or motor failure, or excessive vibration can be detected for corrective action. The CREVS train OPERABILITY will be demonstrated by initiating, from the control room, flow through the HEPA filter and charcoal adsorber train.

BASES

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SURVEILLANCE REQUIREMENTS (continued)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.7.10.3

This SR verifies that the required CREVS testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The VFTP includes testing the performance of the HEPA filter, charcoal adsorber efficiency, minimum flow rate, and the physical properties of the activated charcoal. Specific test Frequencies and additional information are discussed in detail in the VFTP.

SR 3.7.10.4

This SR verifies that each CREVS train starts automatically, diverts its inlet flow through the HEPA filters and charcoal adsorbers, and operates on an actual or simulated (i.e., safety injection signal or a high radiation signal from the air intake stream) actuation signal.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.7.10.5

This SR verifies the OPERABILITY of the CRE boundary by testing for unfiltered air leakage past the CRE boundary and into the CRE. The details of the testing are specified in the Control Room Envelope Habitability Program.

The CRE is considered habitable when the radiological dose to CRE occupants calculated in the licensing basis analyses of DBA consequences is no more than 5 rem whole body or its equivalent to any part of the body and the CRE occupants are protected from hazardous chemicals and smoke. This SR verifies that the unfiltered air leakage into the CRE is no greater than the flow rate assumed in the licensing basis analyses of DBA consequences. When unfiltered air leakage is greater than the assumed flow rate, Condition B must be entered. Required Action B.3 allows time to restore the CRE boundary to OPERABLE status provided mitigating actions can ensure that the CRE remains within the licensing basis habitability limits for the occupants following an accident. Compensatory measures are discussed in Regulatory Guide 1.196, Section C.2.7.3, (Ref. 6) which endorses, with exceptions, NEI 99-03, Section 8.4 and Appendix F (Ref. 7). These

BASES

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SURVEILLANCE REQUIREMENTS (continued)

compensatory measures may also be used as mitigating actions as required by Required Action B.2. Temporary analytical methods may also be used as compensatory measures to restore OPERABILITY (Ref. 8). Options for restoring the CRE boundary to OPERABLE status include changing the licensing basis DBA consequence analysis, repairing the CRE boundary, or a combination of these actions. Depending upon the nature of the problem and the corrective action, a full scope inleakage test may not be necessary to establish that the CRE boundary has been restored to OPERABLE status.

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REFERENCES

1. UFSAR, Section 6.4.
  2. UFSAR, Section 9.4.
  3. UFSAR, Chapter 15.
  4. UFSAR, Section 2.2.
  5. UFSAR, Section 8.3.1.2.3.
  6. Regulatory Guide 1.196.
  7. NEI 99-03, "Control Room Habitability Assessment," June 2001.
  8. Letter from Eric J. Leeds (NRC) to James W. Davis (NEI) dated January 30, 2004, "NEI Draft White Paper, Use of Generic Letter 91-18 Process and Alternative Source Terms in the Context of Control Room Habitability," (ADAMS Accession No. ML040300694).
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## B 3.7 PLANT SYSTEMS

### B 3.7.11 Control Room Air-Conditioning System (CRACS)

#### BASES

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BACKGROUND	<p>The CRACS provides temperature control for the control room following isolation of the control room.</p> <p>The CRACS consists of two independent and redundant trains that provide cooling and heating of recirculated control room air. Each train consists of a chiller package, cooling coils, air handling unit, instrumentation, and controls to provide for control room temperature control. The CRACS is a subsystem providing air temperature control for the control room.</p> <p>The CRACS 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 at approximately 75°F. The CRACS operation in maintaining the control room temperature is discussed in the UFSAR, Section 9.4 (Ref. 1).</p>
APPLICABLE SAFETY ANALYSES	<p>The design basis of the CRACS is to maintain the control room temperature for 30 days of continuous occupancy.</p> <p>The CRACS components are arranged in redundant, safety related trains. During emergency operation, the CRACS maintains the temperature at approximately 75°F. In addition, the CRACS is designed to maintain the control room temperature at less than the maximum abnormal postulated temperature of 104°F. A single active failure of a component of the CRACS, with a loss of offsite power, does not impair the ability of the system to perform its design function. Redundant detectors and controls are provided for control room temperature control. The CRACS is designed in accordance with Seismic Category I requirements. The CRACS is capable of removing sensible and latent heat loads from the control room, which include consideration of equipment heat loads and personnel occupancy requirements, to ensure equipment OPERABILITY.</p> <p>The CRACS satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).</p>
LCO	<p>Two independent and redundant trains of the CRACS are required to be OPERABLE to ensure that at least one is available, assuming a single failure disabling the other train. Total system failure could result in the equipment operating temperature exceeding limits in the event of an accident.</p>

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BASES

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LCO (continued)

The CRACS is considered to be OPERABLE when the individual components necessary to maintain the control room temperature are OPERABLE in both trains. These components include the cooling coils, chiller package, air handling unit, and associated temperature control instrumentation. In addition, the CRACS must be OPERABLE to the extent that air circulation can be maintained.

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APPLICABILITY

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

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ACTIONS

A.1

With one CRACS train inoperable, action must be taken to restore OPERABLE status within 30 days. In this Condition, the remaining OPERABLE CRACS train is adequate to maintain the control room temperature within limits. However, the overall reliability is reduced because a single failure in the OPERABLE CRACS train could result in loss of CRACS function. The 30 day Completion Time is based on the low probability of an event requiring control room isolation, the consideration that the remaining train can provide the required protection, and that alternate safety or nonsafety related cooling means are available.

B.1 and B.2

In MODE 1, 2, 3, or 4, if the inoperable CRACS train cannot be restored to OPERABLE status within the required Completion Time, the unit must be placed in a MODE that minimizes the 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.

BASES

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ACTIONS (continued)

C.1 and C.2

In MODE 5 or 6, or during movement of irradiated fuel, if the inoperable CRACS train cannot be restored to OPERABLE status within the required Completion Time, the OPERABLE CRACS 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 active failures will be readily detected.

An alternative to Required Action C.1 is to immediately suspend activities that present a potential for releasing radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes accident risk. This does not preclude the movement of fuel to a safe position.

D.1

In MODE 5 or 6, or during movement of irradiated fuel assemblies, with two CRACS 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 risk. This does not preclude the movement of fuel to a safe position.

E.1

If both CRACS trains are inoperable in MODE 1, 2, 3, or 4, the control room CRACS may not be capable of performing its intended function. Therefore, LCO 3.0.3 must be entered immediately.

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SURVEILLANCE  
REQUIREMENTS

SR 3.7.11.1

This SR verifies that the heat removal capability of the system is sufficient to remove the heat load assumed in the safety analyses in the control room. This SR consists of a combination of testing and calculations.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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REFERENCES

1. UFSAR, Section 9.4.
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## B 3.7 PLANT SYSTEMS

### B 3.7.12 Auxiliary Building Gas Treatment System (ABGTS)

#### BASES

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<b>BACKGROUND</b>	<p>The ABGTS filters airborne radioactive particulates from the area of the fuel pool following a fuel handling accident or loss of coolant accident (LOCA).</p> <p>The ABGTS consists of two independent and redundant trains. Each train consists of a heater, a prefilter, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and a fan. Ductwork, valves or dampers, and instrumentation also form part of the system. The system initiates filtered ventilation of the auxiliary building following receipt of a high radiation signal from the fuel handling area radiation monitors, a high radiation signal from the train-specific Auxiliary Building exhaust vent monitor, a Phase A containment isolation signal from either reactor, or a high temperature signal from the Auxiliary Building air intakes. During plant operations with the containment open to the auxiliary building, the Auxiliary Building Secondary Containment Enclosure (ABSCE) boundary is extended to include the area inside the containment building and the shield building.</p> <p>The ABGTS is a standby system. Upon receipt of the actuating signal, normal air discharge from the auxiliary building is isolated and the stream of ventilation air discharges through the system filter trains.</p> <p>The ABGTS is discussed in the UFSAR, Sections 6.2.3, 15.5.3, and 15.5.6 (Refs. 1, 2 and 3, respectively).</p>
<b>APPLICABLE SAFETY ANALYSES</b>	<p>The ABGTS design basis is established by the consequences of Design Basis Accidents (DBAs), a LOCA during MODES 1, 2, 3 and 4, and a fuel handling accident during operations involving irradiated fuel assemblies. The analysis of the LOCA, given in Reference 2, assumes that radioactive materials leaked from the Emergency Core Cooling System (ECCS) are filtered and adsorbed by the ABGTS. The analysis of the fuel handling accident, given in Reference 3, assumes that the auxiliary building secondary containment enclosure (ABSCE) boundary is capable of being established to ensure the releases from the auxiliary and containment buildings are consistent with the dose consequence analysis, no credit is taken for filtration by the ABGTS.</p> <p>The amount of fission products available for release from the auxiliary building is determined for a fuel handling accident and for a LOCA. These assumptions and the analysis follow the guidance provided in Regulatory Guide 1.4 (Ref. 4) for a LOCA and Regulatory Guide 1.183 (Ref. 5) for the fuel handling accident.</p>

BASES

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APPLICABLE SAFETY ANALYSIS (continued)

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

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LCO

Two independent and redundant trains of the ABGTS are required to be OPERABLE to ensure that at least one train is available, assuming a single failure that disables the other train, coincident with a loss of offsite power. Total system failure could result in the atmospheric release from the auxiliary building exceeding the 10 CFR 100 (Ref. 6) limits in the event of a LOCA.

One train of the ABGTS is required to be OPERABLE to mitigate the consequences of a fuel handling accident involving handling irradiated fuel to limit releases to the environment to within the 10 CFR 50.67 limits.

The ABGTS is considered OPERABLE when the individual components necessary to control exposure in the auxiliary building are OPERABLE in both trains. An ABGTS train is considered OPERABLE when its associated:

- a. Fan is OPERABLE,
- b. HEPA filter and charcoal adsorber are not excessively restricting flow, and are capable of performing their filtration function, and
- c. Heater, ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained.

The LCO is modified by a Note that specifies that only one ABGTS train is required to be OPERABLE during the movement of irradiated fuel assemblies or with fuel stored in the spent fuel pool.

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APPLICABILITY

In MODE 1, 2, 3, or 4, the ABGTS is required to be OPERABLE to provide fission product removal associated with ECCS leaks due to a LOCA and leakage from containment and annulus.

In MODE 5 or 6, the ABGTS is not required to be OPERABLE to provide fission product removal associated with ECCS leaks due to a LOCA since the ECCS is not required to be OPERABLE.

During movement of irradiated fuel, one train of ABGTS is required to be OPERABLE to alleviate the consequences of a fuel handling accident. With fuel stored in the spent fuel pool, one train of ABGTS is required to be OPERABLE to support the mitigation of any potential fuel damage resulting from a load drop.

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## BASES

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### ACTIONS

LCO 3.0.3 is not applicable while in MODE 5 or 6. However, since crane travel with loads over fuel stored in the spent fuel pool and irradiated fuel assembly movement can occur in MODE 1, 2, 3, or 4, the ACTIONS have been modified by a Note stating that LCO 3.0.3 is not applicable. If storing or moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If storing or moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement or storage is independent of reactor operations. Entering LCO 3.0.3, while in MODE 1, 2, 3, or 4 would require the unit to be shutdown unnecessarily.

#### A.1

With one ABGTS train inoperable in MODE 1, 2, 3, or 4, action must be taken to restore OPERABLE status within 7 days. During this period, the remaining OPERABLE train is adequate to perform the ABGTS function. The 7 day Completion Time is based on the risk from an event occurring requiring the inoperable ABGTS train, and the remaining ABGTS train providing the required protection.

#### B.1

If the ABSCE boundary is inoperable in MODE 1, 2, 3, or 4, the ABGTS trains cannot perform their intended functions. Actions must be taken to restore an OPERABLE ABSCE boundary within 24 hours. During the period that the ABSCE boundary is inoperable, appropriate compensatory measures consistent with the intent, as applicable, of GDC 19, 60, 61, 63, 64 and 10 CFR Part 100 should be utilized to protect plant personnel from potential hazards such as radioactive contamination, toxic chemicals, smoke, temperature and relative humidity, and physical security. Preplanned measures should be available to address these concerns for intentional and unintentional entry into the condition. The 24 hour Completion Time is reasonable based on the low probability of a DBA occurring during this time period, and the use of compensatory measures. The 24 hour Completion Time is a typically reasonable time to diagnose, plan and possibly repair, and test most problems with the ABSCE boundary.

BASES

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ACTIONS (continued)

C.1 and C.2

In MODE 1, 2, 3, or 4, when Required Action A.1 or B.1 cannot be completed within the associated Completion Time, or when both ABGTS trains are inoperable for reasons other than an inoperable ABSCE boundary (i.e., Condition B), the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in MODE 3 within 6 hours and in MODE 5 within 36 hours. The Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

D.1

When the required train of ABGTS is inoperable during movement of irradiated fuel assemblies, action must be taken to place the unit in a condition in which the LCO does not apply. Action must be taken immediately to suspend movement of irradiated fuel assemblies. This does not preclude the movement of fuel to a safe position.

E.1

When the required train of ABGTS is inoperable with fuel stored in the spent fuel pool, action must be taken to prevent the possibility of a load drop over fuel stored in the spent fuel pool. Suspending all crane operation with loads over the spent fuel pool will eliminate the possibility of dropping a load onto fuel assemblies stored in the spent fuel pool.

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SURVEILLANCE  
REQUIREMENTS

SR 3.7.12.1

Standby systems should be checked periodically to ensure that they function properly. As the environmental and normal operating conditions on this system are not severe, testing each train once every month provides an adequate check on this system.

Operation with heaters on for  $\geq 15$  continuous minutes demonstrates OPERABILITY of the system. Periodic operation ensures that heater failure, blockage, fan or motor failure, or excessive vibration can be detected for corrective action. Operation will be demonstrated by initiating, from the control room, flow through the HEPA filter and charcoal adsorber train.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.12.2

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

SR 3.7.12.3

This SR verifies that each ABGTS train starts and operates on an actual or simulated actuation signal. The SR is modified by two Notes that specify when verification of ABGTS actuation for each actuation signal is required to be met. ABGTS actuation on a Containment Phase A isolation signal is required to be met in MODES 1, 2, 3 and 4. ABGTS actuation on fuel storage pool area high radiation signal is required to be met during movement of irradiated fuel assemblies and with fuel stored in the spent fuel pool.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.7.12.4

This SR verifies the integrity of the auxiliary building enclosure (i.e., spent fuel storage area and the ESF pump rooms). The ability of the auxiliary building to maintain negative pressure with respect to potentially uncontaminated adjacent areas is periodically tested to verify proper function of the ABGTS. During the post accident mode of operation, the ABGTS is designed to maintain a slight negative pressure in the auxiliary building, to prevent unfiltered LEAKAGE. The ABGTS is designed to maintain a pressure  $\leq -0.25$  inches water gauge with respect to atmospheric pressure at a flow rate  $\geq 8,100$  and  $\leq 9,900$  cfm to the auxiliary building.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

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- REFERENCES
1. UFSAR, Section 6.2.3.
  2. UFSAR, Section 15.5.3.
  3. UFSAR, Section 15.5.6.
  4. Regulatory Guide 1.4.
  5. Regulatory Guide 1.183.
  6. 10 CFR 100.
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## B 3.7 PLANT SYSTEMS

### B 3.7.13 Spent Fuel Pool Water Level

#### BASES

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**BACKGROUND** The minimum water level in the spent fuel 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 spent fuel pool design is given in the UFSAR, Section 9.1.2 (Ref. 1). A description of the Spent Fuel Pool Cooling and Cleanup System is given in the UFSAR, Section 9.1.3 (Ref. 2). The assumptions of the fuel handling accident are given in the UFSAR, Section 15.5.6 (Ref. 3).

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**APPLICABLE SAFETY ANALYSES** The minimum water level in the spent fuel pool meets the assumptions of the fuel handling accident described in Regulatory Guide 1.183 (Ref. 4). The resultant 2 hour thyroid dose per person at the exclusion area boundary is a small fraction of the 10 CFR 50.67 (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 during a fuel handling accident. With 23 ft of water, 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.

The spent fuel pool water level satisfies Criteria 2 and 3 of 10 CFR 50.36(c)(2)(ii).

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**LCO** The spent fuel pool water level is required to be  $\geq 23$  ft over the top of irradiated fuel assemblies seated in the storage racks. 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 spent fuel pool.

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**APPLICABILITY** This LCO applies whenever irradiated fuel assemblies are in the spent fuel pool, since the potential for a release of fission products exists.

BASES

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ACTIONS

A.1 and A.2

The Required Actions are modified by a Note indicating that LCO 3.0.3 does not apply.

When the initial conditions for prevention of an accident cannot be met, steps should be taken to preclude the accident from occurring. When the spent fuel pool water level is lower than the required level, the movement of fuel assemblies and crane operations with loads in the fuel storage areas is immediately suspended. This action effectively precludes the occurrence of a fuel handling accident. This does not preclude movement of a fuel assembly or crane load to a safe position.

If the spent fuel pool water level is not within the limit while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If the spent fuel pool water level is not within the limit while in MODES 1, 2, 3, and 4, the Required Actions are independent of reactor operations. Therefore, inability to suspend movement of fuel assemblies, suspend crane operations with loads, or restore spent fuel pool water level to within the limit is not sufficient reason to require a reactor shutdown.

The design basis fuel handling accident assumes the drop and damage of an irradiated fuel assembly; however, there are other potential failure mechanisms of the irradiated fuel in the spent fuel pool that could result in the release of fission product gases. As a result, with the spent fuel pool water level less than 23 feet above the top of irradiated fuel assemblies seated in storage racks, the iodine decontamination factor assumption in the design basis fuel handling accident analysis cannot be met.

Required Action A.2 requires the restoration of the spent fuel pool water level to the minimum required level to preserve the assumptions of the fuel handling accident analysis (Ref. 3). The Completion Time of 4 hours is considered sufficient to correct minor problems and restore the water level.

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SURVEILLANCE  
REQUIREMENTS

SR 3.7.13.1

This SR verifies sufficient spent fuel pool water is available in the event of a fuel handling accident. The water level in the spent fuel pool must be checked periodically.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

During refueling operations, the level in the spent fuel pool is in equilibrium with the refueling canal, and the level in the refueling canal is checked daily in accordance with SR 3.9.7.1.

BASES

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- REFERENCES
1. UFSAR, Section 9.1.2.
  2. UFSAR, Section 9.1.3.
  3. UFSAR, Section 15.5.6.
  4. Regulatory Guide 1.183, Rev. 0.
  5. 10 CFR 50.67.
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## B 3.7 PLANT SYSTEMS

### B 3.7.14 Spent Fuel Pool Boron Concentration

#### BASES

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**BACKGROUND** The spent fuel racks have been analyzed in accordance with the Holtec International methodology contained in Holtec Report HI - 992349 (Ref. 1). This methodology ensures that the spent fuel rack multiplication factor,  $k_{\text{eff}}$  is less than or equal to 0.95, as recommended by the NRC guidance contained in NRC Letter to All Power Reactor Licensees from B.K. Grimes, "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," April 14, 1978 and USNRC Internal Memorandum from L. Kopp, "Guidance On The Regulatory Requirements For Criticality Analysis Of Fuel Storage At Light-Water Reactor Power Plants," August 19, 1998 (Refs. 2 and 3). The codes, methods, and techniques contained in the methodology are used to satisfy the  $k_{\text{eff}}$  criterion. The spent fuel storage racks were analyzed using Westinghouse 17x17 Vantage 5H (V5H) fuel assemblies, with enrichments up to 4.95 ( $\pm 0.05$ ) wt% U-235 utilizing credit for checkerboarding, burnup, soluble boron, integral fuel burnable absorbers, gadolinia, and cooling time to ensure that  $k_{\text{eff}}$  is maintained less than or equal to 0.95, including uncertainties, tolerances, and accident conditions. In addition, the Spent Fuel Pool  $k_{\text{eff}}$  is maintained  $< 1.0$ , including uncertainties, tolerances on a 95/95 basis without any soluble boron. Calculations were performed to evaluate the reactivity of fuel types used at SQN. The results show that the Westinghouse 17x17 V5H fuel assembly exhibits the highest reactivity, thereby bounding all fuel types utilized and stored at SQN.

In the high density Spent Fuel Storage Rack design, the spent fuel pool is divided into three separate and distinct regions which, for the purpose of criticality considerations, are considered as separate pools. Region 1 is designed to accommodate new fuel with a maximum enrichment of 4.95 ( $\pm 0.05$ ) wt% U-235, or spent fuel regardless of the discharge fuel burnup in a 1-in-4 checkerboard arrangement of 1 fresh assembly with 3 spent fuel assemblies with specified enrichment, burnup and cooling times. Region 2 is designed to accommodate fuel which has 4.95 ( $\pm 0.05$ ) wt% initial enrichment burned to at least 30.27 megawatt days per kilogram uranium (MWD/KgU) (assembly average), or fuel of other enrichment with a burnup yielding an equivalent reactivity in the fuel racks. Region 3 is designed to accommodate fuel of 4.95 ( $\pm 0.05$ ) wt% initial enrichment or fuel assemblies of any lower reactivity in a 2-out-of-4 checkerboard arrangement with water-filled cells. Fuel assemblies shall be stored in accordance with LCO 3.7.15, "Spent Fuel Pool Storage."

The water in the spent fuel pool normally contains soluble boron, which results in large subcriticality margins under actual operating conditions. However, the NRC guidelines, based upon the accident condition in

BASES

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BACKGROUND (continued)

which all soluble poison is assumed to have been lost, specify that the limiting  $k_{\text{eff}}$  of  $< 1.0$  be evaluated in the absence of soluble boron. Hence, the design of each region is based on the use of unborated water, which maintains each region in a subcritical condition during normal operation with the regions fully loaded. The double contingency principle discussed in ANSI N16.1-1975 and the April 1978 NRC letter (Ref. 4) allows credit for soluble boron under other abnormal or accident conditions, since only a single accident need be considered at one time. For example, the most severe accident scenario is associated with the accidental mishandling of a fresh fuel assembly face adjacent to a fresh fuel assembly of Region 3. This could potentially increase the criticality of Region 3. To mitigate these postulated criticality related accidents, boron is dissolved in the pool water. The soluble boron concentration required to maintain  $k_{\text{eff}} \leq 0.95$  under normal conditions is 300 ppm and 700 ppm under the most severe postulated fuel mis-location accident. Safe operation of the spent fuel storage racks with no movement of assemblies may therefore be achieved by controlling the location of each assembly in accordance with LCO 3.7.15. Prior to movement of an assembly, it is necessary to perform SR 3.7.14.1.

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APPLICABLE  
SAFETY  
ANALYSES

Most accident conditions do not result in an increase in the activity of any of the regions. Examples of these accident conditions are the loss of cooling (reactivity increase with decreasing water density) and the dropping of a fuel assembly on the top of the rack. However, accidents can be postulated that could increase the reactivity. This increase in reactivity is unacceptable with unborated water in the storage pool. Thus, for these accident occurrences, the presence of soluble boron in the spent fuel pool prevents criticality in each region. The most limiting postulated accident with respect to the storage configurations assumed in the spent fuel rack criticality analysis is the misplacement of a nominal  $4.95 (\pm 0.05)$  wt% U-235 fuel assembly into an empty storage cell location in the Region 3 checkerboard storage arrangement.

The amount of soluble boron required to maintain  $k_{\text{eff}} \leq 0.95$  due to either fuel misload accident is 700 ppm (Ref. 1).

A spent fuel boron dilution analysis was performed to ensure that sufficient time is available to detect and mitigate dilution of the spent fuel pool prior to exceeding the  $k_{\text{eff}}$  design basis limit of 0.95 (Ref. 5). The spent fuel pool boron dilution analysis concluded that an inadvertent or unplanned event that would result in a dilution of the spent fuel pool boron concentration from 2000 ppm to 700 ppm is not a credible event. The accident analyses are provided in the UFSAR, Section 4.3.2.7 (Ref. 6).

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BASES

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APPLICABLE SAFETY ANALYSES (continued)

The concentration of dissolved boron in the spent fuel pool satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

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LCO The spent fuel pool boron concentration is required to be  $\geq 2000$  ppm. The specified concentration of dissolved boron in the spent fuel pool preserves the assumptions used in the analyses of the potential critical accident scenarios as described in Reference 6. This concentration of dissolved boron is the minimum required concentration for fuel assembly storage and movement within the spent fuel pool.

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

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ACTIONS

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

The Required Actions are modified by a Note indicating that LCO 3.0.3 does not apply.

When the concentration of boron in the spent fuel pool is less than required, immediate action must be taken to preclude the occurrence of an accident or to mitigate the consequences of an accident in progress. This is most efficiently achieved by immediately suspending the movement of fuel assemblies. The concentration of boron is restored simultaneously with suspending movement of fuel assemblies. Alternatively, beginning a verification of the spent fuel pool fuel locations, to ensure proper locations of the fuel, can be performed. However, prior to resuming movement of fuel assemblies, the concentration of boron must be restored. This does not preclude movement of a fuel assembly to a safe position.

If the LCO is not met while moving irradiated fuel assemblies in MODE 5 or 6, LCO 3.0.3 would not be applicable. 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.

BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.7.14.1

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 accidents are fully addressed.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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REFERENCES

1. Stanely E. Turner (Holtec International), "Criticality Safety Analyses of Sequoyah Spent Fuel Racks with Alternative Arrangements," HI-992349.
  2. B.K. Grimes (NRC GL78011), "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," April 14, 1978.
  3. L. Kopp, "Guidance On The Regulatory Requirements For Criticality Analysis Of Fuel Storage At Light-Water Reactor Power Plants," August 19, 1998.
  4. Double contingency principle of ANSI N16.1-1975, as specified in the April 14, 1978 NRC letter (Section 1.2) and implied in the proposed revision to Regulatory Guide 1.13 (Section 1.4, Appendix A).
  5. K K Niyogi (Holtec International), "Boron Dilution Analysis," HI-992302.
  6. UFSAR, Section 4.3.2.7.
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## B 3.7 PLANT SYSTEMS

### B 3.7.15 Spent Fuel Pool Storage

#### BASES

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**BACKGROUND** In the high density Spent Fuel Storage Rack design, the spent fuel storage pool is divided into three separate and distinct regions which, for the purpose of criticality considerations, are considered as separate pools. Region 1 is designed to accommodate new fuel with a maximum enrichment of 4.95 ( $\pm 0.05$ ) wt% U-235, or spent fuel regardless of the discharge fuel burnup in a 1-in-4 checkerboard arrangement of 1 fresh assembly with 3 spent fuel assemblies with specified enrichment, burnup and cooling times. Region 2 is designed to accommodate fuel of 4.95 ( $\pm 0.05$ ) wt% initial enrichment burned to at least 30.27 megawatt days per kilo gram uranium (MWD/KgU) (assembly average), or fuel of other enrichment with a burnup yielding an equivalent reactivity in the fuel racks. Region 3 is designed to accommodate fuel of 4.95 ( $\pm 0.05$ ) wt% initial enrichment or fuel assemblies of any lower reactivity in a 2-out-of-4 checkerboard arrangement with water-filled cells.

The water in the spent fuel storage pool normally contains soluble boron, which results in large subcriticality margins under actual operating conditions. However, the NRC guidelines, based upon the accident condition in which all soluble poison is assumed to have been lost, specify that the limiting  $k_{\text{eff}}$  of 0.95 be evaluated in the absence of soluble boron. Hence, the design of the regions is based on the use of unborated water, which maintains each region in a subcritical condition during normal operation with the regions fully loaded. The double contingency principle discussed in ANSI N-16.1-1975 and the April 1978 NRC letter (Ref. 1) allows credit for soluble boron under other abnormal or accident conditions, since only a single accident need be considered at one time. For example, the most severe accident scenario is associated with the accidental mishandling of a fresh fuel assembly face adjacent to a fresh fuel assembly of Region 3. This could potentially increase the criticality of Region 3. To mitigate these postulated criticality related accidents, boron is dissolved in the pool water. The soluble boron concentration required to maintain  $k_{\text{eff}} \leq 0.95$  under normal conditions is 300 ppm and 700 ppm under the most severe postulated fuel mis-location accident. Safe operation of the spent fuel storage racks may therefore be achieved by controlling the location of each assembly in accordance with the accompanying LCO. Prior to movement of an assembly, it is necessary to perform SR 3.7.14.1.

BASES

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APPLICABLE  
SAFETY  
ANALYSES

Most accident conditions do not result in an increase in the reactivity of any one of the three regions (Ref. 2). Examples of these accident conditions are the loss of cooling and the dropping of a fuel assembly on the top of the rack. However, accidents can be postulated that could increase the reactivity. This increase in reactivity is unacceptable with unborated water in the storage pool. Thus, for these accident occurrences, the presence of soluble boron in the storage pool prevents criticality in all regions. The most limiting postulated accident with respect to the storage configurations assumed in the spent fuel rack criticality analysis is the misplacement of a nominal 4.95 ( $\pm 0.05$ ) wt% U-235 fuel assembly into an empty storage cell location in the Region 3 checkerboard storage arrangement. The amount of soluble boron required to maintain  $k_{\text{eff}} \leq 0.95$  due to this fuel misload accident is 700 ppm.

The configuration of fuel assemblies in the fuel storage pool satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

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LCO

The restrictions on the placement of fuel assemblies within the spent fuel pool, in accordance with Figures 3.7.15-1 through 3.7.15-4 and Tables 3.7.15-1 through 3.7.15-3, in the accompanying LCO, ensures the  $k_{\text{eff}}$  of the spent fuel storage pool will always remain  $\leq 0.95$ , assuming the pool to be flooded with unborated water.

The arrangements in the spent fuel storage pool have the following definitions:

Region 1 is designed to accommodate new fuel with a maximum enrichment of 4.95 ( $\pm 0.05$ ) wt% U-235, (or spent fuel regardless of the fuel burnup), in a 1-in-4 checkerboard arrangement of 1 fresh assembly with 3 spent fuel assemblies with enrichment-burnup and cooling times illustrated in Figure 3.7.15-3 and defined by the equations in Table 3.7.15-1. Cooling time is defined as the period since reactor shutdown at the end of the last operating cycle for the discharged spent fuel assembly. The presence of a removable, non-fissile insert such as a burnable poison rod assembly (BPRA) or either gadolinia or integral fuel burnable absorber (IFBA) in a fresh fuel assembly does not affect the applicability of Figure 3.7.15-3 or Table 3.7.15-1.

Two alternative storage arrays (or sub-arrays) are acceptable in Region 1 if the fresh fuel assemblies contain rods with either gadolinia or IFBA. For these types of assemblies, the minimum burnup of the spent fuel in the 1-of-4 sub-array are defined by the equations in Table 3.7.15-2.

BASES

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LCO (continued)

For Region 1, any of the three sub-arrays illustrated in Figure 3.7.15-2 may be used in any combination provided that:

- a. Each sub-array of 4 fuel assemblies includes, in addition to the fresh fuel assembly, 3 assemblies with enrichment and minimum burnup requirements defined by the equations in Tables 3.7.15-1 and 3.7.15-2, as appropriate.
- b. The arrangement of Region 1 sub-arrays must not allow a configuration with fresh assemblies adjacent to each other.
- c. If Region 1 arrays are used in conjunction with Region 2 or Region 3 arrangements (see below), the arrangements shall not allow fresh fuel assemblies to be adjacent to each other (see also Figure 3.7.15-1).

Region 2 is designed to accommodate fuel of 4.95 ( $\pm$  0.05) wt% U-235 initial enrichment burned to at least 30.27 MWD/KgU (assembly average), or fuel of other enrichments with a burnup yielding an equivalent reactivity in the fuel racks. The minimum required assembly average burnup in MWD/KgU and cooling time is given by the equations in Table 3.7.15-3 in terms of E, where E is the initial enrichment in the axial zone of highest enrichment (wt% U-235). The minimum required burnups are illustrated in Figure 3.7.15-4 in terms of the initial enrichment and cooling time.

The following restrictions apply to the storage of spent fuel in the Region 2 cells:

- a. The spent fuel shall conform to the minimum burnup requirements defined by the equations in Table 3.7.15-3. Linear interpolation between cooling times may be made if desired.
- b. For the interface with Region 1 storage cells, fresh fuel in Region 1 shall not be stored adjacent to spent fuel assemblies in the Region 2 storage cells.

Region 3 is designed to accommodate fuel of 4.95 ( $\pm$  0.05) wt% U-235 initial enrichment (or fuel assemblies of any lower reactivity) in a 2-out-of-4 checkerboard arrangement with water-filled cells. The water-filled cells shall not contain any components bearing any fissile material, but may accommodate miscellaneous items or equipment.

BASES

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LCO (continued)

The following restrictions apply to the storage of spent fuel in the Region 3 cells:

- a. For the interface between Region 1 and Region 3 storage regions, fresh fuel assemblies shall not be stored adjacent to each other.
- b. If miscellaneous items or equipment are stored in the water cells of Region 3, the total volume of the miscellaneous items shall be no more than 75% of the storage cell volume.
- c. No fuel rods, assemblies, or items containing fissile material shall be stored in the water cells of Region 3.

An empty cell is less reactive than any cell containing fuel and therefore may be used as a Region 1, Region 2, or Region 3 cell in any arrangement.

Region 2 array described above may be used in the 15 x 15 storage rack module in the cask loading area of the cask pit.

A nominal concentration of 2000 ppm boron in the pool water. This concentration of soluble boron provides a margin sufficient to allow timely detection of a boron dilution accident and corrective action before the minimum concentration (700 ppm) required to protect against the most severe postulated fuel handling accident or before the minimum concentration (300 ppm) required to maintain the storage configuration design basis ( $k_{eff}$  less than 0.95) is reached.

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APPLICABILITY	This LCO applies whenever any fuel assembly is stored in Regions 1 through 3 of the fuel storage pool.
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ACTIONS	<p><u>A.1</u></p> <p>Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply.</p> <p>When the configuration of fuel assemblies stored in Regions 1 through 3 of the spent fuel storage pool is not in accordance with Figures 3.7.15-1 through 3.7.15-4 and Tables 3.7.15-1 through 3.7.15-3, the immediate action is to initiate action to make the necessary fuel assembly movement(s) to bring the configuration into compliance with Figures 3.7.15-1 through 3.7.15-4 and Tables 3.7.15-1 through 3.7.15-3.</p> <p>If unable to move irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not be applicable. If unable to move irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the action is independent of</p>
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BASES

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ACTIONS (continued)

reactor operation. Therefore, inability to move fuel assemblies is not sufficient reason to require a reactor shutdown.

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SURVEILLANCE  
REQUIREMENTS

SR 3.7.15.1

This SR verifies by administrative means that the initial enrichment and burnup of the fuel assembly is in accordance with Figures 3.7.15-1 through 3.7.15-4 and Tables 3.7.15-1 through 3.7.15.3, in the accompanying LCO.

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REFERENCES

1. Double contingency principle of ANSI N16.1-1975, as specified in the April 14, 1978 NRC letter (Section 1.2) and implied in the proposed revision to Regulatory Guide 1.13 (Section 1.4, Appendix A).
  2. UFSAR, Section 4.3.2.7.
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## B 3.7 PLANT SYSTEMS

### B 3.7.16 Secondary Specific Activity

#### BASES

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##### 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, indicates current conditions. During transients, I-131 spikes have been observed as well as increased releases of some noble gases. Other fission product isotopes, as well as activated corrosion products in lesser amounts, may also be found in the secondary coolant.

A limit on secondary coolant specific activity during power operation minimizes releases to the 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 0.35  $\mu\text{Ci/gm}$  (LCO 3.4.16, "RCS Specific Activity"). The steam line failure is assumed to result in the release of the noble gas and iodine activity contained in the steam generator inventory, the feedwater, and the reactor coolant LEAKAGE. Most of the iodine isotopes have short half lives (i.e., < 20 hours).

With the specified activity limit, the resultant 8 hour thyroid dose to a person at the exclusion area boundary (EAB) would be about 5.4 rem for 8 hours following a trip from full power, with a steam line break and a loss of AC power to plant auxiliaries.

Operating a unit at the allowable limits could result in a 8 hour EAB exposure of a small fraction of the 10 CFR 100 (Ref. 1) limits, or the limits established as the NRC staff approved licensing basis.

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##### APPLICABLE SAFETY ANALYSES

The accident analysis of the main steam line break (MSLB), as discussed in the UFSAR, Chapter 15 (Ref. 2) assumes the initial secondary coolant specific activity to have a radioactive isotope concentration of 0.10  $\mu\text{Ci/gm}$  DOSE EQUIVALENT I-131. This assumption is used in the analysis for determining the radiological consequences of the postulated accident. The accident analysis, based on this and other assumptions, shows that the radiological consequences of an MSLB do not exceed 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 generators are available for core decay heat dissipation by venting steam to the atmosphere through the MSSVs and steam generator atmospheric relief valves (ARVs). The Auxiliary Feedwater System supplies the necessary

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BASES

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APPLICABLE SAFETY ANALYSES (continued)

makeup to the steam generators. Venting continues until the reactor coolant temperature and pressure have decreased sufficiently for the Residual Heat Removal System to complete the cooldown.

In the evaluation of the radiological consequences of this accident, the activity released from the steam generator connected to the failed steam line is assumed to be released directly to the environment. The unaffected steam generators are assumed to discharge steam and any entrained activity through the MSSVs and ARVs during the event. Since no credit is taken in the analysis for activity plateout or retention, the resultant radiological consequences represent a conservative estimate of the potential integrated dose due to the postulated steam line failure.

Secondary specific activity limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

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LCO

As indicated in the Applicable Safety Analyses, the specific activity of the secondary coolant is required to be  $\leq 0.10 \mu\text{Ci/gm DOSE EQUIVALENT I-131}$  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.

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APPLICABILITY

In MODES 1, 2, 3, and 4, the limits on secondary specific activity apply due to the potential for secondary steam releases to the atmosphere.

In MODES 5 and 6, the steam generators are not being used for heat removal. Both the RCS and steam generators are depressurized, and primary to secondary LEAKAGE is minimal. Therefore, monitoring of secondary specific activity is not required.

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ACTIONS

A.1 and A.2

DOSE EQUIVALENT I-131 exceeding the allowable value in the secondary coolant, is an indication of a problem in the RCS and contributes to increased post accident doses. If the secondary specific activity cannot be restored to within limits within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

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BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.7.16.1

This SR verifies that the secondary specific activity is within the limits of the accident analysis. A gamma isotopic analysis of the secondary coolant, which determines DOSE EQUIVALENT I-131, confirms the validity of the safety analysis assumptions as 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 Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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REFERENCES

1. 10 CFR 100.11.
  2. UFSAR, Chapter 15.
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## B 3.7 PLANT SYSTEMS

### B 3.7.17 Cask Pit Pool Boron Concentration

#### BASES

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**BACKGROUND** The cask pit pool consists of a deep pool with adjacent shelf area. The cask pit is intended to be used for spent fuel shipment activities. High density spent fuel storage racks have been approved for addition and use in the cask loading area of the cask pit (Ref. 1). The 15 x 15 module could store 225 fuel assemblies and is designed to maintain stored fuel having an initial enrichment of up to 5 wt% U-235, in a safe, coolable, and sub-critical configuration during normal discharge, full core offload storages and postulated accident conditions.

A description of the spent fuel rack analysis is provided in the Bases for LCO 3.7.14, "Spent Fuel Pool Boron Concentration." Fuel assemblies shall be stored in accordance with LCO 3.7.15, "Spent Fuel Pool Storage." As described in the Bases for LCO 3.7.15, Region 2 arrays may be used in the 15 x 15 storage rack module in the cask loading area of the cask pit.

The water in the cask pit pool normally contains soluble boron, which results in large subcriticality margins under actual operating conditions. However, the NRC guidelines, based upon the accident condition in which all soluble poison is assumed to have been lost, specify that the limiting  $k_{\text{eff}}$  of  $< 1.0$  be evaluated in the absence of soluble boron. Hence, the design of each region is based on the use of unborated water, which maintains each region in a subcritical condition during normal operation with the regions fully loaded. The double contingency principle discussed in ANSI N-16.1-1975 and the April 1978 NRC letter (Ref. 2) allows credit for soluble boron under other abnormal or accident conditions, since only a single accident need be considered at one time. For example, the most severe accident scenario is associated with the accidental mishandling of a fresh fuel assembly face adjacent to a fresh fuel assembly of Region 3 in the spent fuel pool. This could potentially increase the criticality of Region 3. To mitigate these postulated criticality related accidents, boron is dissolved in the pool water. The soluble boron concentration required to maintain  $k_{\text{eff}} \leq 0.95$  under normal conditions is 300 ppm and 700 ppm under the most severe postulated fuel mis-location accident. Safe operation of the spent fuel storage racks with no movement of assemblies may therefore be achieved by controlling the location of each assembly in accordance with LCO 3.7.15. Prior to movement of an assembly, it is necessary to perform SR 3.7.17.1.

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**APPLICABLE SAFETY ANALYSES** Most accident conditions do not result in an increase in the activity of any of the regions. Examples of these accident conditions are the loss of cooling (reactivity increase with decreasing water density) and the dropping of a fuel assembly on the top of the rack. However, accidents

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BASES

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APPLICABLE SAFETY ANALYSES (continued)

can be postulated that could increase the reactivity. This increase in reactivity is unacceptable with unborated water in the storage pool. Thus, for these accident occurrences, the presence of soluble boron in the cask pit pool prevents criticality in Region 2. The most limiting postulated accident with respect to the cask pit pool has been determined to occur in the spent fuel pool. The postulated accident with respect to the storage configurations assumed in the spent fuel rack criticality analysis is the misplacement of a nominal 4.95 ( $\pm 0.05$ ) wt% U-235 fuel assembly into a storage cell location in the Region 3 checkerboard storage arrangement for an irradiated fuel assembly.

The amount of soluble boron required to maintain  $k_{\text{eff}} \leq 0.95$  due to either fuel misload accident is 700 ppm (Ref. 1).

A spent fuel boron dilution analysis was performed to ensure that sufficient time is available to detect and mitigate dilution of the spent fuel pool prior to exceeding the  $k_{\text{eff}}$  design basis limit of 0.95 (Ref. 3). The spent fuel pool boron dilution analysis concluded that an inadvertent or unplanned event that would result in a dilution of the spent fuel pool boron concentration from 2000 ppm to 700 ppm is not a credible event. The accident analyses are provided in the UFSAR, Section 4.3.2.7 (Ref. 4).

The concentration of dissolved boron in the cask pit pool satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

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LCO

The cask pit pool boron concentration is required to be  $\geq 2000$  ppm. The specified concentration of dissolved boron in the cask pit pool preserves the assumptions used in the analyses of the potential critical accident scenarios as described in Reference 4. This concentration of dissolved boron is the minimum required concentration for fuel assembly storage and movement within the cask pit pool.

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APPLICABILITY

This LCO applies whenever fuel assemblies are stored in the cask pit pool, until a complete cask pit pool verification has been performed following the last movement of fuel assemblies in the cask pit 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.

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ACTIONS

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

The Required Actions are modified by a Note indicating that LCO 3.0.3 does not apply.

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BASES

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ACTIONS (continued)

When the concentration of boron in the cask pit pool is less than required, immediate action must be taken to preclude the occurrence of an accident or to mitigate the consequences of an accident in progress. This is most efficiently achieved by immediately suspending the movement of fuel assemblies. The concentration of boron is restored simultaneously with suspending movement of fuel assemblies. Alternatively, beginning a verification of the cask pit pool fuel locations, to ensure proper locations of the fuel, can be performed. However, prior to resuming movement of fuel assemblies, the concentration of boron must be restored. This does not preclude movement of a fuel assembly to a safe position.

If the LCO is not met while moving irradiated fuel assemblies in MODE 5 or 6, LCO 3.0.3 would not be applicable. 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.

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SURVEILLANCE  
REQUIREMENTS

SR 3.7.17.1

This SR verifies that the concentration of boron in the cask pit pool is within the required limit. As long as this SR is met, the analyzed accidents are fully addressed.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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REFERENCES

1. Stanely E. Turner (Holtec International), "Criticality Safety Analyses of Sequoyah Spent Fuel Racks with Alternative Arrangements," HI-992349.
  2. Double contingency principle of ANSI N16.1-1975, as specified in the April 14, 1978 NRC letter (Section 1.2) and implied in the proposed revision to Regulatory Guide 1.13 (Section 1.4, Appendix A).
  3. K K Niyogi (Holtec International), "Boron Dilution Analysis," HI-992302.
  4. UFSAR, Section 4.3.2.7.
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## B 3.8 ELECTRICAL POWER SYSTEMS

### B 3.8.1 AC Sources - Operating

#### BASES

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##### BACKGROUND

The Class 1E AC Electrical Power Distribution System AC sources consist of the offsite power sources (preferred power sources), 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 Electrical Power Distribution System is divided into two redundant and independent load groups with two 6.9 kV Shutdown Boards in each load group. Each 6.9 kV Shutdown Board has a connection to a preferred offsite power source and a DG. The 6.9 kV Shutdown Boards in a load group (i.e., 1A-A and 2A-A, or 1B-B and 2B-B) are normally powered by the same offsite power circuit. Each 6.9 kV Shutdown Board can also be powered by a dedicated DG. Two DGs associated with one load group can provide all safety related functions to mitigate a loss of coolant accident (LOCA) in one unit and safely shut down the other unit. The Train A and Train B ESF systems each provide for the minimum safety functions necessary to shut down the plant and maintain it in a safe shutdown condition.

The Unit 2 core cooling systems and containment systems (e.g., Safety Injection (SI), Auxiliary Feedwater (AFW), Residual Heat Removal (RHR), Centrifugal Charging pump, Containment Spray, and Air Return System (ARS) fan) are unitized (not shared with Unit 1) and are powered from 6.9 kV Shutdown Boards 2A-A and 2B-B. However, some safety-related systems (e.g., Essential Raw Cooling Water (ERCW), Component Cooling Water (CCS), Emergency Gas Treatment (EGTS), Auxiliary Building Gas Treatment (ABGTS), Control Room Emergency Ventilation (CREVS), and Control Room Air-Conditioning (CRACS)) are shared with Unit 1. The AC sources for these loads are distributed across all four 6.9 kV Shutdown Boards. Therefore, two qualified circuits between the offsite transmission network and the onsite Class 1E AC Electrical Power Distribution System and a separate and independent DG for each 6.9 kV Shutdown Board ensure the availability of power to each of these systems.

The offsite power distribution system consists of two 161 kV buses supplied by eight 161 kV feeders and two 500 kV buses supplied by five 500 kV feeders. The output of the Unit 1 main generator is fed to the 500 kV buses and the output of the Unit 2 main generator is fed to the 161 kV buses. The output of each unit's main generator is also capable of

## BASES

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### BACKGROUND (continued)

supplying power to an offsite circuit (via the Unit Station Service Transformers (USSTs)) with a Generator Circuit Breaker (GCB) providing isolation between the main generator and the main bank transformers. When the main generator is not operating, the main bank transformers function as step down transformers and supply electrical power from the grid to the USSTs.

Offsite power can also be supplied by the Common Station Service Transformers (CSSTs) via the 6.9 kV Start Buses and 6.9 kV Unit Boards. Offsite power will normally be supplied from the USSTs to the 6.9 kV Shutdown Boards via the 6.9 kV Unit Boards, and will automatically transfer at least one power supply to an alternate power supply (CSST A or CSST C) on a trip of the Power Circuit Breakers (PCBs). CSST C is the alternate power source for 6.9 kV Shutdown Boards 1A-A and 2A-A, and CSST A is the alternate power source for 6.9 kV Shutdown Boards 1B-B and 2B-B. (CSST B is a spare transformer with two sets of secondary windings that can be used to supply a total of two Start Buses for CSST A and/or CSST C, with each Start Bus on a separate CSST B secondary winding.) Therefore, two electrically and physically separated circuits provide AC power through a combination of the USSTs and/or CSSTs to the 6.9 kV Shutdown Boards. Each offsite circuit is capable of providing power to one train of ESF loads. A detailed description of the offsite power network and the circuits to the Class 1E 6.9 kV Shutdown Boards is found in UFSAR, 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 (beginning at the switchyard) to one load group of Class 1E 6.9 kV Shutdown Boards (ending at the supply side of the normal or alternate supply circuit breaker).

The onsite standby power source for each 6.9 kV Shutdown Board is a dedicated DG. DGs 1A-A, 1B-B, 2A-A, and 2B-B are separate and independent and are dedicated to 6.9 kV Shutdown Boards 1A-A, 1B-B, 2A-A, and 2B-B, respectively. Each diesel generator set consists of two diesel engines in tandem driving a common generator with a normal synchronous speed of approximately 900 rpm. A DG starts automatically on a safety injection (SI) signal (i.e., low pressurizer pressure, high containment pressure, or low steam line pressure signals) or on a 6.9 kV Shutdown Board degraded voltage, unbalanced voltage, or loss-of-voltage signal (refer to LCO 3.3.5, "Loss of Power (LOP) Diesel Generator (DG) Start Instrumentation"). After the DG has started, it will automatically tie to its respective 6.9 kV Shutdown Board after offsite power is tripped as a consequence of a 6.9 kV Shutdown Board degraded voltage, unbalanced voltage, or loss-of-voltage signal, independent of or

## BASES

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### BACKGROUND (continued)

coincident with an SI signal. The DGs will also start and operate in the standby mode without tying to the 6.9 kV Shutdown Board on an SI signal alone. Following the trip of offsite power, a loss-of-voltage signal strips nonpermanent loads from the 6.9 kV Shutdown Board. When the DG is tied to the 6.9 kV Shutdown Board, loads are then sequentially connected to its respective 6.9 kV Shutdown Board by individual load sequence timers.

In the event of a loss of preferred power, the 6.9 kV Shutdown Boards 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 LOCA.

Certain required unit loads are returned to service in a predetermined sequence in order to prevent overloading the DG in the process. Within the required time interval (UFSAR Table 8.3.1-2) after the initiating signal is received, automatic and permanently connected loads needed to recover the unit or maintain it in a safe condition are returned to service.

Ratings for the 1A-A, 1B-B, 2A-A and 2B-B DGs satisfy the requirements of Regulatory Guide 1.9 (Ref. 3). The continuous service rating of each DG is 4400 kW with 10% overload permissible for up to 2 hours in any 24 hour period. The ESF loads that are powered from the 6.9 kV Shutdown Boards are listed in Reference 2.

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### APPLICABLE SAFETY ANALYSES

The initial conditions of DBA and transient analyses in the UFSAR, 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 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 10 CFR 50.36(c)(2)(ii).

## BASES

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### LCO

Two qualified circuits between the offsite transmission network and the onsite Class 1E Electrical Power System and separate and independent DGs for each 6.9 kV Shutdown Board 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.

Each qualified offsite circuit must be physically independent, capable of maintaining rated frequency and voltage, and accepting required loads during an accident, while connected to the 6.9 kV Shutdown Boards.

The minimum required switchyard voltages are determined by evaluation of plant accident loading and the associated voltage drops between the transmission network and these loads. These minimum voltage values are provided to TVA's Transmission Operations for use in system studies to support operation of the transmission network in a manner that will maintain the necessary voltages. Transmission Operations is required to notify SQN Operations if it is determined that the transmission network may not be able to support accident loading or shutdown operations as required by 10 CFR 50, Appendix A, GDC-17. Any offsite power circuits supplied by the transmission network that are not able to support accident loading or shutdown operations are inoperable.

The USSTs utilize auto load tap changers to provide the required voltage response for accident loading. The load tap changer associated with a USST is required to be functional and in "automatic" for the USST to supply power to a 6.9 kV Unit Board.

Each required offsite circuit is that combination of power sources described below that are either connected to the Class 1E AC Electrical Power Distribution System, or is available to be connected to the Class 1E AC Electrical Power Distribution System through automatic transfer at the 6.9 kV Unit Boards.

The following offsite power configurations meet the requirements of the LCO:

1. Two offsite circuits consisting of a AND b (no board transfers required; a loss of either circuit will not prevent the minimum safety functions from being performed):
  - a. From the 161 kV transmission network, through CSST A (winding X) to Start Bus 1A to 6.9 kV Shutdown Board 1B-B (through 6.9 kV Unit Board 1C), and CSST A (winding Y) to Start Bus 2A to 6.9 kV Shutdown Board 2B-B (through 6.9 kV Unit Board 2C); AND

## BASES

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### LCO (continued)

- b. From the 161 kV transmission network, through CSST C (winding X) to Start Bus 2B to 6.9 kV Shutdown Board 2A-A (through 6.9 kV Unit Board 2B), and CSST C (winding Y) to Start Bus 1B to 6.9 kV Shutdown Board 1A-A (through 6.9 kV Unit Board 1B).
2. Two offsite circuits consisting of a AND b (relies on automatic transfer from alignment a.1) to b.2)(b), or a.2) to b.1)(a) on a loss of USSTs 1A and 1B, OR relies on automatic transfer from alignment a.3) to b.2)(a), or a.4) to b.1)(b) on a loss of USSTs 2A and 2B):
    - a. Normal power source alignments
      - 1) From the 500 kV switchyard through USST 1A to 6.9 kV Shutdown Board 1A-A (through 6.9 kV Unit Board 1B);
      - 2) From the 500 kV switchyard through USST 1B to 6.9 kV Shutdown Board 1B-B (through 6.9 kV Unit Board 1C);
      - 3) From the 161 kV switchyard through USST 2A to 6.9 kV Shutdown Board 2A-A (through 6.9 kV Unit Board 2B); AND
      - 4) From the 161 kV switchyard through USST 2B to 6.9 kV Shutdown Board 2B-B (through 6.9 kV Unit Board 2C).
    - b. Alternate power source alignments
      - 1) From the 161 kV transmission network, through:
        - (a) CSST A (winding X) to Start Bus 1A to 6.9 kV Shutdown Board 1B-B (through 6.9 kV Unit Board 1C); AND
        - (b) CSST A (winding Y) to Start Bus 2A to 6.9 kV Shutdown Board 2B-B (through 6.9 kV Unit Board 2C); OR
      - 2) From the 161 kV transmission network, through:
        - (a) CSST C (winding X) to Start Bus 2B to 6.9 kV Shutdown Board 2A-A (through 6.9 kV Unit Board 2B); AND
        - (b) CSST C (winding Y) to Start Bus 1B to 6.9 kV Shutdown Board 1A-A (through 6.9 kV Unit Board 1B).

## BASES

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### LCO (continued)

3. Two offsite circuits consisting of a AND b (relies on automatic transfer from alignment a.1) to b.1) and b.2) on a loss of the Unit 2 USSTs; a loss of alignment a.2) or a.3) will not prevent the minimum safety functions from being performed):
  - a. Normal power source alignments
    - 1) From the 161 kV switchyard through USST 2A to 6.9 kV Shutdown Board 2A-A (through 6.9 kV Unit Board 2B), and USST 2B to 6.9 kV Shutdown Board 2B-B (through 6.9 kV Unit Board 2C);
    - 2) From the 161 kV transmission network, through CSST A (winding X) to Start Bus 1A to 6.9 kV Shutdown Board 1B-B (through 6.9 kV Unit Board 1C); AND
    - 3) From the 161 kV transmission network, through CSST C (winding Y) to Start Bus 1B to 6.9 kV Shutdown Board 1A-A (through 6.9 kV Unit Board 1B).
  - b. Alternate power source alignments
    - 1) From the 161 kV transmission network, through CSST A (winding Y) to Start Bus 2A to 6.9 kV Shutdown Board 2B-B (through 6.9 kV Unit Board 2C); AND
    - 2) From the 161 kV transmission network, through CSST C (winding X) to Start Bus 2B to 6.9 kV Shutdown Board 2A-A (through 6.9 kV Unit Board 2B).
4. Two offsite circuits consisting of a AND b (relies on automatic transfer from alignment a.1) to b.1) and b.2) on a loss of the Unit 1 USSTs; a loss of alignment a.2) or a.3) will not prevent the minimum safety functions from being performed):
  - a. Normal power source alignments
    - 1) From the 500 kV switchyard through USST 1A to 6.9 kV Shutdown Board 1A-A (through 6.9 kV Unit Board 1B), and USST 1B to 6.9 kV Shutdown Board 1B-B (through 6.9 kV Unit Board 1C);
    - 2) From the 161 kV transmission network, through CSST A (winding Y) to Start Bus 2A to 6.9 kV Shutdown Board 2B-B (through 6.9 kV Unit Board 2C); AND

## BASES

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### LCO (continued)

3) From the 161 kV transmission network, through CSST C (winding X) to Start Bus 2B to 6.9 kV Shutdown Board 2A-A (through 6.9 kV Unit Board 2B).

b. Alternate power source alignments

1) From the 161 kV transmission network, through CSST A (winding X) to Start Bus 1A to 6.9 kV Shutdown Board 1B-B (through 6.9 kV Unit Board 1C); AND

2) From the 161 kV transmission network, through CSST C (winding Y) to Start Bus 1B to 6.9 kV Shutdown Board 1A-A (through 6.9 kV Unit Board 1B).

Other offsite power configurations are possible using different combinations of available USSTs and CSSTs, as long as the alignments are consistent with the analyzed configurations, and the alignments comply with the requirements of GDC 17.

Each DG must be capable of starting, accelerating to rated speed and voltage, and connecting to its respective 6.9 kV Shutdown Board on detection of board 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 6.9 kV Shutdown Board. 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.

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.

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### APPLICABILITY

The AC sources 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

## BASES

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### APPLICABILITY (continued)

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

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

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### ACTIONS

The ACTIONS are modified by a Note that prohibits the application of LCO 3.0.4.b to an inoperable DG. There is an increased risk associated with entering a MODE or other specified condition in the Applicability with an inoperable DG and the provisions of LCO 3.0.4.b, which allow entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, should not be applied in this circumstance.

A second Note provides the appropriate restrictions for a de-energized shutdown board. 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 ACTIONS are modified by a Note to indicate that when any Condition(s) is entered with no AC power source to any shutdown board resulting in a de-energized shutdown board, the Conditions and Required Actions for LCO 3.8.9, "Distribution Systems - Operating," must be immediately entered. This allows LCO 3.8.1 Conditions to provide requirements for the loss of any combination of AC Sources, without regard to whether a shutdown board is de-energized and LCO 3.8.9 to provide the appropriate restrictions for a de-energized shutdown board.

#### A.1

To ensure a highly reliable power source remains with one offsite circuit inoperable for reasons other than Condition C, 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 D, for two offsite circuits inoperable, is entered.

#### A.2

Required Action A.2, which only applies if a Unit 2 6.9 kV Shutdown Board cannot be powered from an offsite source, is intended to provide assurance that an event coincident with a single failure of the associated

## BASES

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### ACTIONS (continued)

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. However, due to flow requirements of accident scenarios such as Feedwater Line Break (FWLB) and Small Break Loss of Coolant Accident (SBLOCA), the Turbine Driven Auxiliary Feedwater Pump should also be considered a required redundant feature.

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. 6.9 kV Shutdown Board 2A-A or 2B-B has no offsite power supplying it loads and
- b. A required feature on the other train is inoperable.

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

Discovering no offsite power to a Unit 2 6.9 kV Shutdown Board 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 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.

## BASES

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### ACTIONS (continued)

#### A.3

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

#### B.1

To ensure a highly reliable power source remains with one or more Train A DGs, or one or more Train B DGs inoperable, 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.

#### B.2

Required Action B.2 is intended to provide assurance that a loss of offsite power, during the period that a DG(s) 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. However, due to flow requirements of accident scenarios such as Feedwater Line Break (FWLB) and Small Break Loss of Coolant Accident (SBLOCA), the Turbine Driven Auxiliary Feedwater Pump should also be considered a required redundant feature. Redundant required feature failures consist of inoperable features associated with a train, redundant to the train that has an inoperable DG(s).

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"

## BASES

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### ACTIONS (continued)

for beginning the allowed outage time "clock." In this Required Action, the Completion Time only begins on discovery that both:

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

If at any time during the existence of this Condition (one or more DGs in a train inoperable) a required feature subsequently becomes inoperable, this Completion Time would begin to be tracked.

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

#### B.3.1 and B.3.2

Required Action B.3.1 provides an allowance to avoid unnecessary testing of OPERABLE DG(s). If it can be determined that the cause of the inoperable DG does not exist on the OPERABLE DG, SR 3.8.1.2 does not have to be performed. If the cause of inoperability exists on other DGs, the other DGs would be declared inoperable upon discovery and Condition F of LCO 3.8.1 would be entered if one or more DG(s) in Train A and Train B are inoperable. 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 DGs, performance of SR 3.8.1.2 suffices to provide assurance of continued OPERABILITY of that DG.

## BASES

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### ACTIONS (continued)

In the event the inoperable DG(s) is restored to OPERABLE status prior to completing either B.3.1 or B.3.2, the 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

In Condition B, the remaining OPERABLE DG(s) and offsite circuits are adequate to supply electrical power to the onsite Class 1E AC Electrical Power Distribution System. The 7 day 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.

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

Condition C is entered for an offsite circuit inoperable solely due to an inoperable power source to 6.9 kV Shutdown Board 1A-A or 1B-B. Required Action C.1 verifies the OPERABILITY of the remaining offsite circuit within an hour of the inoperability and every 8 hours thereafter. Since the Required Action only specifies "perform," a failure of the SR 3.8.1.1 acceptance criteria does not result in a Required Action not met.

The Completion Time for Required Action C.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. 6.9 kV Shutdown Board 1A-A or 1B-B has no offsite power; and
- b. A required feature on the other train is inoperable.

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### ACTIONS (continued)

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

A Completion Time of 24 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 support these functions. The Completion Time takes into account the component OPERABILITY of the remaining redundant feature(s), a reasonable time for repairs, and the low probability of a DBA occurring during this period.

Operation may continue in Condition C for a period of 7 days. With one offsite circuit inoperable, the reliability of the functions is degraded. The potential for the loss of offsite power to the redundant feature(s) is increased, with the attendant potential for a challenge to their safety functions.

The required offsite circuit must be returned to OPERABLE status within 7 days. The 7 days Completion Time takes into account the capacity and capability of the remaining AC sources providing electrical power to the required feature(s), a reasonable time for repairs and the low probability of a DBA occurring during this period of time.

#### D.1 and D.2

Required Action D.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.

## BASES

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### ACTIONS (continued)

The Completion Time for Required Action D.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.

If at any time during the existence of Condition D (two offsite circuits inoperable) 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 D 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.

## BASES

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### ACTIONS (continued)

With both 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 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 or Condition C, as applicable.

#### E.1 and E.2

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

In Condition E, individual redundancy is lost in both the offsite electrical power system and the onsite AC electrical power system. Since power system redundancy is provided by two diverse sources of power, however, the reliability of the power systems in this Condition may appear higher than that in Condition D (loss of both 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.

## BASES

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### ACTIONS (continued)

#### F.1

With one or more Train A DG(s) and one or more Train B DG(s) inoperable, there are insufficient standby AC sources available to power an entire load group. Thus, with an assumed loss of offsite electrical power, insufficient standby AC sources are available to power the minimum required ESF functions. Since the offsite electrical power system is the only source of AC power for this level of degradation, the risk associated with continued operation for a very short time could be less than that associated with an immediate controlled shutdown (the immediate shutdown could cause grid instability, which could result in a total loss of AC power). Since any inadvertent generator trip could also result in a total loss of offsite AC power, however, the time allowed for continued operation is severely restricted. The intent here is to avoid the risk associated with an immediate controlled shutdown and to minimize the risk associated with this level of degradation.

In this Condition, operation may continue for a period that should not exceed 2 hours, consistent with the guidance provided in Reference 6.

#### G.1 and G.2

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

#### H.1 and I.1

Conditions H and I correspond to a level of degradation in which redundancy in the AC electrical power supplies cannot be assured. 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.

## BASES

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### SURVEILLANCE REQUIREMENTS

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

Where the SRs discussed herein specify voltage and frequency tolerances, the following is applicable. The minimum steady state output voltage of 6210 V is 90% of the nominal 6900 V output voltage. This value, which is specified in ANSI C84.1 (Ref. 10), allows for voltage drop to the terminals of 6600 V motors whose minimum operating voltage is specified as 90% or 5940 V. It also allows for voltage drops to motors and other equipment down through the 120 V level where minimum operating voltage is also usually specified as 90% of name plate rating. The specified maximum steady state output voltage of 7260 V is equal to the maximum operating voltage specified for 6600 V motors. It ensures that for a lightly loaded distribution system, the voltage at the terminals of 6600 V motors is no more than the maximum rated operating voltages. The steady state minimum and maximum frequency values are 59.8 Hz and 60.2 Hz, which are consistent with the recommendations in Regulatory Guide 1.9 (Ref. 3). These values ensure that the safety related plant equipment powered from the DGs is capable of performing its safety functions.

#### SR 3.8.1.1

This SR ensures proper circuit continuity for the offsite AC electrical power supply to the onsite distribution network and availability of offsite AC electrical power. The breaker alignment verifies that each breaker is in its correct position to ensure that distribution buses and loads are connected to their preferred power source, and that appropriate independence of offsite circuits is maintained.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

## BASES

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### SURVEILLANCE REQUIREMENTS (continued)

#### 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 1 for SR 3.8.1.2 and Note for SR 3.8.1.7) 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.

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

In order to reduce stress and wear on diesel engines, the manufacturer recommends 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. These start procedures are the intent of Note 2.

SR 3.8.1.7 requires that 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 UFSAR, Chapter 15 (Ref. 5).

The 10 second start requirement is not applicable to SR 3.8.1.2 (see Note 2) when a modified start procedure as described above is used. During this testing, the diesel is not in an accident mode and the frequency is controlled by the operator instead of the governor's accident speed reference. If a modified start is not used, the 10 second start requirement of SR 3.8.1.7 applies.

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

In addition to the SR requirements, the time for the DG to reach steady state operation, unless the modified DG start method is employed, is periodically monitored and the trend evaluated to identify degradation of governor and voltage regulator performance.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

## BASES

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### SURVEILLANCE REQUIREMENTS (continued)

#### SR 3.8.1.3

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

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

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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 board loads, do not invalidate this test. Similarly, momentary power factor transients above the limit do 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 engine-mounted "day" 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 Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

## BASES

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### 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 engine-mounted “day” tank 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 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 Frequency is controlled under the Surveillance Frequency Control Program.

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 the storage system to the engine-mounted “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 Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

#### SR 3.8.1.7

See SR 3.8.1.2.

BASES

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.8

Transfer of the power supply to each 6.9 kV Unit Board from the normal supply to the alternate supply demonstrates the OPERABILITY of the alternate supply to power the shutdown loads. This SR is modified by two Notes.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

This SR is modified by two Notes. The reason for Note 1 is that, during operation with the reactor critical, performance of this SR for the 2A, 2B, 2C, and 2D Unit Boards could cause perturbations to the electrical distribution systems that could challenge continued steady state operation and, as a result, unit safety systems. This restriction from normally performing the Surveillance in MODE 1 or 2 is further amplified to allow the Surveillance to be performed for the purpose of reestablishing OPERABILITY (e.g., post work testing following corrective maintenance, corrective modification, deficient or incomplete surveillance testing, and other unanticipated OPERABILITY concerns) provided an assessment determines plant safety is maintained or enhanced. This assessment shall, as a minimum, consider the potential outcomes and transients associated with a failed Surveillance, a successful Surveillance, and a perturbation of the offsite or onsite system when they are tied together or operated independently for the Surveillance; as well as the operator procedures available to cope with these outcomes. These shall be measured against the avoided risk of a plant shutdown and startup to determine that plant safety is maintained or enhanced when the Surveillance is performed in MODE 1 or 2. Risk insights or deterministic methods may be used for this assessment. Credit may be taken for unplanned events that satisfy this SR.

Note 2 specifies that transfer capability is only required to be met for 6.9 kV Unit Boards that require normal and alternate power supplies. When both load groups are being supplied power by the USSTs, only the 6.9 kV Unit Boards associated with one load group are required to have normal and alternate power supplies. Therefore, only one CSST is required to be OPERABLE and available as an alternate power supply. Manual transfers between the normal supply and the alternate supply are also required to meet the SR. However, delayed access to an offsite circuit is not credited in the accident analysis.

## BASES

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### SURVEILLANCE REQUIREMENTS (continued)

#### SR 3.8.1.9

Each DG is provided with an engine overspeed trip to prevent damage to the engine. Recovery from the transient caused by the loss of a large load could cause diesel engine overspeed, which, if excessive, might result in a trip of the engine. This Surveillance demonstrates the DG load response characteristics and capability to reject the largest single load (600 kW) without exceeding predetermined voltage and frequency and while maintaining a specified margin to the overspeed trip. 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 board, or
- b. Tripping its associated single largest post-accident load with the DG solely supplying the board.

Consistent with Regulatory Guide 1.9 (Ref. 3), 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 and voltage 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 maximum transient 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 Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

This SR is modified by a Note. The Note ensures that the DG is tested under load conditions that are as close to design basis conditions as possible. When synchronized with offsite power, testing should be performed at a power factor of  $\leq 0.89$ . This power factor is representative of the actual inductive loading a DG would see under design basis accident conditions. Under certain conditions, however, the Note allows the Surveillance to be conducted at a power factor other than  $\leq 0.89$ . These conditions occur when grid voltage is high, and the additional field excitation needed to get the power factor to  $\leq 0.89$  results in voltages on the emergency boards that are too high.

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### SURVEILLANCE REQUIREMENTS (continued)

Under these conditions, the power factor should be maintained as close as practicable to 0.89 while still maintaining acceptable voltage limits on the emergency boards. In other circumstances, the grid voltage may be such that the DG excitation levels needed to obtain a power factor of 0.89 may not cause unacceptable voltages on the emergency boards, but the excitation levels are in excess of those recommended for the DG. In such cases, the power factor shall be maintained as close as practicable to 0.89 without exceeding the DG excitation limits.

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

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

This SR has been modified by a Note. The Note ensures that the DG is tested under load conditions that are as close to design basis conditions as possible. When synchronized with offsite power, testing should be performed at a power factor of  $\leq 0.89$ . This power factor is representative of the actual inductive loading a DG would see under design basis accident conditions. Under certain conditions, however, the Note allows the Surveillance to be conducted at a power factor other than  $\leq 0.89$ . These conditions occur when grid voltage is high, and the additional field excitation needed to get the power factor to  $\leq 0.89$  results in voltages on the emergency boards that are too high. Under these conditions, the power factor should be maintained as close as practicable to 0.89 while still maintaining acceptable voltage limits on the emergency boards. In other circumstances, the grid voltage may be such that the DG excitation levels needed to obtain a power factor of 0.89 may not cause unacceptable voltages on the emergency boards, but the excitation levels are in excess of those recommended for the DG. In such cases, the power factor shall be maintained as close as practicable to 0.89 without exceeding the DG excitation limits.

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### SURVEILLANCE REQUIREMENTS (continued)

#### SR 3.8.1.11

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

The DG autostart 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 is achieved.

The requirement to verify the connection and power supply 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, Emergency Core Cooling Systems (ECCS) injection valves are not desired to be stroked open, or high pressure injection systems are not capable of being operated at full flow, or residual heat removal (RHR) 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 systems 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 Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

This SR is modified by two Notes. The reason for Note 1 is to minimize wear and tear on the DGs during testing. For the purpose of this testing, the DGs must be started from standby conditions, that is, with the engine coolant and oil continuously circulated and temperature maintained consistent with manufacturer recommendations. The reason for Note 2 is that performing the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems. This restriction from normally performing the Surveillance for DGs 2A-A and 2B-B in MODE 1 or 2 is further amplified to allow portions of the Surveillance to be performed for the purpose of

## BASES

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### SURVEILLANCE REQUIREMENTS (continued)

reestablishing OPERABILITY (e.g., post work testing following corrective maintenance, corrective modification, deficient or incomplete surveillance testing, and other unanticipated OPERABILITY concerns) provided an assessment determines plant safety is maintained or enhanced. This assessment shall, as a minimum, consider the potential outcomes and transients associated with a failed partial Surveillance, a successful partial Surveillance, and a perturbation of the offsite or onsite system when they are tied together or operated independently for the partial Surveillance; as well as the operator procedures available to cope with these outcomes. These shall be measured against the avoided risk of a plant shutdown and startup to determine that plant safety is maintained or enhanced when portions of the Surveillance are performed in MODE 1 or 2. Risk insights or deterministic methods may be used for the assessment. Credit may be taken for unplanned events that satisfy this SR.

#### SR 3.8.1.12

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

The requirement to verify the connection of permanent and 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, or high pressure injection systems are not capable of being operated at full flow, or RHR 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 Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

## BASES

## SURVEILLANCE REQUIREMENTS (continued)

This SR is modified by two Notes. The reason for Note 1 is to minimize wear and tear on the DGs during testing. For the purpose of this testing, the DGs must be started from standby conditions, that is, with the engine coolant and oil continuously circulated and temperature maintained consistent with manufacturer recommendations. The reason for Note 2 is that 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. This restriction from normally performing the Surveillance for DGs 2A-A and 2B-B in MODE 1 or 2 is further amplified to allow portions of the Surveillance to be performed for the purpose of reestablishing OPERABILITY (e.g., post work testing following corrective maintenance, corrective modification, deficient or incomplete surveillance testing, and other unanticipated OPERABILITY concerns) provided an assessment determines plant safety is maintained or enhanced. This assessment shall, as a minimum, consider the potential outcomes and transients associated with a failed partial Surveillance, a successful partial Surveillance, and a perturbation of the offsite or onsite system when they are tied together or operated independently for the partial Surveillance; as well as the operator procedures available to cope with these outcomes. These shall be measured against the avoided risk of a plant shutdown and startup to determine that plant safety is maintained or enhanced when portions of the Surveillance are performed in MODE 1 or 2. Risk insights or deterministic methods may be used for the assessment. 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, an ESF actuation test signal, or both. Noncritical automatic trips are all automatic trips except:

- a. Engine overspeed; and
- b. Generator differential current.

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.

## BASES

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### SURVEILLANCE REQUIREMENTS (continued)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

The SR is modified by a Note. The reason for the Note is that performing the Surveillance would remove a required DG from service. This restriction from normally performing the Surveillance for DGs 2A-A and 2B-B in MODE 1 or 2 is further amplified to allow the Surveillance to be performed for the purpose of reestablishing OPERABILITY (e.g., post work testing following corrective maintenance, corrective modification, deficient or incomplete surveillance testing, and other unanticipated OPERABILITY concerns) provided an assessment determines plant safety is maintained or enhanced. This assessment shall, as a minimum, consider the potential outcomes and transients associated with a failed Surveillance, a successful Surveillance, and a perturbation of the offsite or onsite system when they are tied together or operated independently for the Surveillance; as well as the operator procedures available to cope with these outcomes. These shall be measured against the avoided risk of a plant shutdown and startup to determine that plant safety is maintained or enhanced when the Surveillance is performed in MODE 1 or 2. Risk insights or deterministic methods may be used for this assessment. Credit may be taken for unplanned events that satisfy this SR.

#### SR 3.8.1.14

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

The load band is provided to avoid routine overloading of the DG. Routine overloading may result in more frequent teardown inspections in accordance with vendor recommendations in order to maintain DG OPERABILITY.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

## BASES

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### SURVEILLANCE REQUIREMENTS (continued)

This Surveillance is modified by two Notes. Note 1 states that momentary transients due to changing board loads do not invalidate this test. Similarly, momentary power factor transients above the power factor limit will not invalidate the test. Note 2 ensures that the DG is tested under load conditions that are as close to design basis conditions as possible. When synchronized with offsite power, testing should be performed at a power factor of  $\leq 0.89$ . This power factor is representative of the actual inductive loading a DG would see under design basis accident conditions. Under certain conditions, however, Note 2 allows the Surveillance to be conducted at a power factor other than  $\leq 0.89$ . These conditions occur when grid voltage is high, and the additional field excitation needed to get the power factor to  $\leq 0.89$  results in voltages on the emergency boards that are too high. Under these conditions, the power factor should be maintained as close as practicable to 0.89 while still maintaining acceptable voltage limits on the emergency boards. In other circumstances, the grid voltage may be such that the DG excitation levels needed to obtain a power factor of 0.89 may not cause unacceptable voltages on the emergency boards, but the excitation levels are in excess of those recommended for the DG. In such cases, the power factor shall be maintained close as practicable to 0.89 without exceeding the DG excitation limits.

#### 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 Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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 board 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.

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### 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 the DG can be returned to ready to load status when offsite power is restored. It also ensures that the autostart 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 an autoclose signal on board undervoltage, and the load sequence timers are reset.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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. This restriction from normally performing the Surveillance for DGs 2A-A and 2B-B in MODE 1 or 2 is further amplified to allow the Surveillance to be performed for the purpose of reestablishing OPERABILITY (e.g., post work testing following corrective maintenance, corrective modification, deficient or incomplete surveillance testing, and other unanticipated OPERABILITY concerns) provided an assessment determines plant safety is maintained or enhanced. This assessment shall, as a minimum, consider the potential outcomes and transients associated with a failed Surveillance, a successful Surveillance, and a perturbation of the offsite or onsite system when they are tied together or operated independently for the Surveillance; as well as the operator procedures available to cope with these outcomes. These shall be measured against the avoided risk of a plant shutdown and startup to determine that plant safety is maintained or enhanced when the Surveillance is performed in MODE 1 or 2. Risk insights or deterministic methods may be used for this assessment. Credit may be taken for unplanned events that satisfy this SR.

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### SURVEILLANCE REQUIREMENTS (continued)

#### SR 3.8.1.17

Under accident and loss of offsite power conditions loads are sequentially connected to the board by the load sequence timers. 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 5% load sequence time interval tolerance ensures that sufficient time exists for the DG to restore frequency and voltage prior to applying the next load and that safety analysis assumptions regarding ESF equipment time delays are not violated. Reference 2 provides a summary of the automatic loading of Shutdown Boards.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

#### SR 3.8.1.18

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 Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

## BASES

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### SURVEILLANCE REQUIREMENTS (continued)

This SR is modified by two Notes. The reason for Note 1 is to minimize wear and tear on the DGs during testing. For the purpose of this testing, the DGs must be started from standby conditions, that is, with the engine coolant and oil continuously circulated and temperature maintained consistent with manufacturer recommendations for DGs. The reason for Note 2 is that the performance of the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems. This restriction from normally performing the Surveillance for DGs 2A-A and 2B-B in MODE 1 or 2 is further amplified to allow portions of the Surveillance to be performed for the purpose of reestablishing OPERABILITY (e.g., post work testing following corrective maintenance, corrective modification, deficient or incomplete surveillance testing, and other unanticipated OPERABILITY concerns) provided an assessment determines plant safety is maintained or enhanced. This assessment shall, as a minimum, consider the potential outcomes and transients associated with a failed partial Surveillance, a successful partial Surveillance, and a perturbation of the offsite or onsite system when they are tied together or operated independently for the partial Surveillance; as well as the operator procedures available to cope with these outcomes. These shall be measured against the avoided risk of a plant shutdown and startup to determine that plant safety is maintained or enhanced when portions of the Surveillance are performed in MODE 1 or 2. Risk insights or deterministic methods may be used for the assessment. Credit may be taken for unplanned events that satisfy this SR.

#### SR 3.8.1.19

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 Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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.

BASES

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- REFERENCES
1. 10 CFR 50, Appendix A, GDC 17.
  2. UFSAR, Chapter 8.
  3. Regulatory Guide 1.9, Rev. 0.
  4. UFSAR, Chapter 6.
  5. UFSAR, Chapter 15.
  6. Regulatory Guide 1.93, Rev. 0, December 1974.
  7. Generic Letter 84-15, "Proposed Staff Actions to Improve and Maintain Diesel Generator Reliability," July 2, 1984.
  8. 10 CFR 50, Appendix A, GDC 18.
  9. Regulatory Guide 1.108, Rev. 1, August 1977.
  10. ANSI C84.1, Voltage Ratings for Electric Power Systems and Equipment (60 Hz).
  11. UFSAR Chapter 10.
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## B 3.8 ELECTRICAL POWER SYSTEMS

### B 3.8.2 AC Sources - Shutdown

#### BASES

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BACKGROUND	A description of the AC sources is provided in the Bases for LCO 3.8.1, "AC Sources - Operating."
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APPLICABLE SAFETY ANALYSES	<p>The OPERABILITY of the minimum AC sources during MODES 5 and 6 and during movement of irradiated fuel assemblies ensures that:</p> <ol style="list-style-type: none"><li>The unit can be maintained in the shutdown or refueling condition for extended periods,</li><li>Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status, and</li><li>Adequate AC electrical power is provided to mitigate events postulated during shutdown, such as a fuel handling accident.</li></ol>
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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. This is because many Design Basis Accidents (DBAs) that are analyzed in MODES 1, 2, 3, and 4 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 result in minimal consequences. These limitations during shutdown conditions are reflected in the LCO for required systems.

During MODES 1, 2, 3, and 4, various deviations from the analysis assumptions and design requirements are allowed within the Required Actions. This allowance is in 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 planned and administratively controlled. Relaxations from MODE 1, 2, 3, and 4 LCO requirements are acceptable during shutdown modes based on:

BASES

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APPLICABLE SAFETY ANALYSES (continued)

- 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 10 CFR 50.36(c)(2)(ii).

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LCO

One offsite circuit capable of supplying the onsite Class 1E power distribution subsystem(s) of LCO 3.8.10, "Distribution Systems - Shutdown," ensures that all required loads are powered from offsite power. Two OPERABLE DGs, associated with a distribution system train required to be OPERABLE by LCO 3.8.10, ensures a diverse power source is available to provide electrical power support, assuming a loss of the offsite circuit. Together, OPERABILITY of the required offsite circuit and DGs 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 6.9 kV shutdown boards. Qualified offsite circuits are those that are described in the Bases for LCO 3.8.1 and are part of the licensing basis for the unit.

Each required offsite circuit is that combination of power sources described in the Bases of LCO 3.8.1.

## BASES

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### LCO (continued)

The DGs must be capable of starting, accelerating to rated speed and voltage, and connecting to their respective 6.9 kV shutdown board on detection of board undervoltage. This sequence must be accomplished within 10 seconds. The DG must be capable of accepting required loads within the assumed loading sequence intervals, and continue to operate until offsite power can be restored to the 6.9 kV shutdown boards. 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.

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### APPLICABILITY

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

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

The AC power requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.1.

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### ACTIONS

LCO 3.0.3 is not applicable while in MODE 5 or 6. However, since irradiated fuel assembly movement can occur in MODE 1, 2, 3, or 4, the ACTIONS have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operations. Entering LCO 3.0.3, while in MODE 1, 2, 3, or 4 would require the unit to be shutdown unnecessarily.

#### A.1

An offsite circuit would be considered inoperable if it were not available to one required ESF train. Although two trains are required by LCO 3.8.10, the one train with offsite power available may be capable of supporting sufficient required features to allow continuation of irradiated fuel

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## BASES

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### ACTIONS (continued)

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.

#### 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 one or more required DGs inoperable, the minimum required diversity of AC power sources is not available. It is, therefore, required to suspend movement of irradiated fuel assemblies, CORE ALTERATIONS, and operations involving positive reactivity additions that could result in loss of required SDM (MODE 5) or boron concentration (MODE 6). Suspending positive reactivity additions that could result in failure to meet the minimum SDM or boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than that what would be required in the RCS for minimum SDM or refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation. Introduction of temperature changes including temperature increases when operating with a positive MTC must also be evaluated to ensure they do not result in a loss of required SDM.

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 would not be 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 6.9 kV shutdown board, the ACTIONS for

BASES

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ACTIONS (continued)

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 would provide the appropriate restrictions for the situation involving a de-energized train.

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SURVEILLANCE  
REQUIREMENTS

SR 3.8.2.1

SR 3.8.2.1 requires the SRs from LCO 3.8.1 that are necessary for ensuring the OPERABILITY of the AC sources in other than MODES 1, 2, 3, and 4. SR 3.8.1.8 is not required to be met since only one offsite circuit is required to be OPERABLE. SR 3.8.1.12 and SR 3.8.1.18 are not required to be met because the ESF actuation signal is not required to be OPERABLE. SR 3.8.1.19 is excepted because starting independence is not required with the DG(s) that is not required to be OPERABLE.

This SR is modified by a Note. The reason for the Note is to preclude requiring the OPERABLE DG(s) from being paralleled with the offsite power network or otherwise rendered inoperable during performance of SRs, and to preclude deenergizing a required 6.9 kV shutdown board 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 DGs. It is the intent that these SRs must still be capable of being met, but actual performance is not required during periods when the DGs and offsite circuit is required to be OPERABLE. Refer to the corresponding Bases for LCO 3.8.1 for a discussion of each SR.

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REFERENCES

None.

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## B 3.8 ELECTRICAL POWER SYSTEMS

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

#### BASES

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**BACKGROUND** Each diesel generator (DG) is provided with a 7-day storage tank, comprised of four inter-connected tanks, having a fuel oil capacity sufficient to operate that diesel for a period of 7 days while the DG is supplying maximum post loss of coolant accident load demand discussed in the UFSAR, Section 9.5.4.3 (Ref. 1) and Regulatory Guide 1.137 (Ref. 2). The maximum load demand assumed is the diesel generator operating at 110% rated load for the first two hours then operating at 100% rated load for the remaining 166 hours. 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 each diesel generator 7-day storage tank to two engine-mounted "day" tanks by either of two transfer pumps associated with the 7-day 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. Each 7-day tank is embedded in the concrete substructure below its respective DG with the two 550-gallon engine-mounted "day" tanks located in the respective DG room.

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

The DG lubrication system is designed to provide sufficient lubrication to permit proper operation of its associated DG under all loading conditions. The system is required to circulate the lube oil to the diesel engine working surfaces and to remove excess heat generated by friction during operation. Each engine oil sump contains an inventory capable of supporting a minimum of 7 days of operation without requiring replenishment. This supply is sufficient to allow the operator to replenish lube oil from outside sources.

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

BASES

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BACKGROUND (continued)

The Starting Air System for each diesel engine includes two tanks. The tanks are aligned in series to the starter motors. The upstream tank (Tank A) is maintained at 250-300 psig by the air compressor while the downstream tank (Tank B) is maintained at approximately 195 psig via a pressure control valve (PCV) between the two tanks.

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APPLICABLE  
SAFETY  
ANALYSES

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

Since diesel fuel oil, lube oil, and the air start subsystem support the operation of the standby AC power sources, they satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

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LCO

Stored diesel fuel oil is required to have sufficient supply for 7 days of full load operation. It is also required to meet specific standards for quality. Additionally, sufficient 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 engine-mounted "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 tank(s).

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APPLICABILITY

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

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## BASES

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### ACTIONS

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

#### A.1

In this Condition, the 7 day fuel oil supply for a DG is not available. However, the Condition is restricted to fuel oil level reductions that maintain at least a 6 day supply. The fuel oil level equivalent to a 6 day supply is 53719 gallons. 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

In this Condition, the 7 day lube oil inventory i.e., sufficient lubricating oil to support 7 days of continuous DG operation at full load conditions is not available. However, the Condition is restricted to lube oil volume reductions that maintain at least a 6 day supply. The lube oil inventory equivalent to a 6 day supply is 120 gallons (per diesel engine). This restriction allows sufficient time to obtain the requisite replacement volume. A period of 48 hours is considered sufficient to complete 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 3.8.3.3 for the stored fuel. Normally, trending of particulate levels allows sufficient time to correct high particulate levels prior to reaching the limit of acceptability. Poor sample procedures (bottom

## BASES

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### ACTIONS (continued)

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

#### E.1

With starting air Tank A pressure < 200 psig, sufficient capacity for five successive DG start attempts does not exist. However, as long as Tank B pressure is > 150 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 DG's 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.

BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.8.3.1

This SR provides verification that there is an adequate inventory of fuel oil in the storage tanks to support each DG's operation for 7 days at full load. The fuel oil level equivalent to a 7 day supply is 62,000 gallons when calculated in accordance with References 2 and 3. The required fuel storage volume is determined using the most limiting energy content of the stored fuel. Using the known correlation of diesel fuel oil absolute specific gravity or API gravity to energy content, the required diesel generator output, the corresponding fuel consumption rate, the onsite fuel storage volume required for 7 days of operation can be determined. SR 3.8.3.3 requires a new fuel to be tested to verify that the absolute specific gravity or API gravity is within the range assumed in the diesel fuel oil consumption calculations. 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 Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.8.3.2

This Surveillance ensures that sufficient lube oil inventory is available to support at least 7 days of full load operation for each DG. The lube oil inventory equivalent to a 7 day supply is 142 gallons (per diesel engine) and is based on the DG manufacturer consumption values for the run time of the DG.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

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SURVEILLANCE REQUIREMENTS (continued)

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 7-day storage tanks without concern for contaminating the entire volume of fuel oil in the 7-day storage tanks. These tests are to be conducted prior to adding the new fuel to the 7-day 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 D4057-1988 (Ref. 6),
- b. Verify in accordance with the tests specified in ASTM D975-1990 (Ref. 6) that the sample has an absolute specific gravity at 60/60°F of  $\geq 0.83$  and  $\leq 0.89$  or an API gravity at 60°F of  $\geq 27^\circ$  and  $\leq 39^\circ$  when tested in accordance with ASTM D1298-1985 (Ref. 6), a kinematic viscosity at 40°C of  $\geq 1.9$  centistokes and  $\leq 4.1$  centistokes, and a flash point of  $\geq 125^\circ\text{F}$ , and
- c. Verify that the new fuel oil has a clear and bright appearance with proper color when tested in accordance with ASTM D4176-2004 or a water and sediment content within limits when tested in accordance with ASTM D1796-1997 (Ref. 6).

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 7-day 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 D975-1990 (Ref. 7) are met for new fuel oil when tested in accordance with ASTM D975-1990 (Ref. 6), except that the analysis for sulfur may be performed in accordance with ASTM D1552-1990, ASTM D2622-1987, or ASTM D4294-2002 (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

## BASES

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### SURVEILLANCE REQUIREMENTS (continued)

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 D6217-11 (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. Each fuel oil storage tank (7-day 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.

#### 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. The pressure specified in this SR is intended to reflect the lowest value at which the five starts can be accomplished.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

#### 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 (7-day tanks) 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 Frequency is controlled under the Surveillance Frequency Control Program.

BASES

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REFERENCES

1. UFSAR, Section 9.5.4.3.
  2. Regulatory Guide 1.137, 1979.
  3. ANSI N195, 1976.
  4. UFSAR, Chapter 6.
  5. UFSAR, Chapter 15.
  6. ASTM Standards: D4057-1988; D975-1990; D1298-1985;  
D4176-2004; D1796-1997; D1552-1990; D2622-1987; D4294-2002;  
D6217-11.
  7. ASTM Standards, D975-1990, Table 1.
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## B 3.8 ELECTRICAL POWER SYSTEMS

### B 3.8.4 DC Sources - Operating

#### BASES

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**BACKGROUND** The DC electrical power system consists of the 125 V Vital DC electrical power subsystem and the diesel generator (DG) DC electrical power subsystem. The 125 V Vital DC electrical power subsystem 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 instrument power board power (via inverters). Control power and generator field flashing for each DG is provided by the DG DC electrical power subsystem. 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 V Vital DC electrical power subsystem consists of two independent and redundant safety related Class 1E DC electrical power trains (Train A is associated with channels I and III and Train B is associated with channels II and IV). Each train consists of two 125 VDC batteries, the associated battery charger(s) for each battery, and all the associated control equipment and interconnecting cabling.

The 125 V Vital DC electrical power subsystem has manual access to a fifth vital battery system. The fifth 125 VDC Vital Battery System is intended to serve as a replacement for any one of the four 125 VDC vital batteries during testing, maintenance, and outages with no loss of system reliability under any mode of operation. Additionally there is one spare battery charger for channels I and II and one spare charger for channels III and IV, 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 trains are maintained.

During normal operation, the 125 V Vital DC 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 Vital DC electrical power trains provide the control power for its associated Class 1E AC power load group, 6.9 kV Shutdown Boards, and 480 V load centers. The DC electrical power trains also provide DC electrical power to the inverters, which in turn power the AC vital instrument power boards.

## BASES

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### BACKGROUND (continued)

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

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

Each Vital battery has adequate storage capacity to meet the duty cycle(s) discussed in the UFSAR, Chapter 8 (Ref 4). The battery is designed with additional capacity above that required by the design duty cycle to allow for temperature variations and other factors.

The batteries for 125 V Vital DC electrical power trains are sized to produce required capacity at 82% of nameplate rating, corresponding to warranted capacity at end of life cycles and the 100% design demand. The minimum design voltage limit is 105 V.

The Vital battery cells are of flooded lead acid construction with a nominal specific gravity of 1.215. This specific gravity corresponds to an open circuit battery voltage of 123.78 V for a 60 cell battery (i.e., cell voltage of 2.063 volts per cell (Vpc)). The open circuit voltage is the voltage maintained when there is no charging or discharging. Optimal long term performance however, is obtained by maintaining a minimum float voltage of 129 V DC. This provides adequate over-potential, which limits the formation of lead sulfate and self discharge.

Each Vital DC electrical power train battery charger 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 excess capacity to restore the battery from the design minimum charge to its fully charged state within 12 hours (with accident loads being supplied) following a 30 minute AC power outage and in approximately 36 hours (with normal loads being supplied) following a 4 hour AC power outage.

Each vital battery charger is normally in the float-charge mode. Float-charge is the condition in which the charger is supplying the connected loads and the battery cells are receiving adequate current to optimally charge the battery. This assures the internal losses of a battery are overcome and the battery is maintained in a fully charged state.

BASES

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BACKGROUND (continued)

When desired, each Vital battery charger can be placed in the equalize mode. The equalize mode is at a higher voltage than the float mode and charging current is correspondingly higher. The battery charger is operated in the equalize mode after a battery discharge or for routine maintenance. Following a battery discharge, the battery recharge characteristic accepts current at the current limit of the battery charger (if the discharge was significant, e.g., following a battery service test) until the battery terminal voltage approaches the charger voltage setpoint. Charging current then reduces exponentially during the remainder of the recharge cycle. Lead-calcium batteries have recharge efficiencies of greater than 95%, so once at least 105% of the ampere-hours discharged have been returned, the battery capacity would be restored to the same condition as it was prior to the discharge. This can be monitored by direct observation of the exponentially decaying charging current or by evaluating the amp-hours discharged from the battery and amp-hours returned to the battery.

Control power for the DGs is provided by the four DG battery systems, one per DG. Each system is comprised of one 125 VDC battery, the associated charger for each battery, and all associated control equipment and interconnecting cabling.

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APPLICABLE  
SAFETY  
ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the UFSAR, Chapter 6 (Ref. 5) and Chapter 15 (Ref. 6), assume that Engineered Safety Feature (ESF) systems are OPERABLE. The DC electrical power system provides normal and emergency DC electrical power for the DGs, emergency auxiliaries, and control and switching during all MODES of operation.

The OPERABILITY of the DC sources is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the 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 10 CFR 50.36(c)(2)(ii).

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LCO

The Vital DC electrical power trains, each train consisting of two batteries, battery charger for each battery and the corresponding control equipment and interconnecting cabling supplying power to the associated board 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

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## BASES

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### LCO (continued)

postulated DBA. Loss of any DC electrical power train does not prevent the minimum safety function from being performed (Ref. 4).

The DG DC electrical power subsystems, each subsystem consisting of one battery, one battery charger and the corresponding control equipment and interconnecting cabling supplying power to the associated DG control circuit are required to be OPERABLE to ensure the availability of the required power to support the OPERABILITY of the diesel generator.

An OPERABLE Vital DC electrical power train requires all required batteries and respective chargers to be operating and connected to the associated DC board(s).

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### APPLICABILITY

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

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

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

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### ACTIONS

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

Condition A represents one train with one or two vital battery chargers inoperable (e.g., the voltage limit of SR 3.8.4.1 is not maintained). The ACTIONS provide a tiered response that focuses on returning the battery to the fully charged state and restoring a fully qualified charger to OPERABLE status in a reasonable time period. Required Action A.1 requires that the battery terminal voltage be restored to greater than or equal to the minimum established float voltage within 2 hours. This time provides for returning the inoperable charger to OPERABLE status or providing an alternate means of restoring battery terminal voltage to greater than or equal to the minimum established float voltage. Restoring the battery terminal voltage to greater than or equal to the minimum established float voltage provides good assurance that, within 12 hours, the battery will be restored to its fully charged condition (Required Action A.2) from any discharge that might have occurred due to the charger inoperability.

A discharged battery having terminal voltage of at least the minimum established float voltage indicates that the battery is on the exponential charging current portion (the second part) of its recharge cycle. The time

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## BASES

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### ACTIONS (continued)

to return a battery to its fully charged state under this condition is simply a function of the amount of the previous discharge and the recharge characteristic of the battery. Thus there is good assurance of fully recharging the battery within 12 hours, avoiding a premature shutdown with its own attendant risk.

If established battery terminal float voltage cannot be restored to greater than or equal to the minimum established float voltage within 2 hours, and the charger is not operating in the current-limiting mode, a faulty charger is indicated. A faulty charger that is incapable of maintaining established battery terminal float voltage does not provide assurance that it can revert to and operate properly in the current limit mode that is necessary during the recovery period following a battery discharge event that the DC system is designed for.

If the charger is operating in the current limit mode after 2 hours that is an indication that the battery is partially discharged and its capacity margins will be reduced. The time to return the battery to its fully charged condition in this case is a function of the battery charger capacity, the amount of loads on the associated DC system, the amount of the previous discharge, and the recharge characteristic of the battery. The charge time can be extensive, and there is not adequate assurance that it can be recharged within 12 hours (Required Action A.2).

Required Action A.2 requires that the battery float current be verified as less than or equal to 2 amps. This indicates that, if the battery had been discharged as the result of the inoperable battery charger, it is now fully capable of supplying the maximum expected load requirement. The 2 amp value is based on returning the battery to 98% charge and assumes a 5% design margin for the battery. If at the expiration of the initial 12 hour period the battery float current is not less than or equal to 2 amps this indicates there may be additional battery problems and the battery must be declared inoperable.

Required Action A.3 limits the restoration time for the inoperable vital battery charger to 7 days. This action is applicable if an alternate means of restoring battery terminal voltage to greater than or equal to the minimum established float voltage has been used (e.g., fifth battery charger). The 7 day Completion Time reflects a reasonable time to effect restoration of the qualified battery charger to OPERABLE status.

## BASES

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### ACTIONS (continued)

#### B.1

Condition B represents one vital DC 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 vital DC distribution subsystem.

If one of the required vital DC electrical power trains is inoperable for reasons other than Condition A (e.g., inoperable battery charger), the remaining vital DC electrical power train has the capacity to support a safe shutdown and to mitigate an accident condition. Since a subsequent worst case single failure could, however, result in the loss of the minimum necessary vital DC electrical trains to mitigate a worst case accident, continued power operation should not exceed 2 hours. The 2 hour Completion Time is based on Regulatory Guide 1.93 (Ref. 7) and reflects a reasonable time to assess unit status as a function of the inoperable vital DC electrical power train and, if the vital DC electrical power train is not restored to OPERABLE status, to prepare to effect an orderly and safe unit shutdown.

#### C.1 and C.2

If the inoperable vital DC electrical power train cannot be restored to OPERABLE status within the required Completion Time, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging plant systems. The Completion Time to bring the unit to MODE 5 is consistent with the time required in Regulatory Guide 1.93 (Ref. 7).

#### D.1

If the DG DC electrical power subsystem(s) is inoperable, the associated DG(s) may be incapable of performing their intended function and must be immediately declared inoperable. This declaration also requires entry into applicable Conditions and Required Actions for inoperable DG(s), LCO 3.8.1, "AC Sources – Operating."

## BASES

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### SURVEILLANCE REQUIREMENTS

#### SR 3.8.4.1

Verifying battery terminal voltage while on float charge for the batteries helps to ensure the effectiveness of the battery chargers, which support 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 and maintain the battery in a fully charged state while supplying the continuous steady state loads of the associated DC train or subsystem. On float charge, battery cells will receive adequate current to optimally charge the battery. The voltage requirements are based on the nominal design voltage of the battery and are consistent with the minimum float voltage established by the battery manufacturer (129 V for the Vital batteries and 124 V for the DG batteries). This voltage maintains the battery plates in a condition that supports maintaining the grid life.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

#### SR 3.8.4.2

This SR verifies the design capacity of the vital battery chargers. According to Regulatory Guide 1.32 (Ref. 8), the battery charger supply is recommended to be based on the largest combined demands of the various steady state loads and the charging capacity to restore the battery from the design minimum charge state to the fully charged state, irrespective of the status of the unit during these demand occurrences. The minimum required amperes and duration ensures that these requirements can be satisfied.

This SR provides two options. One option requires that each battery charger be capable of supplying 150 amps at the minimum established float voltage (129 V DC) for 4 hours. The ampere requirements are based on the output rating of the chargers. The voltage requirements are based on the charger voltage level after a response to a loss of AC power.

The other option requires that each vital battery charger be capable of recharging the battery after a service test coincident with supplying the largest coincident demands of the various continuous steady state loads (irrespective of the status of the plant during which these demands occur). This level of loading may not normally be available following the battery service test and will need to be supplemented with additional loads. The duration for this test may be longer than the charger sizing criteria since the battery recharge is affected by float voltage,

## BASES

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### SURVEILLANCE REQUIREMENTS (continued)

temperature, and the exponential decay in charging current. The battery is recharged when the measured charging current is  $\leq 2$  amps.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

#### SR 3.8.4.3

A battery service test is a special test of the battery capability, as found, to satisfy the design requirements (battery duty cycle (2 hours for Vital batteries)) 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 is controlled under the Surveillance Frequency Control Program.

This SR is modified by three Notes. Notes 1 and 2 allow the performance of a modified performance discharge test in lieu of a service test.

The reason for Note 3 is that performing the Surveillance on in-service vital batteries would perturb the electrical distribution system and challenge safety systems. Therefore, prior to performing a battery service test, the in-service vital battery to be tested is removed from service and the spare vital battery is placed in-service. Credit may be taken for unplanned events that satisfy this SR.

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### REFERENCES

1. 10 CFR 50, Appendix A, GDC 17.
  2. Regulatory Guide 1.6, March 10, 1971.
  3. IEEE-308-1971.
  4. UFSAR, Chapter 8.
  5. UFSAR, Chapter 6.
  6. UFSAR, Chapter 15.
  7. Regulatory Guide 1.93, December 1974.
  8. Regulatory Guide 1.32, February 1972.
  9. Regulatory Guide 1.129, December 1974.
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## B 3.8 ELECTRICAL POWER SYSTEMS

### B 3.8.5 DC Sources - Shutdown

#### BASES

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BACKGROUND	A description of the DC sources is provided in the Bases for LCO 3.8.4, "DC Sources - Operating."
APPLICABLE SAFETY ANALYSES	<p>The initial conditions of Design Basis Accident (DBA) and transient analyses in the UFSAR, Chapter 6 (Ref. 1) and Chapter 15 (Ref. 2), assume that Engineered Safety Feature systems are OPERABLE. The DC electrical power system provides normal and emergency DC electrical power for the diesel generators, emergency auxiliaries, and control and switching during all MODES of operation.</p> <p>The OPERABILITY of the DC subsystems is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY.</p> <p>The OPERABILITY of the minimum DC electrical power sources during MODES 5 and 6 and during movement of irradiated fuel assemblies ensures that:</p> <ol style="list-style-type: none"><li>The unit can be maintained in the shutdown or refueling condition for extended periods,</li><li>Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status, and</li><li>Adequate DC electrical power is provided to mitigate events postulated during shutdown, such as a fuel handling accident.</li></ol> <p>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 DBAs that are analyzed in MODES 1, 2, 3, and 4 have no specific analyses 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.</p>

BASES

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APPLICABLE SAFETY ANALYSES (continued)

The shutdown Technical Specification requirements are designed to ensure that the unit has the capability to mitigate the consequences of certain postulated accidents. Worst case DBAs which are analyzed for operating MODES are generally viewed not to be a significant concern during shutdown MODES due to the lower energies involved. The Technical Specifications therefore require a lesser complement of electrical equipment to be available during shutdown than is required during operating MODES. More recent work completed on the potential risks associated with shutdown, however, have found significant risk associated with certain shutdown evolutions. As a result, in addition to the requirements established in the Technical Specifications, the industry has adopted NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management," as an Industry initiative to manage shutdown tasks and associated electrical support to maintain risk at an acceptable low level. This may require the availability of additional equipment beyond that required by the shutdown Technical Specifications.

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

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LCO

One vital DC electrical power train consists of two channels. Train A consists of channels I and III and Train B consists of channels II and IV. The required train consisting of two batteries, one battery charger per battery, and the corresponding control equipment and interconnecting cabling within the train, is required to be OPERABLE. 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).

Control power for the DGs is provided by four DG DC electrical power subsystems, one per DG. Each DG DC electrical power subsystem is comprised of one 125 VDC battery, an associated charger, and associated control equipment and interconnecting cabling. One DG DC electrical power subsystem is required to be OPERABLE for each required DG.

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APPLICABILITY

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

- a. Required features to provide adequate coolant inventory makeup are available for the irradiated fuel assemblies in the core,
  - b. Required features needed to mitigate a fuel handling accident are available,
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## BASES

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### APPLICABILITY (continued)

- c. Required features necessary to mitigate the effects of events that can lead to core damage during shutdown are available, and
- d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

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

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### ACTIONS

LCO 3.0.3 is not applicable while in MODE 5 or 6. However, since irradiated fuel assembly movement can occur in MODE 1, 2, 3, or 4, the ACTIONS have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operations. Entering LCO 3.0.3, while in MODE 1, 2, 3, or 4 would require the unit to be shutdown unnecessarily.

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

With the required train of DC electrical power sources inoperable, the minimum required DC electrical power source is not available. Therefore, the allowance for sufficiently conservative actions is made (i.e., to suspend movement of irradiated fuel assemblies, and operations involving positive reactivity additions) that could result in loss of required SDM (MODE 5) or boron concentration (MODE 6). Suspending positive reactivity additions that could result in failure to meet the minimum SDM or boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than that what would be required in the RCS for minimum SDM or refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation. Introduction of temperature changes including temperature increases when operating with a positive MTC must also be evaluated to ensure they do not result in a loss of required SDM.

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 train and to continue this action until restoration is accomplished in order to provide the necessary DC electrical power to the unit safety systems.

BASES

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ACTIONS (continued)

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.

B.1

If one or more DG DC electrical power subsystems are inoperable, the associated DGs may be incapable of performing their intended function and must be immediately declared inoperable. This declaration also requires entry into the applicable Conditions and Required Actions for inoperable DGs, LCO 3.8.2, "AC Sources – Shutdown."

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SURVEILLANCE  
REQUIREMENTS

SR 3.8.5.1

SR 3.8.5.1 requires performance of all Surveillances required by SR 3.8.4.1 through SR 3.8.4.3. Therefore, see the corresponding Bases for LCO 3.8.4 for a discussion of each SR.

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REFERENCES

1. UFSAR, Chapter 6.
  2. UFSAR, Chapter 15.
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## B 3.8 ELECTRICAL POWER SYSTEMS

### B 3.8.6 Battery Parameters

#### BASES

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**BACKGROUND** This LCO delineates the limits on battery float current as well as electrolyte temperature, level, and float voltage for the Vital and diesel generator (DG) 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." In addition to the limitations of this Specification, the Battery Monitoring and Maintenance Program also implements a program specified in Specification 5.5.15 for monitoring various battery parameters.

The Vital battery cells are of flooded lead acid construction with a nominal specific gravity of 1.215. This specific gravity corresponds to an open circuit battery voltage of approximately 123.78 V for 60 cell battery (i.e., cell voltage of 2.063 volts per cell (Vpc)). The open circuit voltage is the voltage maintained when there is no charging or discharging. Optimal long term performance however, is obtained by maintaining a float voltage 2.17 Vpc. This provides adequate over-potential which limits the formation of lead sulfate and self discharge.

The DG battery cells are of flooded lead acid construction with a nominal specific gravity of 1.215. Each DG battery consists of 58 cells; however, a battery is considered OPERABLE with 57 cells if one is strapped out. Optimal long term performance is obtained by maintaining a float voltage of 2.20 to 2.25 Vpc. This provides adequate over-potential which limits the formation of lead sulfate and self-discharge.

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**APPLICABLE SAFETY ANALYSES** The initial conditions of Design Basis Accident (DBA) and transient analyses in the UFSAR, Chapter 6 (Ref. 3) and Chapter 15 (Ref. 4), 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 parameters satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

BASES

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LCO Battery 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. Battery parameter limits are conservatively established, allowing continued DC electrical system function even with limits not met. Additional preventative maintenance, testing, and monitoring performed in accordance with the Battery Monitoring and Maintenance Program is conducted as specified in Specification 5.5.15.

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

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

With one or more cells in one or more batteries < 2.07 V, the battery cell is degraded. Within 2 hours verification of the required battery charger OPERABILITY is made by monitoring the battery terminal voltage (SR 3.8.4.1) and of the overall battery state of charge by monitoring the battery float charge current (SR 3.8.6.1). This assures that there is still sufficient battery capacity to perform the intended function. Therefore, the affected battery is not required to be considered inoperable solely as a result of one or more cells in one or more batteries < 2.07 V, and continued operation is permitted for a limited period up to 24 hours.

Since the Required Actions only specify "perform," a failure of SR 3.8.4.1 or SR 3.8.6.1 acceptance criteria does not result in this Required Action not met. However, if one of the SRs is failed the appropriate Condition(s), depending on the cause of the failures, is entered. If SR 3.8.6.1 is failed then there is no assurance that there is still sufficient battery capacity to perform the intended function and the battery must be declared inoperable immediately.

B.1 and B.2

One or more vital batteries with float current > 2 amps or one or more DG batteries with float current > 1 amp indicates that a partial discharge of the battery capacity has occurred. This may be due to a temporary loss of a battery charger or possibly due to one or more battery cells in a low voltage condition reflecting some loss of capacity. Within 2 hours verification of the required battery charger OPERABILITY is made by monitoring the battery terminal voltage. If the terminal voltage is found to be less than the minimum established float voltage there are two possibilities, the battery charger is inoperable or is operating in the

## BASES

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### ACTIONS (continued)

current limit mode. Condition A addresses charger inoperability. If the charger is operating in the current limit mode after 2 hours that is an indication that the battery has been substantially discharged and likely cannot perform its required design functions. The time to return the battery to its fully charged condition in this case is a function of the battery charger capacity, the amount of loads on the associated DC system, the amount of the previous discharge, and the recharge characteristic of the battery. The charge time can be extensive, and there is not adequate assurance that it can be recharged within 12 hours (Required Action B.2.1 or B.2.2). The battery must therefore be declared inoperable.

If the float voltage is found to be satisfactory but there are one or more battery cells with float voltage less than 2.07 V, the associated "OR" statement in Condition F is applicable and the battery must be declared inoperable immediately. If float voltage is satisfactory and there are no cells less than 2.07 V there is good assurance that, within 12 hours, the battery will be restored to its fully charged condition (Required Action B.2.1 or B.2.2) from any discharge that might have occurred due to a temporary loss of the battery charger.

A discharged battery with float voltage (the charger setpoint) across its terminals indicates that the battery is on the exponential charging current portion (the second part) of its recharge cycle. The time to return a battery to its fully charged state under this condition is simply a function of the amount of the previous discharge and the recharge characteristic of the battery. Thus there is good assurance of fully recharging the battery within 12 hours, avoiding a premature shutdown with its own attendant risk.

If the condition is due to one or more cells in a low voltage condition but still greater than 2.07 V and float voltage is found to be satisfactory, this is not indication of a substantially discharged battery and 12 hours is a reasonable time prior to declaring the battery inoperable.

Since Required Action B.1 only specifies "perform," a failure of SR 3.8.4.1 acceptance criteria does not result in the Required Action not met. However, if SR 3.8.4.1 is failed, the appropriate Condition(s), depending on the cause of the failure, is entered.

## BASES

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### ACTIONS (continued)

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

With one or more batteries with one or more cells electrolyte level above the top of the plates, but below the minimum established design limits, the battery still retains sufficient capacity to perform the intended function. Therefore, the affected battery is not required to be considered inoperable solely as a result of electrolyte level not met. Within 31 days the minimum established design limits for electrolyte level must be re-established.

With electrolyte level below the top of the plates there is a potential for dryout and plate degradation. Required Actions C.1 and C.2 address this potential (as well as provisions in Specification 5.5.15, Battery Monitoring and Maintenance Program). They are modified by a Note that indicates they are only applicable if electrolyte level is below the top of the plates. Within 8 hours level is required to be restored to above the top of the plates. The Required Action C.2 requirement to verify that there is no leakage by visual inspection and the Specification 5.5.15.b item to initiate action to equalize and test in accordance with manufacturer's recommendation are taken from IEEE Standard 450. They are performed following the restoration of the electrolyte level to above the top of the plates. Based on the results of the manufacturer's recommended testing the batteries may have to be declared inoperable and the affected cells replaced.

#### D.1

With one or more batteries with pilot cell temperature less than the minimum established design limits, 12 hours is allowed to restore the temperature to within limits. A low electrolyte temperature limits the current and power available. Since the battery is sized with margin, while battery capacity is degraded, sufficient capacity exists to perform the intended function and the affected battery is not required to be considered inoperable solely as a result of the pilot cell temperature not met.

#### E.1

With one or more batteries in redundant trains with battery parameters not within limits there is not sufficient assurance that battery capacity has not been affected to the degree that the batteries can still perform their required function, given that redundant batteries are involved. With redundant batteries involved this potential could result in a total loss of function on multiple systems that rely upon the batteries. The longer Completion Times specified for battery parameters on non-redundant

BASES

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ACTIONS (continued)

batteries not within limits are therefore not appropriate, and the parameters must be restored to within limits on at least one train within 2 hours.

E.1

With one or more batteries with any battery parameter outside the allowances of the Required Actions for Condition A, B, C, D, or E, sufficient capacity to supply the maximum expected load requirement is not assured and the corresponding battery must be declared inoperable. Additionally, discovering one or more vital batteries with one or more battery cells float voltage less than 2.07 V and float current greater than 2 amps indicates that the battery capacity may not be sufficient to perform the intended functions. Similarly, discovering one or more DG batteries with one or more battery cells float voltage less than 2.07 V and float current greater than 1 amp indicates that the battery capacity may not be sufficient to perform the intended functions. The associated vital or DG battery must therefore be declared inoperable. In addition, if SR 3.8.6.6 or SR 3.8.6.7 are not met, the associated vital or DG battery is declared inoperable.

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SURVEILLANCE  
REQUIREMENTS

SR 3.8.6.1

Verifying battery float current while on float charge is used to determine the state of charge of the battery. Float charge is the condition in which the charger is supplying the continuous charge required to overcome the internal losses of a battery and maintain the battery in a charged state. The float current requirements are based on the float current indicative of a charged battery. Use of float current to determine the state of charge of the battery is consistent with IEEE-450 (Ref. 1). The minimum required procedural time to measure battery float current will be 30 seconds or as recommended by the float current measurement instrument manufacturer. The minimum float current measurement time is required to provide a more accurate battery float current reading.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

This SR is modified by a Note that states the float current requirement is not required to be met when battery terminal voltage is less than the minimum established float voltage of SR 3.8.4.1. When this float voltage is not maintained the Required Actions of LCO 3.8.4 ACTION A are being taken, which provide the necessary and appropriate verifications of the battery condition. Furthermore, the float current limit of 2 amps is

## BASES

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### SURVEILLANCE REQUIREMENTS (continued)

established based on the nominal float voltage value and is not directly applicable when this voltage is not maintained.

#### SR 3.8.6.2 and SR 3.8.6.5

Optimal long term battery performance is obtained by maintaining a float voltage greater than or equal to the minimum established design limits provided by the battery manufacturer, which corresponds to 129 V for vital batteries and 124 V for DG batteries at the battery terminals. This provides adequate over-potential, which limits the formation of lead sulfate and self discharge, which could eventually render the battery inoperable. Float voltages in this range or less, but greater than 2.07 Vpc, are addressed in Specification 5.5.15. SRs 3.8.6.2 and 3.8.6.5 require verification that the cell float voltages are equal to or greater than the short term absolute minimum voltage of 2.07 V.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

#### SR 3.8.6.3

The limit specified for electrolyte level ensures that the plates suffer no physical damage and maintains adequate electron transfer capability.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

#### SR 3.8.6.4

This Surveillance verifies that the pilot cell temperature is greater than or equal to the minimum established design limit (i.e., 60°F). Pilot cell electrolyte temperature is maintained above this temperature to assure the battery can provide the required current and voltage to meet the design requirements. Temperatures lower than assumed in battery sizing calculations act to inhibit or reduce battery capacity.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

#### SR 3.8.6.5

See SR 3.8.6.2 Bases

BASES

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.6.6 and SR 3.8.6.7

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.

Either the battery performance discharge test or the modified performance discharge test is acceptable for satisfying SR 3.8.6.6; however, only the modified performance discharge test may be used to satisfy the battery service test requirements of SR 3.8.4.3.

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.

It may consist of just two rates; for instance the one minute rate 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 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 must 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.

The acceptance criteria for this Surveillance are consistent with IEEE-450 (Ref. 1) and IEEE-485 (Ref. 5). These references recommend that the battery be replaced if its capacity is below 80% of the manufacturer's rating. A capacity of 80% shows that the battery rate of deterioration is increasing, even if there is ample capacity to meet the load requirements. Furthermore, the battery is sized to meet the assumed duty cycle loads when the battery design capacity reaches this 80% limit. The minimum battery capacity for the vital batteries has been raised from 80% to 82% to allow for possible discharge during the 5-minute delay associated with the Diesel Generator Start and Load Shed Timer.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

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SURVEILLANCE REQUIREMENTS (continued)

If the battery shows degradation, or if the battery has reached 85% of its expected life and capacity is < 100% of the manufacturer's rating, the Surveillance Frequency is reduced to 12 months. However, if the battery shows no degradation but has reached 85% of its expected life, the Surveillance Frequency is only reduced to 24 months for batteries that retain capacity  $\geq$  100% of the manufacturer's ratings. Degradation is indicated, according to IEEE-450 (Ref. 1), when the battery capacity drops by more than 10% relative to its capacity on the previous performance test or when it is  $\geq$  10% below the manufacturer's rating. These Frequencies are consistent with the recommendations in IEEE-450 (Ref. 1).

SR 3.8.6.6 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. This restriction from normally performing the Surveillance in MODE 1, 2, 3, or 4 is further amplified to allow portions of the Surveillance to be performed for the purpose of reestablishing OPERABILITY (e.g., post work testing following corrective maintenance, corrective modification, deficient or incomplete surveillance testing, and other unanticipated OPERABILITY concerns) provided an assessment determines plant safety is maintained or enhanced. This assessment shall, as a minimum, consider the potential outcomes and transients associated with a failed partial Surveillance, a successful partial Surveillance, and a perturbation of the offsite or onsite system when they are tied together or operated independently for the partial Surveillance; as well as the operator procedures available to cope with these outcomes. These shall be measured against the avoided risk of a plant shutdown and startup to determine that plant safety is maintained or enhanced when portions of the Surveillance are performed in MODE 1 or 2. Risk insights or deterministic methods may be used for the assessment. Credit may be taken for unplanned events that satisfy this SR.

SR 3.8.6.7 is modified by a Note stating that credit may be taken for unplanned events that satisfy this SR.

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REFERENCES

1. IEEE-450-2002.
  2. UFSAR, Chapter 8.
  3. UFSAR, Chapter 6.
  4. UFSAR, Chapter 15.
  5. IEEE-485-1983, June 1983.
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## B 3.8 ELECTRICAL POWER SYSTEMS

### B 3.8.7 Inverters - Operating

#### BASES

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**BACKGROUND** The inverters are the preferred source of power for the AC vital instrument power boards because of the stability and reliability they achieve. There are two unit inverters and one spare inverter per channel, each capable of supplying its associated AC vital instrument power boards, making a total of twelve inverters. Inverters 1-I and 2-I are connected to DC Channel I, inverters 1-II and 2-II are connected to DC Channel II, inverters 1-III and 2-III are connected to DC Channel III, and inverters 1-IV and 2-IV are connected to DC Channel IV. The spare inverter for a specified channel may be substituted for one of the two inverters of the same channel. The function of the inverter is to provide AC electrical power to the vital instrument power boards. The inverters can be powered from an internal AC source/rectifier or from the station battery. The station batteries provide 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 UFSAR, Chapter 8 (Ref. 1).

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**APPLICABLE SAFETY ANALYSES** The initial conditions of Design Basis Accident (DBA) and transient analyses in the UFSAR, 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 instrument power boards 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 10 CFR 50.36(c)(2)(ii).

## BASES

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### LCO

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

Maintaining the required inverters OPERABLE ensures that the redundancy incorporated into the design of the RPS and ESFAS instrumentation and controls is maintained. The eight inverters (two per channel) ensure an uninterruptible supply of AC electrical power to the AC vital instrument power boards even if the 480 volt safety boards are de-energized.

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

This LCO is modified by a Note that allows two inverters to be disconnected from a common battery for  $\leq 24$  hours, if the vital instrument power board(s) is powered from an inverter using internal AC source during the period and the remaining required inverters are OPERABLE. This allows an equalizing charge to be placed on one battery. If the inverters were not disconnected, the resulting voltage condition might damage the inverters. 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 instrument power board 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 remaining required inverters must be aligned to their associated batteries, regardless of the number of inverters or unit design.

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### APPLICABILITY

The inverters are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure that:

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

BASES

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APPLICABILITY (continued)

Inverter requirements for MODES 5 and 6 are covered in the Bases for LCO 3.8.8, "Inverters - Shutdown."

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ACTIONS

A.1

With a required inverter inoperable, its associated AC vital instrument power board becomes inoperable until it is manually re-energized from its inverter using internal AC source.

For this reason a Note has been included in Condition A requiring the entry into the Conditions and Required Actions of LCO 3.8.9, "Distribution Systems - Operating." This ensures that the vital instrument power board is re-energized within 8 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 instrument power board 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 instrument power board is the preferred source for powering instrumentation trip setpoint devices.

B.1 and B.2

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

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SURVEILLANCE  
REQUIREMENTS

SR 3.8.7.1

This Surveillance verifies that the inverters are functioning properly with all required circuit breakers closed and AC vital instrument power boards 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 instrument power boards.

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BASES

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SURVEILLANCE REQUIREMENTS (continued)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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REFERENCES

1. UFSAR, Chapter 8.
  2. UFSAR, Chapter 6.
  3. UFSAR, Chapter 15.
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## B 3.8 ELECTRICAL POWER SYSTEMS

### B 3.8.8 Inverters - Shutdown

#### BASES

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BACKGROUND	A description of the inverters is provided in the Bases for LCO 3.8.7, "Inverters - Operating."
APPLICABLE SAFETY ANALYSES	<p>The initial conditions of Design Basis Accident (DBA) and transient analyses in the UFSAR, 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.</p> <p>The OPERABILITY of the inverters is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY.</p> <p>The OPERABILITY of the minimum inverters to each AC vital board during MODES 5 and 6 ensures that:</p> <ol style="list-style-type: none"><li>The unit can be maintained in the shutdown or refueling condition for extended periods,</li><li>Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status, and</li><li>Adequate power is available to mitigate events postulated during shutdown, such as a fuel handling accident.</li></ol> <p>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 DBAs that are analyzed in MODES 1, 2, 3, and 4 have no specific analyses 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.</p>

BASES

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APPLICABLE SAFETY ANALYSES (continued)

The shutdown Technical Specification requirements are designed to ensure that the unit has the capability to mitigate the consequences of certain postulated accidents. Worst case DBAs which are analyzed for operating MODES are generally viewed not to be a significant concern during shutdown MODES due to the lower energies involved. The Technical Specifications therefore require a lesser complement of electrical equipment to be available during shutdown than is required during operating MODES. More recent work completed on the potential risks associated with shutdown, however, have found significant risk associated with certain shutdown evolutions. As a result, in addition to the requirements established in the Technical Specifications, the industry has adopted NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management," as an Industry initiative to manage shutdown tasks and associated electrical support to maintain risk at an acceptable low level. This may require the availability of additional equipment beyond that required by the shutdown Technical Specifications.

The inverters were previously identified as part of the distribution system and, as such, satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

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LCO

The inverters ensure the availability of electrical power for the instrumentation for systems required to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence or a postulated DBA. The battery powered inverters provide an uninterruptible supply of AC electrical power to the AC vital boards even if the 480 volt shutdown boards are de-energized. OPERABILITY of the inverters requires that each required vital board is powered by an inverter. This ensures the availability of sufficient inverter power sources to operate the unit in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents).

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APPLICABILITY

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

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

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### APPLICABILITY (continued)

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

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### ACTIONS

LCO 3.0.3 is not applicable while in MODE 5 or 6. However, since irradiated fuel assembly movement can occur in MODE 1, 2, 3, or 4, the ACTIONS have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operations. Entering LCO 3.0.3, while in MODE 1, 2, 3, or 4 would require the unit to be shutdown unnecessarily.

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

With one or more required inverters inoperable, the minimum required vital AC electrical power source is not available. Therefore, the allowance for sufficiently conservative actions is made (i.e., to suspend movement of irradiated fuel assemblies, and operations involving positive reactivity additions) that could result in loss of required SDM (MODE 5) or boron concentration (MODE 6). Suspending positive reactivity additions that could result in failure to meet the minimum SDM or boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than that what would be required in the RCS for minimum SDM or refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation. Introduction of temperature changes including temperature increases when operating with a positive MTC must also be evaluated to ensure they do not result in a loss of required SDM.

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.

BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.8.8.1

This Surveillance verifies that the inverters are functioning properly with all required circuit breakers closed and AC vital boards energized from the inverter. The verification of proper voltage and frequency output ensures that the required power is readily available for the instrumentation connected to the AC vital boards.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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REFERENCES

1. UFSAR, Chapter 6.
  2. UFSAR, Chapter 15.
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## B 3.8 ELECTRICAL POWER SYSTEMS

### B 3.8.9 Distribution Systems - Operating

#### BASES

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**BACKGROUND** The two units share several structures and systems including the preferred and emergency (standby) electric power systems (UFSAR, Chapter 8.0). The vital DC Power System is shared to the extent that a few loads (e.g., the vital inverters) in one nuclear unit are energized by the DC power channels assigned primarily to power loads of the other unit. In no case does the sharing inhibit the safe shutdown of one unit while the other unit is experiencing an accident. The Standby Power System serving each unit is divided into two redundant load groups (power trains). These power trains (Train A and Train B for each unit) supply power to safety-related equipment. Generally, the Engineered Safety Feature (ESF) loads assigned to a unit are supplied by the unit designated trains. For example, Safety Injection (SI) pump 1A-A (associated with Unit 1) is supplied by Shutdown Board 1A-A (also associated with Unit 1) while SI pump 2A-A (associated with Unit 2) is supplied by Shutdown Board 2A-A (also associated with Unit 2).

Separate and similar systems and equipment are provided for each unit when required. In certain instances, both units share systems or some components of a system. Shared systems are the exception to the unit/power system association. Because both units share the power system, one unit's power system(s) supports certain components required by the other unit (e.g., emergency gas treatment system).

The onsite Class 1E AC, vital DC, DG DC, and AC vital instrument electrical power distribution systems are divided into two redundant and independent trains. Each electrical power distribution train consists of:

- a. an AC electrical power distribution subsystem,
- b. an AC vital instrument power distribution subsystem,
- c. a vital DC electrical power distribution subsystem, and
- d. a diesel generator (DG) DC electrical power distribution subsystem.

Each AC electrical power subsystem consists of two 6.9 kV Shutdown Boards and four 480 V Shutdown Boards. Each train of 6.9 kV Shutdown Boards has two separate and independent offsite sources of power as well as a dedicated onsite diesel generator (DG) source for each 6.9 kV Shutdown Board. Each 6.9 kV Shutdown Board is normally connected to a preferred offsite source. If the offsite sources are unavailable, the onsite emergency DGs supply power to the affected 6.9 kV Shutdown Boards.

## BASES

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### BACKGROUND (continued)

Control power for the 6.9 kV Shutdown Board 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."

Each 120 VAC vital instrument power distribution subsystem consists of two Unit 1 120 V AC vital instrument power boards and two Unit 2 120 V AC vital instrument power boards and are powered from the inverters.

Each 125 V vital DC electrical power distribution subsystem consists of two 125 V boards. The 125 V vital DC electrical power distribution subsystem has manual access to a fifth vital battery. The fifth vital battery is intended to serve as a replacement for any one of the four 125 V DC vital batteries with no loss of system reliability under any mode of operation.

Each DG DC electrical power distribution subsystem consists of two 125 V DC distribution panels that supply power to the respective DG's auxiliary loads. During normal operation, power is supplied to the distribution panel by a 480 VAC board through a battery charger. During emergency operation of the DG (loss of offsite power source), the distribution panel is supplied power from a dedicated battery. This panel supplies power for DG control, protection, and the engine DC lube oil circulating pump.

The list of electrical power boards and distribution panels required for the AC, vital DC, DG DC, and AC vital instrument electrical power distribution subsystems is presented in Table B 3.8.9-1.

Associated with each board listed in Table B 3.8.9-1 are a number of safety significant electrical loads. When one or more of the boards specified in Table B 3.8.9-1 becomes inoperable, entry into the appropriate ACTIONS of LCO 3.8.9 is required. Some boards, distribution panels, and motor control centers (MCCs), which help comprise the AC and DC electrical power distribution subsystems, are not listed in Table B 3.8.9-1. The loss of electrical loads associated with these boards, panels, or MCCs may not result in a complete loss of a safety function necessary to shut down the reactor and maintain it in a safe condition. Therefore, should one or more of these boards, panels, or MCCs become inoperable due to a failure not affecting the OPERABILITY of a board listed in Table B 3.8.9-1 (e.g., a breaker supplying a single distribution panel fails open), the individual loads associated with the board, panel, or MCC are declared inoperable, and the appropriate Conditions and Required Actions of the LCOs governing the individual loads are entered.

## BASES

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### APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the UFSAR, Chapter 6 (Ref. 1), and in the UFSAR, Chapter 15 (Ref. 2), assume ESF systems are OPERABLE. The AC, vital DC, DG DC, and AC vital instrument 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, vital DC, DG DC, and AC vital instrument 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 electrical 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 electrical power distribution systems satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

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### LCO

The required electrical power distribution subsystems listed in Table B 3.8.9-1 ensure the availability of AC, vital DC, DG DC, and AC vital instrument 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. Two electrical power distribution trains are required to be OPERABLE. Each train includes:

- a. an AC electrical power distribution subsystem (i.e., one Unit 1 6.9 kV shutdown board, one Unit 2 6.9 kV shutdown board, and associated 480 V shutdown boards),
  - b. an AC vital instrument power distribution subsystem (i.e., two Unit 1 120 V AC instrument power boards and two Unit 2 120 V AC instrument power boards),
  - c. a vital DC electrical power distribution subsystem (i.e., two 125 V DC boards), and
  - d. a DG DC electrical power distribution subsystem (i.e., two 125 V DG distribution panels).
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## BASES

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LCO (continued) Maintaining two electrical power distribution trains 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 trains will not prevent safe shutdown of the reactor.

OPERABLE AC electrical power distribution subsystems require the associated boards to be energized to their proper voltages. OPERABLE vital DC and DG DC electrical power distribution subsystems require the associated boards and distribution panels, as applicable, to be energized to their proper voltage from either the associated battery or charger. OPERABLE AC vital instrument power distribution subsystems require the associated boards to be energized to their proper voltage from the associated inverter via inverted DC voltage, or inverter using internal 120 volt regulated AC source.

In addition, tie breakers between redundant safety related AC, vital DC, DG DC, and AC vital instrument electrical power distribution subsystems, if they exist, must be open. This prevents any electrical malfunction in any power distribution subsystem from propagating to the redundant subsystem, that could cause the failure of a redundant subsystem and a loss of essential safety function(s). If any tie breakers are closed, the affected redundant electrical power distribution subsystems are considered inoperable. This applies to the onsite, safety related redundant electrical power distribution subsystems. It does not, however, preclude redundant Class 1E 6.9 kV Shutdown Boards from being powered from the same offsite circuit.

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

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

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

BASES

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ACTIONS

A.1

With one or more AC electrical power distribution subsystems (except AC vital instrument boards), inoperable due to one or more inoperable Unit 2 AC shutdown boards, and a loss of function has not occurred, the remaining portions of the AC 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 portions of the power distribution subsystems could result in the minimum required ESF functions not being supported. Therefore, the Unit 2 AC electrical distribution subsystems must be restored to OPERABLE status within 8 hours.

Condition A worst case scenario is one train of Unit 2 boards 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 Unit 2 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.

Required Action A.1 is modified by a Note that requires the applicable Conditions and Required Actions of LCO 3.8.4, "DC Sources - Operating," to be entered for vital DC electrical power trains made inoperable by inoperable AC electrical power distribution subsystems. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components. Inoperability of one or more AC electrical power distribution subsystems can result in loss of charging power to batteries and eventual loss of DC power. This Note ensures that the appropriate attention is given to restoring charging power to batteries, if necessary, after loss of distribution subsystems.

## BASES

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### ACTIONS (continued)

#### B.1

With one or more AC vital instrument power distribution subsystems inoperable, and a loss of function has not yet occurred, the remaining OPERABLE portions of the AC vital instrument power distribution subsystems 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 AC vital instrument power distribution subsystem(s) must be restored to OPERABLE status within 8 hours by powering the affected subsystems from the associated inverter via inverted DC or inverter using internal 120 volt regulated AC source.

Condition B represents one or more AC vital instrument power distribution subsystems 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 instrument power boards and restoring power to the affected vital instrument power boards.

This 8 hour limit is more conservative than Completion Times allowed for the vast majority of components that are without adequate vital AC power. Taking exception to LCO 3.0.2 for components without adequate vital AC power, that would have the Required Action Completion Times shorter than 8 hours if declared inoperable, is acceptable because of:

- 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 8 hour Completion Time takes into account the importance to safety of restoring the AC vital instrument power distribution subsystems to OPERABLE status, the redundant capability afforded by the remaining

## BASES

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### ACTIONS (continued)

OPERABLE AC vital instrument power boards, and the low probability of a DBA occurring during this period.

#### C.1

With one or more vital DC electrical power distribution subsystems inoperable, and a loss of function has not yet occurred, the remaining portions of the vital 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 portions of the vital DC electrical power distribution subsystems could result in the minimum required ESF functions not being supported. Therefore, the vital DC subsystems must be restored to OPERABLE status within 2 hours by powering the subsystem from the associated battery or charger.

Condition C represents one or more vital DC electrical power distribution subsystems 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 panels and restoring power to the affected panels.

This 2 hour limit is more conservative than Completion Times allowed for the vast majority of components that 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 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 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 vital DC electrical power distribution subsystems is consistent with Regulatory Guide 1.93 (Ref. 3).

## BASES

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### ACTIONS (continued)

#### D.1

With one or more AC electrical power distribution subsystems (except AC vital instrument boards) inoperable due to one or more inoperable Unit 1 AC shutdown boards and a loss of function has not occurred, the remaining AC 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 power distribution subsystems could result in the minimum required ESF functions not being supported. Therefore, the associated required feature(s) must be declared inoperable immediately and the corresponding Condition(s) entered. The Required Action(s) of these Condition(s) will determine the impact of the inoperable Unit 1 AC shutdown board(s).

Condition D is modified by two notes that limit the conditions and parameters that allow entry into Condition D. The first note states that Condition D is only applicable during planned maintenance. This will allow the plant configuration to be aligned to minimize features being inoperable when the opposite unit shutdown board is made inoperable. The second note limits the applicability of Condition D to the time period when the opposite unit is either defueled or in MODE 6 following defueled with refueling water cavity level  $\geq 23$  ft. above the top of the reactor vessel flange. This note limits the time period allowing Condition D to be entered, minimizing when the allowance can be utilized. The allowance for Condition D is acceptable based on the following:

- a. The opposite unit's AC shutdown boards are not as critical to the operating unit (fewer operating unit loads) as the operating unit's AC shutdown boards.
- b. Performing maintenance on these components will increase the reliability of the Class 1E AC Electrical Power Distribution System.
- c. The Required Actions associated with the features declared inoperable provide compensatory measures during the performance of the planned maintenance.
- d. The limited opportunities that allow the planned maintenance to occur.

During the planned maintenance of the Unit 1 AC shutdown boards, if a condition is discovered on these boards requiring corrective maintenance, this maintenance may be performed under Condition D.

## BASES

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### ACTIONS (continued)

#### E.1

With one or more AC electrical power distribution subsystems (except AC vital instrument boards) inoperable due to one or more inoperable Unit 1 AC shutdown boards for reasons other than Condition D and a loss of function has not occurred, the remaining AC 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 power distribution subsystems could result in the minimum required ESF functions not being supported. Therefore, the inoperable Unit 1 AC electrical power distribution subsystem(s) must be returned to OPERABLE status within 24 hours. The 24 hour time limit before requiring a unit shutdown in this Condition is acceptable because the opposite unit's AC shutdown boards are not as critical to the operating unit (fewer operating unit loads) as the operating unit's AC shutdown boards.

#### F.1

With one or more DG DC electrical power distribution panels inoperable there is no longer assurance the supported DG(s) is able to start and perform its necessary safety function. The affected DG(s) must therefore be declared inoperable immediately and the corresponding Condition(s) entered.

#### G.1 and G.2

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

#### H.1

Condition H corresponds to a level of degradation in the electrical power distribution system that results in a loss of safety function. When more than one inoperable electrical power distribution subsystem results in the loss of a safety function, the plant is in a condition outside the accident analysis. Therefore, no additional time is justified for continued operation. LCO 3.0.3 must be entered immediately to commence a controlled shutdown.

BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.8.9.1

This Surveillance verifies that the required AC, vital DC, DG DC, and AC vital instrument electrical power distribution subsystems are functioning properly, with the correct circuit breaker alignment. The correct breaker alignment ensures the appropriate separation and independence of the electrical power distribution trains is maintained, and the appropriate voltage is available to each required board. The verification of proper voltage availability on the boards ensures that the required voltage is readily available for motive as well as control functions for critical system loads connected to these boards.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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REFERENCES

1. UFSAR, Chapter 6.
  2. UFSAR, Chapter 15.
  3. Regulatory Guide 1.93, December 1974.
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Table B 3.8.9-1 (page 1 of 1)  
AC and DC Electrical Power Distribution Systems

TYPE	VOLTAGE (nominal)	SR 3.8.9.1 Voltage Range	TRAIN A SUBSYSTEMS		TRAIN B SUBSYSTEMS	
			<u>Unit 1</u>	<u>Unit 2</u>	<u>Unit 1</u>	<u>Unit 2</u>
AC electrical power	6900 V	$\geq 6560$ V and $\leq 7260$ V	SD BD 1A-A	SD BD 2A-A	SD BD 1B-B	SD BD 2B-B
	480 V	$\geq 440$ V and $\leq 508$ V	SD BD 1A1-A 1A2-A	SD BD 2A1-A 2A2-A	SD BD 1B1-B 1B2-B	SD BD 2B1-B 2B2-B
AC vital instrument electrical power	120 V	$\geq 120.6$ V and $\leq 126.6$ V	<u>Unit 1</u> Board 1-I Board 1-III	<u>Unit 2</u> Board 2-I Board 2-III	<u>Unit 1</u> Board 1-II Board 1-IV	<u>Unit 2</u> Board 2-II Board 2-IV
Vital DC electrical power	125 V	$\geq 129$ V and $\leq 140$ V	Board I	Board III	Board II	Board IV
DG DC electrical power	125 V	$\geq 124$ V and $\leq 135$ V	DG 1A-A Dist. Pnl.	DG 2A-A Dist. Pnl.	DG 1B-B Dist. Pnl.	DG 2B-B Dist. Pnl.

## B 3.8 ELECTRICAL POWER SYSTEMS

### B 3.8.10 Distribution Systems - Shutdown

#### BASES

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BACKGROUND	A description of the AC, vital DC, diesel generator (DG) DC, and AC vital instrument electrical power distribution systems is provided in the Bases for LCO 3.8.9, "Distribution Systems - Operating."
APPLICABLE SAFETY ANALYSES	<p>The initial conditions of Design Basis Accident and transient analyses in the UFSAR, Chapter 6 (Ref. 1) and Chapter 15 (Ref. 2), assume Engineered Safety Feature (ESF) systems are OPERABLE. The AC, vital DC, DG DC, and AC vital instrument 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.</p> <p>The OPERABILITY of the AC, vital DC, DG DC, and AC vital instrument electrical power distribution system is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY.</p> <p>The OPERABILITY of the minimum AC, vital DC, DG DC, and AC vital instrument electrical power distribution subsystems during MODES 5 and 6, and during movement of irradiated fuel assemblies ensures that:</p> <ol style="list-style-type: none"><li>The unit can be maintained in the shutdown or refueling condition for extended periods,</li><li>Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status, and</li><li>Adequate power is provided to mitigate events postulated during shutdown, such as a fuel handling accident.</li></ol> <p>The AC and DC electrical power distribution systems satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).</p>

BASES

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

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

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

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

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

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ACTIONS LCO 3.0.3 is not applicable while in MODE 5 or 6. However, since irradiated fuel assembly movement can occur in MODE 1, 2, 3, or 4, the ACTIONS have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operations. Entering LCO 3.0.3, while in MODE 1, 2, 3, or 4 would require the unit to be shutdown unnecessarily.

BASES

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ACTIONS (continued)

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

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 irradiated 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 subsystem 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 movement of irradiated fuel assemblies, and operations involving positive reactivity additions that could result in loss of required SDM (MODE 5) or boron concentration (MODE 6). Suspending positive reactivity additions that could result in failure to meet the minimum SDM or boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than that what would be required in the RCS for minimum SDM or refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation. Introduction of temperature changes including temperature increases when operating with a positive MTC must also be evaluated to ensure they do not result in a loss of required SDM.

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

Notwithstanding performance of the above conservative Required Actions, a required residual heat removal (RHR) subsystem may be inoperable. In this case, Required Actions A.2.1 through A.2.3 do not adequately address the concerns relating to coolant circulation and heat removal. Pursuant to LCO 3.0.6, the RHR ACTIONS would not be entered. Therefore, Required Action A.2.4 is provided to direct declaring RHR inoperable and not in operation, which results in taking the appropriate RHR actions.

BASES

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ACTIONS (continued)

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.

B.1

If one or more required DG DC electrical power distribution panels are inoperable, the associated DGs may be incapable of performing their intended function and must be immediately declared inoperable. This declaration also requires entry into the applicable Conditions and Required Actions for inoperable DGs, LCO 3.8.2, "AC Sources – Shutdown."

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SURVEILLANCE  
REQUIREMENTS

SR 3.8.10.1

This Surveillance verifies that the AC, vital DC, DG DC, and AC vital instrument electrical power distribution subsystems are functioning properly, with all the boards energized. The verification of proper voltage availability on the boards ensures that the required power is readily available for motive as well as control functions for critical system loads connected to these boards.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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REFERENCES

1. UFSAR, Chapter 6.
  2. UFSAR, Chapter 15.
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## B 3.9 REFUELING OPERATIONS

### B 3.9.1 Boron Concentration

#### BASES

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

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

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 storage tank through the open reactor vessel by gravity feeding or by the use of the Residual Heat Removal (RHR) System pumps.

The pumping action of the RHR System in the RCS and the natural circulation due to thermal driving heads in the reactor vessel and refueling cavity mix the added concentrated boric acid with the water in the refueling canal. The RHR System is in operation during refueling (see LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level," and LCO 3.9.6, "Residual Heat Removal (RHR) 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.

BASES

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APPLICABLE  
SAFETY  
ANALYSES

An uncontrolled boron dilution accident is not credible during refueling. This accident is prevented by administrative controls which isolate the RCS from significant sources of unborated water. 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 plant refueling procedures that verify the correct fuel loading plan (including full core mapping) ensure that the  $k_{\text{eff}}$  of the core will remain  $\leq 0.95$  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 Bases B 3.1.1, "SHUTDOWN MARGIN (SDM)."

The RCS boron concentration satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

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LCO

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

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APPLICABILITY

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

The Applicability is modified by a Note. The Note states that the limits on boron concentration are only applicable to the refueling canal and the refueling cavity when those volumes are connected to the RCS. When the refueling canal and the refueling cavity are isolated from the RCS, no potential path for boron dilution exists.

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ACTIONS

A.1

Continuation of 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

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## BASES

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### ACTIONS (continued)

volume in the RCS, the refueling canal, or the refueling cavity is less than its limit, all operations involving positive reactivity additions must be suspended immediately.

Suspension of positive reactivity additions shall not preclude moving a component to a safe position. Operations that individually add limited positive reactivity (e.g., temperature fluctuations from inventory addition or temperature control fluctuations), but when combined with all other operations affecting core reactivity (e.g., intentional boration) result in overall net negative reactivity addition, are not precluded by this action.

#### A.2

In addition to immediately suspending positive reactivity additions, boration to restore the concentration must be initiated immediately.

In determining the required combination of boration flow rate and concentration, no unique Design Basis Event must be satisfied. The only requirement is to restore the boron concentration to its required value as soon as possible. In order to raise the boron concentration as soon as possible, the operator should begin boration with the best source available for unit conditions.

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

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### SURVEILLANCE REQUIREMENTS

#### SR 3.9.1.1

This SR ensures that the coolant boron concentration in the RCS, and connected portions of the refueling canal and the refueling cavity, is within the COLR limits. The boron concentration of the coolant in each required volume is determined periodically by chemical analysis. Prior to re-connecting portions of the refueling canal or the refueling cavity to the RCS, this SR must be met per SR 3.0.4. If any dilution activity has occurred while the cavity or canal were disconnected from the RCS, this SR ensures the correct boron concentration prior to communication with the RCS.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

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- REFERENCES
1. 10 CFR 50, Appendix A, GDC 26.
  2. UFSAR, Chapter 15.
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## B 3.9 REFUELING OPERATIONS

### B 3.9.2 Unborated Water Source Isolation Valves

#### BASES

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**BACKGROUND** During MODE 6 operations, all isolation valves in a specified combination for reactor makeup water sources containing unborated water that are connected to the Reactor Coolant System (RCS) must be closed to prevent unplanned boron dilution of the reactor coolant. The isolation valves must be secured in the closed position.

The Chemical and Volume Control System is capable of supplying borated and unborated water to the RCS through various flow paths. Since a positive reactivity addition made by reducing the boron concentration is inappropriate during MODE 6, isolation of all unborated water sources prevents an unplanned boron dilution.

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**APPLICABLE SAFETY ANALYSES** The possibility of an inadvertent boron dilution event (Ref. 1) occurring during MODE 6 refueling operations is precluded by adherence to this LCO, which requires that potential dilution sources be isolated. Closing the required valves during refueling operations prevents the flow of unborated water to the filled portion of the RCS. The valves are used to isolate unborated water sources. These valves have the potential to indirectly allow dilution of the RCS boron concentration in MODE 6. By isolating unborated water sources, a safety analysis for an uncontrolled boron dilution accident in accordance with the Standard Review Plan (Ref. 2) is not required for MODE 6.

The RCS boron concentration satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

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**LCO** This LCO requires that flow paths to the RCS from unborated water sources be isolated to prevent unplanned boron dilution during MODE 6 and thus avoid a reduction in SDM. These flow paths are isolated by securing, in the closed position, each valve in one of the valve combinations listed in Table B 3.9.2-1.

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**APPLICABILITY** In MODE 6, this LCO is applicable to prevent an inadvertent boron dilution event by ensuring isolation of all sources of unborated water to the RCS.

For all other MODES, the boron dilution accident was analyzed and was found to be capable of being mitigated.

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BASES

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ACTIONS

The ACTIONS Table has been modified by a Note that allows separate Condition entry for each unborated water source isolation valve in the required valve combination.

A.1

Preventing inadvertent dilution of the reactor coolant boron concentration is dependent on maintaining the unborated water isolation valves secured closed. Securing the valves in the closed position ensures that the valves cannot be inadvertently opened. The Completion Time of "immediately" requires an operator to initiate actions to close an open valve and secure the isolation valve in the closed position immediately. The intent of this Required Action is that once actions are initiated, they must be continued until the valves are secured in the closed position.

A.2

Due to the potential of having diluted the boron concentration of the reactor coolant, SR 3.9.1.1 (verification of boron concentration) must be performed whenever Condition A is entered to demonstrate 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.

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SURVEILLANCE  
REQUIREMENTS

SR 3.9.2.1

At least one combination of valves, listed in Table B 3.9.2-1, is to be secured closed to isolate possible dilution paths. The likelihood of a significant reduction in the boron concentration during MODE 6 operations is remote due to the large mass of borated water in the refueling cavity and the fact that all unborated water sources are isolated, precluding a dilution. The boron concentration is checked, during MODE 6, under SR 3.9.1.1. This Surveillance demonstrates that the valves are closed by administrative means.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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REFERENCES

1. UFSAR, Section 15.2.4.
  2. NUREG-0800, Section 15.4.6.
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Table B 3.9.2-1  
Unborated Water Source Isolation Valves

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Isolation Valve Combination	Valve Numbers
Combination A	2-81-536
	2-62-922
	2-62-916
	2-62-933
Combination B	2-81-536
	2-62-922
	2-62-916
	2-62-940
	2-62-696
	2-62-929
	2-62-932
	2-FCV-62-128
Combination C	2-81-536
	2-62-907
	2-62-914
	2-62-921
	2-62-933
Combination D	2-81-536
	2-62-907
	2-62-914
	2-62-921
	2-62-940
	2-62-929
	2-62-932
	2-62-696
2-FCV-62-128	

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## B 3.9 REFUELING OPERATIONS

### B 3.9.3 Nuclear Instrumentation

#### BASES

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**BACKGROUND** The source range neutron flux monitors are used during refueling operations to monitor the core reactivity condition. The installed source range neutron flux monitors are part of the Nuclear Instrumentation System (NIS). These detectors are located external to the reactor vessel and detect neutrons leaking from the core.

The installed source range neutron flux monitors are Dual Chamber Unguarded Fission Chamber detectors. The detectors monitor the neutron flux in counts per second. The instrument range covers six decades of neutron flux (1E+6 cps) with a 7% instrument accuracy. The detectors also provide continuous visual indication in the control room and an audible count rate in the containment and the control room. The NIS is designed in accordance with the criteria presented in Reference 1.

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**APPLICABLE SAFETY ANALYSES** Two OPERABLE source range neutron flux monitors are required to provide a signal to alert the operator to unexpected changes in core reactivity such as with a boron dilution accident (Ref. 2) or an improperly loaded fuel assembly (Ref. 3). The need for a requirement for the source range neutron flux monitors to mitigate an uncontrolled boron dilution accident is eliminated by isolating all unborated water sources as required by LCO 3.9.2, "Unborated Water Source Isolation Valves."

Fuel assembly loading errors are prevented by administrative procedures implemented during core loading (Ref. 3). These administrative procedures include detailed neutron count rate monitoring to determine that the just loaded fuel assembly does not excessively increase the count rate and that the extrapolated inverse count rate ratio is not decreasing for unexplained reasons.

The source range neutron flux monitors are not assumed to function during a MODE 6 design basis accident or transient. However, because the source range neutron flux monitors provide the primary on-scale monitoring of neutron flux levels during refueling, they are retained in the technical specifications.

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**LCO** This LCO requires that two source range neutron flux monitors be OPERABLE to ensure that redundant monitoring capability is available to detect changes in core reactivity. To be OPERABLE, each monitor must provide visual indication in the control room.

## BASES

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**APPLICABILITY** In MODE 6, the source range neutron flux monitors must be OPERABLE to determine changes in core reactivity. There are no other direct means available to check core reactivity levels. In MODES 2, 3, 4, and 5, these same installed source range detectors and circuitry are also required to be OPERABLE by LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation," and LCO 3.3.9, "Boron Dilution Monitoring Instrumentation (BDMI)."

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**ACTIONS** A.1 and A.2

With only one source range neutron flux monitor OPERABLE, redundancy has been lost. Since these instruments are the only direct means of monitoring core reactivity conditions, CORE ALTERATIONS and introduction of coolant into the RCS with boron concentration less than required to meet the minimum boron concentration of LCO 3.9.1 must be suspended immediately. Suspending CORE ALTERATIONS is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than what would be required in the RCS for minimum refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation. Performance of Required Action A.1 shall not preclude completion of movement of a component to a safe position.

### B.1

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

### B.2

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

The Completion Time of once per 12 hours is sufficient to obtain and analyze a reactor coolant sample for boron concentration and ensures that unplanned changes in boron concentration would be identified. The

BASES

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ACTIONS (continued)

12 hour Frequency is reasonable, considering the low probability of a change in core reactivity during this time period.

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SURVEILLANCE  
REQUIREMENTS

SR 3.9.3.1

SR 3.9.3.1 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 Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.9.3.2

SR 3.9.3.2 is the performance of a CHANNEL CALIBRATION. 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 Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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REFERENCES

1. 10 CFR 50, Appendix A, GDC 13, GDC 26, GDC 28, and GDC 29.
  2. UFSAR, Section 15.2.4.
  3. UFSAR, Section 15.3.3.
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## B 3.9 REFUELING OPERATIONS

### B 3.9.4 Containment Penetrations

#### BASES

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

The containment serves to contain fission product radioactivity that may be released from the reactor core following an accident, such that offsite radiation exposures are maintained within the requirements of 10 CFR 50.67. Additionally, the containment provides radiation shielding from the fission products that may be present in the containment atmosphere following accident conditions.

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

The containment air locks, which are also part of the containment pressure boundary, provide a means for personnel access during MODES 1, 2, 3, and 4 unit operation in accordance with LCO 3.6.2, "Containment Air Locks." Each air lock has a door at both ends. The doors are normally interlocked to prevent simultaneous opening when containment OPERABILITY is required. During periods of unit shutdown when containment closure is not required, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary. During 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 capable of being closed.

BASES

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BACKGROUND (continued)

The requirements for containment penetration closure ensure that a release of fission product radioactivity within containment will be restricted to within regulatory limits.

The Reactor Building Purge Ventilation (RBPV) System includes three subsystems. The normal subsystem includes four 24 inch purge penetrations and two 24 inch exhaust penetrations. The second subsystem, a pressure relief system, includes an 8 inch exhaust penetration. The third subsystem includes a 12 inch instrument room supply penetration and a 12 inch exhaust penetration. During MODES 1, 2, 3, and 4, no more than one pair of containment purge lines (one set of supply valves and one set of exhaust valves) may be opened (Ref. 4). None of the subsystems are subject to a Specification in MODE 5.

In MODE 6, large air exchangers are necessary to conduct refueling operations. The normal 24 inch purge system is used for this purpose, and all valves are closed by Containment Ventilation Isolation in accordance with LCO 3.3.6, "Containment Ventilation Isolation Instrumentation."

The other containment penetrations that provide direct access from containment atmosphere to outside atmosphere must be isolated on at least one side. Isolation may be achieved by an OPERABLE automatic isolation valve (either open or closed), 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 irradiated fuel movements (Ref. 1).

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APPLICABLE  
SAFETY  
ANALYSES

During 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 resulting from dropping a single irradiated fuel assembly (Ref. 2). The requirements of LCO 3.9.7, "Refueling Cavity Water Level," in conjunction with a minimum decay time of 100 hours prior to irradiated fuel movement with containment closure capability, ensures that the release of fission product radioactivity, subsequent to a fuel handling accident, results in doses that are within the values specified in 10 CFR 50.67 or the NRC staff approved licensing basis (e.g., Regulatory Guide 1.183, (Ref. 3) limits).

Containment penetrations satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

## BASES

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### LCO

This LCO limits the consequences of a fuel handling accident involving handling irradiated fuel 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 or to the auxiliary building secondary containment enclosure, to be closed except for the OPERABLE containment purge and exhaust penetrations and the containment personnel air locks. For the OPERABLE containment purge and exhaust penetrations, this LCO ensures that these penetrations are isolable by an automatic Containment Ventilation isolation valve. The OPERABILITY requirements for this LCO ensure that the containment ventilation isolation valve closure times specified in the UFSAR can be achieved and, therefore, meet the assumptions used in the safety analysis to ensure that releases through the valves are terminated, such that radiological doses are within the acceptance limit.

During movement of recently irradiated fuel assemblies within containment, the equipment hatch is required to be held in place by at least four bolts.

The LCO is modified by a Note allowing penetration flow paths with direct access from the containment atmosphere that transverse and terminate in the Auxiliary Building Secondary Containment Enclosure to be unisolated under administrative controls. Administrative controls ensure that 1) appropriate personnel are aware of the open status of the penetration flow path during movement of irradiated fuel assemblies within containment, and 2) specified individuals are designated and readily available to isolate the flow path in the event of a fuel handling accident.

The containment personnel air lock doors may be open during movement of irradiated fuel in the containment provided that one door is capable of being closed in the event of a fuel handling accident. Should a fuel handling accident occur inside containment, at least one personnel air lock door will be closed following an evacuation of containment.

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### APPLICABILITY

The containment penetration requirements are applicable when there is a potential for the limiting fuel handling accident (FHA). The applicability requirements are based on the FHA analysis which assumes a fuel assembly is dropped and damaged during refueling. In MODES 1, 2, 3, and 4, containment penetration requirements are addressed by LCO 3.6.1. In MODES 5 and 6, when movement of irradiated fuel assemblies within containment is not being conducted, the potential for a fuel handling accident does not exist. Additionally, due to radioactive decay, a fuel handling accident involving handling irradiated fuel that is not "recently" irradiated (i.e., fuel that has occupied part of a critical reactor core within the previous 100 hours) will result in doses that are within the values specified in 10 CFR 50.67 even without containment closure capability. The applicability of 3.9.4.a. for the Containment Building

BASES

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APPLICABILITY (continued)

Equipment Hatch is "During the movement of recently irradiated fuel in containment" which maintains the containment closure requirements when the fuel has not sufficiently decayed to remain within these limits. The applicability of 3.9.4.b. and c. for the Containment Air Lock Doors and containment penetrations that provide direct access from containment atmosphere to outside atmosphere is "During movement of irradiated fuel in containment."

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ACTIONS

A.1

If the containment equipment hatch, is not in the required status, the unit must be placed in a condition where the isolation function is not needed. This is accomplished by immediately suspending movement of recently irradiated fuel assemblies within containment. Performance of these actions shall not preclude completion of movement of a component to a safe position.

B.1

If the containment building air lock doors or any other containment penetration that provides direct access from the containment atmosphere to the outside atmosphere is not in the required status, including the Containment Ventilation Isolation valve(s) not capable of automatic actuation when the purge and exhaust valves are open, the unit must be placed in a condition where the isolation function is not needed. This is accomplished by immediately suspending movement of irradiated fuel assemblies within containment. Performance of these actions shall not preclude completion of movement of a component to a safe position.

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SURVEILLANCE  
REQUIREMENTS

SR 3.9.4.1

This Surveillance demonstrates that each containment penetration is in its required status. The requirement that penetrations are capable of being closed by an OPERABLE automatic containment ventilation isolation valve, can be verified by ensuring that each required containment ventilation isolation valve operator has motive power.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.9.4.2

This Surveillance demonstrates that each containment ventilation isolation valve, that is not locked, sealed, or otherwise secured in

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BASES

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SURVEILLANCE REQUIREMENTS (continued)

position, actuates to its isolation position on manual initiation or on an actual or simulated actuation signal.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

The SR is modified by a Note stating that this Surveillance is not required to be met for valves in isolated penetrations. The LCO provides the option to close penetrations in lieu of requiring automatic actuation capability.

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REFERENCES

1. GPU Nuclear Safety Evaluation SE-0002000-001, Rev. 0, May 20, 1988.
  2. Document ID: LTR-CRA-02-219, Westinghouse Electric Company, "Radiological Consequences of Fuel Handling Accidents for the Sequoyah Nuclear Plant Units 1 and 2."
  3. Regulatory Guide 1.183, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors, July 2000.
  4. UFSAR, Section 9.4.7.
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## B 3.9 REFUELING OPERATIONS

### B 3.9.5 Residual Heat Removal (RHR) and Coolant Circulation - High Water Level

#### BASES

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**BACKGROUND** The purpose of the RHR System in MODE 6 is to remove decay heat and sensible heat from the Reactor Coolant System (RCS), as required by GDC 34 (Ref. 1). Operation of the RHR system provides mixing of borated coolant and prevents boron stratification. Heat is removed from the RCS by circulating reactor coolant through the RHR heat exchanger(s), where the heat is transferred to the Component Cooling Water System. The coolant is then returned to the RCS via the RCS cold leg(s). Operation of the RHR System for normal cooldown or decay heat removal is manually accomplished from the control room. The heat removal rate is adjusted by controlling the flow of reactor coolant through the RHR heat exchanger(s) and the bypass. Mixing of the reactor coolant is maintained by this continuous circulation of reactor coolant through the RHR System.

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**APPLICABLE SAFETY ANALYSES** If the reactor coolant temperature is not maintained below 200°F, boiling of the reactor coolant could result. This could lead to a loss of coolant in the reactor vessel. Additionally, boiling of the reactor coolant could lead to a reduction in boron concentration in the coolant due to boron plating out on components near the areas of the boiling activity. 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 loop of the RHR System is required to be operational in MODE 6, with the water level  $\geq$  23 ft above the top of the reactor vessel flange, to prevent this challenge. The LCO does permit the RHR pump to be removed from operation for short durations, under the condition that the boron concentration is not diluted. This conditional stopping of the RHR pump does not result in a challenge to the fission product barrier.

The RHR System satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

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

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

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LCO (continued)

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

The LCO is modified by a Note that allows the required operating RHR loop to be removed from operation for up to 1 hour per 8 hour period, provided no operations are permitted that would dilute the RCS boron concentration by introduction of coolant into the RCS with boron concentration less than required to meet the minimum boron concentration of LCO 3.9.1, "Boron Concentration." Boron concentration reduction with coolant at boron concentrations less than required to assure the RCS boron concentration is maintained 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 RHR isolation valve testing. During this 1 hour period, decay heat is removed by natural convection to the large mass of water in the refueling cavity.

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APPLICABILITY

One RHR loop must be OPERABLE and in operation in MODE 6, with the water level  $\geq$  23 ft above the top of the reactor vessel flange, to provide decay heat removal. The 23 ft water level was selected because it corresponds to the 23 ft requirement established for fuel movement in LCO 3.9.7, "Refueling Cavity Water Level." Requirements for the RHR 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). RHR loop requirements in MODE 6 with the water level  $<$  23 ft are located in LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level."

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ACTIONS

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

A.1

If RHR loop requirements are not met, there will be no forced circulation to provide mixing to establish uniform boron concentrations. Suspending positive reactivity additions that could result in failure to meet the minimum boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than that what would be required in the RCS for minimum refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation.

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BASES

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ACTIONS (continued)

A.2

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

A.3

If RHR loop requirements are not met, actions shall be initiated and continued in order to satisfy RHR loop requirements. With the unit in MODE 6 and the refueling water level  $\geq$  23 ft above the top of the reactor vessel flange, corrective actions shall be initiated immediately.

A.4, A.5, A.6.1, and A.6.2

If no RHR is in operation, the following actions must be taken within 4 hours:

- a. The equipment hatch must be closed and secured with four bolts;
- b. One door in each air lock must be closed; and
- c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere must be either closed by a manual or automatic isolation valve, blind flange, or equivalent, or verified to be capable of being closed by an OPERABLE automatic Containment Ventilation isolation valve.

With RHR loop requirements not met, the potential exists for the coolant to boil and release radioactive gas to the containment atmosphere. Performing the actions described above ensures that all containment penetrations are either closed or can be closed so that the dose limits are not exceeded.

The Completion Time of 4 hours allows fixing of most RHR problems and is reasonable, based on the low probability of the coolant boiling in that time.

BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.9.5.1

This Surveillance demonstrates that the RHR 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 Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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REFERENCES

1. UFSAR, Section 5.5.7.
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## B 3.9 REFUELING OPERATIONS

### B 3.9.6 Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level

#### BASES

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**BACKGROUND** The purpose of the RHR System in MODE 6 is to remove decay heat and sensible heat from the Reactor Coolant System (RCS), as required by GDC 34 (Ref. 1). Operation of the RHR system provides mixing of borated coolant, and prevents boron stratification. Heat is removed from the RCS by circulating reactor coolant through the RHR heat exchangers where the heat is transferred to the Component Cooling Water System. The coolant is then returned to the RCS via the RCS cold leg(s). Operation of the RHR System for normal cooldown decay heat removal is manually accomplished from the control room. The heat removal rate is adjusted by controlling the flow of reactor coolant through the RHR heat exchanger(s) and the bypass lines. Mixing of the reactor coolant is maintained by this continuous circulation of reactor coolant through the RHR System.

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**APPLICABLE SAFETY ANALYSES** If the reactor coolant temperature is not maintained below 200°F, boiling of the reactor coolant could result. This could lead to a loss of coolant in the reactor vessel. Additionally, boiling of the reactor coolant could lead to a reduction in boron concentration in the coolant due to the boron plating out on components near the areas of the boiling activity. The loss of reactor coolant and the reduction of boron concentration in the reactor coolant will eventually challenge the integrity of the fuel cladding, which is a fission product barrier. Two loops of the RHR System are required to be OPERABLE, and one loop in operation, in order to prevent this challenge.

The RHR System satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

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**LCO** In MODE 6, with the water level < 23 ft above the top of the reactor vessel flange, both RHR loops must be OPERABLE. Additionally, one loop of RHR must be in operation in order to provide:

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

This LCO is modified by two Notes. Note 1 permits the RHR pumps to be removed from operation for  $\leq 15$  minutes when switching from one train to another. The circumstances for stopping both RHR pumps are to be limited to situations when the outage time is short and the core outlet

BASES

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LCO (continued)

temperature is maintained > 10 degrees F below saturation temperature (e.g., if saturation temperature = 190°F, core outlet temperature must be < 180°F). The Note prohibits boron dilution or draining operations when RHR forced flow is stopped.

Note 2 allows one RHR loop to be inoperable for a period of 2 hours provided the other loop is OPERABLE and in operation. Prior to declaring the loop inoperable, consideration should be given to the existing plant configuration. This consideration should include that the core time to boil is short, there is no draining operation to further reduce RCS water level and that the capability exists to inject borated water into the reactor vessel. This permits surveillance tests to be performed on the inoperable loop during a time when these tests are safe and possible.

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

Both RHR pumps may be aligned to the Refueling Water Storage Tank to support filling or draining the refueling cavity or for performance of required testing.

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APPLICABILITY

Two RHR loops are required to be OPERABLE, and one RHR loop must be in operation in MODE 6, with the water level < 23 ft above the top of the reactor vessel flange, to provide decay heat removal. Requirements for the RHR System in other MODES are covered by LCOs in Section 3.4, Reactor Coolant System (RCS), and Section 3.5, Emergency Core Cooling Systems (ECCS). RHR loop requirements in MODE 6 with the water level  $\geq$  23 ft are located in LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level."

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ACTIONS

A.1 and A.2

If less than the required number of RHR loops are OPERABLE, action shall be immediately initiated and continued until the RHR loop is restored to OPERABLE status or until  $\geq$  23 ft of water level is established above the reactor vessel flange. When the water level is  $\geq$  23 ft above the reactor vessel flange, the Applicability changes to that of LCO 3.9.5, and only one RHR loop is required to be OPERABLE and in operation. An immediate Completion Time is necessary for an operator to initiate corrective actions.

## BASES

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### ACTIONS (continued)

#### B.1

If no RHR loop is in operation, there will be no forced circulation to provide mixing to establish uniform boron concentrations. Suspending positive reactivity additions that could result in failure to meet the minimum boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than that what would be required in the RCS for minimum refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation.

#### B.2

If no RHR loop is in operation, actions shall be initiated immediately, and continued, to restore one RHR loop to operation. Since the unit is in Conditions A and B concurrently, the restoration of two OPERABLE RHR loops and one operating RHR loop should be accomplished expeditiously.

#### B.3, B.4, B.5.1, and B.5.2

If no RHR is in operation, the following actions must be taken within 4 hours:

- a. The equipment hatch must be closed and secured with four bolts;
- b. One door in each air lock must be closed; and
- c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere must be either closed by a manual or automatic isolation valve, blind flange, or equivalent, or verified to be capable of being closed by an OPERABLE automatic Containment Ventilation isolation valve.

With RHR loop requirements not met, the potential exists for the coolant to boil and release radioactive gas to the containment atmosphere. Performing the actions stated above ensures that all containment penetrations are either closed or can be closed so that the dose limits are not exceeded.

The Completion Time of 4 hours allows fixing of most RHR problems and is reasonable, based on the low probability of the coolant boiling in that time.

BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.9.6.1

This Surveillance demonstrates that one RHR 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. In addition, during operation of the RHR loop with the water level in the vicinity of the reactor vessel nozzles, the RHR pump suction requirements must be met.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.9.6.2

Verification that the required pump is OPERABLE ensures that a RHR 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 Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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REFERENCES

1. UFSAR, Section 5.5.7.
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## B 3.9 REFUELING OPERATIONS

### B 3.9.7 Refueling Cavity Water Level

#### BASES

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**BACKGROUND** The movement of irradiated fuel assemblies 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, refueling canal, fuel transfer canal, refueling cavity, and spent fuel pool. Sufficient water is necessary to retain iodine fission product activity in the water in the event of a fuel handling accident (Refs. 1 and 2). Sufficient iodine activity would be retained to limit offsite doses from the accident to 10 CFR 50.67 limits, further restricted by the guidance of Reference 1.

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**APPLICABLE SAFETY ANALYSES** During movement of irradiated fuel assemblies, the water level in the refueling canal and the refueling cavity is an initial condition design parameter in the analysis of a fuel handling accident in containment, as postulated by Regulatory Guide 1.183 (Ref. 1). A minimum water level of 23 ft (Appendix B of Ref. 1) allows a decontamination factor of 200 (Appendix B of Ref. 1) to be used in the accident analysis for iodine. This relates to the assumption that 99.5% 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 8% I-131, 10% Kr-85, and 5% of other iodines and noble gases of the total fuel rod inventory (Ref. 1).

The fuel handling accident analysis inside containment is described in Reference 2. With a minimum water level of 23 ft and a minimum decay time of 100 hours prior to fuel handling, the analysis and test programs demonstrate that the iodine release due to a postulated fuel handling accident is adequately captured by the water and offsite doses are maintained within allowable limits (Refs. 1 and 3).

Refueling cavity water level satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

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**LCO** A minimum refueling cavity water level of 23 ft above the reactor vessel flange is required to ensure that the radiological consequences of a postulated fuel handling accident inside containment are within acceptable limits, as provided by the guidance of Reference 1.

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**APPLICABILITY** LCO 3.9.7 is applicable when moving irradiated fuel assemblies within containment. The LCO minimizes the possibility of a fuel handling accident in containment that is beyond the assumptions of the safety analysis. If irradiated fuel assemblies are 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 3.7.13, "Spent Fuel Pool Water Level."

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BASES

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ACTIONS

A.1

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

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

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SURVEILLANCE  
REQUIREMENTS

SR 3.9.7.1

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

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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REFERENCES

1. Regulatory Guide 1.183, July 2000.
  2. UFSAR, Section 15.5.6.
  3. 10 CFR 50.67.
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## B 3.9 REFUELING OPERATIONS

### B 3.9.8 Decay Time

#### BASES

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BACKGROUND	The primary purpose of the decay time requirement is to ensure that the fission product inventories assumed in the fuel handling accident analysis are met. As soon as the reactor is subcritical, the quantity of fission products in the core decreases as the fission products undergo natural radioactive decay. As long as the reactor remains subcritical, this decrease will continue and the radiation levels will also decrease.
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APPLICABLE SAFETY ANALYSES	The fuel handling accident is the postulated event of concern in MODE 6 during fuel handling operations (Ref. 1). It establishes the minimum decay time. It is assumed that all of the fuel rods in the equivalent of one fuel assembly are damaged to the extent that all the gap activity in the rods is released. The damaged fuel assembly is assumed to be the assembly with the highest fission product inventory. The fission product inventories are those assumed to be present 100 hours after the reactor becomes subcritical.
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The decay time satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

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LCO	The LCO requires that the reactor be subcritical for at least 100 hours prior to commencing CORE ALTERATIONS. The requirement to be subcritical for greater than or equal to 100 hours ensures that the fission product radioactivity has undergone natural radioactive decay and that the consequences of a fuel handling accident will be within the bounds of the safety analysis.
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APPLICABILITY	This LCO applies during CORE ALTERATIONS, since the potential for a release of fission products exists.
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ACTIONS	<p><u>A.1</u></p> <p>With the reactor subcritical for less than 100 hours, there shall be no operations involving CORE ALTERATIONS. This will preclude a fuel handling accident with fuel containing more fission product radioactivity than assumed in the safety analysis.</p> <p>The immediate Completion Time is consistent with the required times for actions to be performed without delay and in a controlled manner.</p>
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SURVEILLANCE REQUIREMENTS	<p><u>SR 3.9.8.1</u></p> <p>Prior to CORE ALTERATIONS, the reactor must be determined to be subcritical for greater than or equal to 100 hours by verifying the date and time that the reactor achieved subcritical conditions.</p>
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BASES

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REFERENCES      1. UFSAR, Section 15.5.6.

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