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Technical Specifications Bases (TS)

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Rev.	Date Issued	Date to NRC
0		May 4, 1998
1	August 28, 1998	October 30, 1998
2	August 28, 1998	October 30, 1998
3	October 28, 1998	October 30, 1998
4	March 16, 1999	October 18, 1999
5	October 18, 1999	October 18, 1999

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B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core SLs

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BACKGROUND

Reference 1, Appendix 1C, Criterion 6 requires, and Safety Limits (SLs) ensure, 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 (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 and cladding and possible cladding perforation that 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 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.

Fuel centerline melting occurs when the local linear heat rate 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.

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of DNB and the resultant sharp reduction in heat transfer coefficient. 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,

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resulting in an uncontrolled release of activity to the reactor coolant.

The Reactor Protective System (RPS), in combination with the Limiting Conditions for Operation (LCOs), is designed to prevent any anticipated combination of transient conditions for Reactor Coolant System (RCS) temperature, pressure, and THERMAL POWER level that would result in a violation of the reactor core SLs.

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 a DNB; and

b. The hot fuel pellet in the core must not experience fuel centerline melting.

The RPS setpoints, LCO 3.3.1, in combination with all the LCOs, are designed to prevent any anticipated combination of transient conditions for RCS temperature, pressure, 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 following functions:

a. Pressurizer Pressure-High trip;

b. Power Level-High trip;

c. Rate of Change of Power-High trip;

d. Reactor Coolant Flow-Low trip;

e. Steam Generator Pressure-Low trip;

f. Steam Generator Level-Low trip;

g. Axial Power Distribution-High trip;

h. Thermal Margin/Low Pressure trip;

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i. Steam Generator Pressure Difference trip; and

j. Steam Generator Safety Valves.

The SL represents a design requirement for establishing the RPS trip setpoints identified previously. Limiting Condition for Operation (LCO) 3.2.1, or the assumed initial conditions of the safety analyses (as indicated in Reference 1, Section 14.1), provide more restrictive limits to ensure that the SLs are not exceeded.

SAFETY LIMITS

The curves provided in Figure 2.1.1-1 show the loci of points of THERMAL POWER, pressurizer pressure, and highest operating loop cold leg temperature for which the minimum DNBR is not less than the safety analysis limit. Safety Limit 2.1.1.2 ensures that fuel centerline temperature remains below melting.

APPLICABILITY

Safety Limit 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.

In MODEs 3, 4, 5, and 6, Applicability is not required, since the reactor is not generating significant THERMAL POWER.

SAFETY LIMIT VIOLATIONS

The following SL violation responses are applicable to the reactor core SLs.

<u>2.2.1</u>

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.

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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.

REFERENCES

1. UFSAR

B 2.0 SAFETY LIMITS (SLs)

B 2.1.2 Reactor Coolant System (RCS) Pressure SL

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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, continued RCS integrity is ensured. According to Reference 1, Appendix 1C, Criteria 9 and 33, the reactor coolant pressure boundary (RCPB) design conditions are not to be exceeded during normal operation and AOOs. Also, according to Reference 1, Appendix 1C, Criterion 32, reactivity accidents do not result in rupturing the RCPB.

The design pressure of the RCS is 2500 psia. During normal operation and AOOs, the RCS pressure is kept from exceeding the design pressure by more than 10%, in accordance with Reference 2, Section III, Article NB-7000. To ensure system integrity, all RCS components are hydrostatically tested at 125% of design pressure, according to the American Society of Mechanical Engineers (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 Reference 2, Section XI, Article IWX-5000.

Overpressurization of the RCS could result in a breach of the RCPB. If this 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 Reference 1, Chapter 14.

APPLICABLE SAFETY ANALYSES

The RCS pressurizer safety valves, the main steam safety valves, and the Reactor Pressure-High 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%, in accordance with Reference 2, Section III,

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Article NB-7000. The transient that establishes the required relief capacity, and hence the valve size requirements and lift settings, is a complete loss of 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 Protective System trip setpoints (LCO 3.3.1), together with the settings of the main steam safety valves (LCO 3.7.1) and the pressurizer safety valves, provide pressure protection for normal operation and AOOs. In particular, the Pressurizer Pressure-High trip setpoint is specifically set to provide protection against overpressurization (Reference 1, Section 14.1). Safety analyses for both the Pressure-High 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 the following:

a. Pressurizer power-operated relief valves;

b. Steam Bypass Control System;

c. Pressurizer Level Control System; or

d. Pressurizer Pressure Control System.

SAFETY LIMITS

The maximum transient pressure allowable in the RCS pressure vessel in Reference 2, Section III, Article NB-7000 is 110% of design pressure. The maximum transient pressure allowable in the RCS piping, valves, and fittings under Reference 3, is 110% 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 established at 2750 psia.

APPLICABILITY

Safety Limit 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 is

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unbolted, making it unlikely that the RCS can be pressurized.

SAFETY LIMIT VIOLATIONS

The following SL violation responses are applicable to the RCS pressure SL.

2.2.2.1

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.

With RCS pressure greater than the value specified in SL 2.1.2 in MODE 1 or 2, the pressure must be reduced below this value. A pressure greater than the value specified in SL 2.1.2 exceeds 110% of the RCS design pressure and may challenge system integrity.

The allowed Completion Time of 1 hour provides the operator time to complete the necessary actions to reduce RCS pressure by terminating the cause of the pressure increase, removing mass or energy from the RCS, or a combination of these actions, and to establish MODE 3 conditions.

2.2.2.2

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 potentially 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. This 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.	UFSAR
	2.	ASME, Boiler and Pressure Vessel Code
	3.	ASME, USAS B31.7, Standard Code for Pressure Piping, 1967

B 3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

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LCOs	Limiting Condition for Operation (LCO) 3.0.1 through LCO 3.0.7 establish the general requirements applicable to all Specifications and apply at all times unless otherwise stated.
LCO 3.0.1	Limiting Condition for Operation 3.0.1 establishes the Applicability statement within each individual Specification 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

Limiting Condition for Operation 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 when 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:

a. Completion of the Required Actions within the specified Completion Times constitutes compliance with a Specification; and

b. Completion of the Required Actions is not required when an LCO is met within the specified Completion Time, unless otherwise specified.

There are two basic types of Required Actions. The first type of Required Action specifies a time limit for meeting the LCO. 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, permiting continued operation of the unit, that are not further restricted by the Completion Time. In this case, compliance with the Required Actions provides an acceptable level of safety for continued operation.

Completing the Required Actions is not required when an LCO is met or is no longer applicable, unless otherwise stated in the individual Specifications.

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 (e.g., LCO 3.4.3).

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 Surveillance Requirements (SRs), 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. Doing so limits the time both subsystems/trains of a safety function are inoperable and limits the time conditions exist that may result in LCO 3.0.3 being entered. Individual Specifications may state 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 when the new Specification becomes applicable and the ACTIONS Condition(s) are entered.

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LCO 3.0.3

Limiting Condition for Operation 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.

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 well within the specified maximum cooldown rate and capabilities of the unit, assuming 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 the 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.

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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 has been met.

b. A Condition exists for which the Required Actions have been performed.

c. ACTIONS exist that do not have expired Completion Times. These Completion Times are applicable from that point in time the Condition is initially entered, and not from the time LCO 3.0.3 is exited.

The time limits of Specification 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, the time allowed for reaching MODE 4 is the next 11 hours, because the total time for reaching 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.

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. Limiting Condition for Operation 3.7.13 has an Applicability of "During movement of irradiated fuel assemblies 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 Action of LCO 3.7.13 of "Suspend movement of irradiated fuel assemblies in spent fuel pool" is the appropriate Required Action to complete in lieu of the actions of LCO 3.0.3. These exceptions are addressed in the individual Specifications.

LCO 3.0.4

Limiting Condition for Operation 3.0.4 establishes limitations on changes in MODEs or other specified conditions in the Applicability when an LCO is not met. It precludes placing the unit in a MODE or other specified condition stated in that Applicability (e.g., Applicability desired to be entered) when the following exist:

a. Unit conditions are such that the requirements of the LCO would not be met in the Applicability desired to be entered; and

b. Continued noncompliance with the LCO requirements, if the Applicability were entered, would result in the unit being required to exit this Applicability to comply with the Required Actions.

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

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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.

Exceptions to LCO 3.0.4 are stated in the individual Specifications. The exceptions allow 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. Exceptions may apply to all the ACTIONS or to a specific Required Action of a Specification.

Limiting Condition for Operation 3.0.4 is only applicable when entering MODE 4 from MODE 5, MODE 3 from MODE 4, MODE 2 from MODE 3, or MODE 1 from MODE 2. Furthermore, LCO 3.0.4 is applicable when entering any other specified condition in the Applicability only while operating in MODE 1, 2, 3, or 4. The requirements of LCO 3.0.4 do not apply in MODEs 5 and 6 or 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.

Surveillance Requirements 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, changing MODEs or other specified conditions while in an ACTIONS Condition, in compliance with LCO 3.0.4 or where an exception to LCO 3.0.4 is stated, is not a violation of SR 3.0.1 or SR 3.0.4 for those SRs that do not have to be performed due to the associated 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.

LCO 3.0.5

Limiting Condition for Operation 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 SRs 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 allowed SRs. 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 SRs.

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 an SR 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 an SR on another channel in the same trip system.

LCO 3.0.6

Limiting Condition for Operation 3.0.6 establishes an exception to LCO 3.0.2 for support systems that have an LCO specified in the Technical Specifications. This exception is provided because LCO 3.0.2 would require the Conditions and Required Actions of the associated inoperable supported system LCO to be entered solely due to the inoperability of the support system. This exception is justified because the actions 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 Technical Specification, the supported system(s) are required to be declared inoperable if they are 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' and Required Actions are eliminated by providing all the actions 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 5.5.15, "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 a 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.

Cross-train checks to identify a loss of safety function for those support systems supporting multiple and redundant safety systems are required. The cross-train check verifies that the supported systems of the redundant OPERABLE support system are OPERABLE, thereby ensuring safety function is retained. A loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to system(s) supported by the inoperable support system is also inoperable; or

EXAMPLE B3.0.6-1
If System 2 of Train A is inoperable, and System 5 of Train B is inoperable, a loss of safety function exists in supported System 5.

b. A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable; or

EXAMPLE B3.0.6-2
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, which is in turn supported by System 5.

c. A required system redundant to support system(s) for the supported systems (a) and (b) above is also inoperable.

EXAMPLE B3.0.6-3
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.

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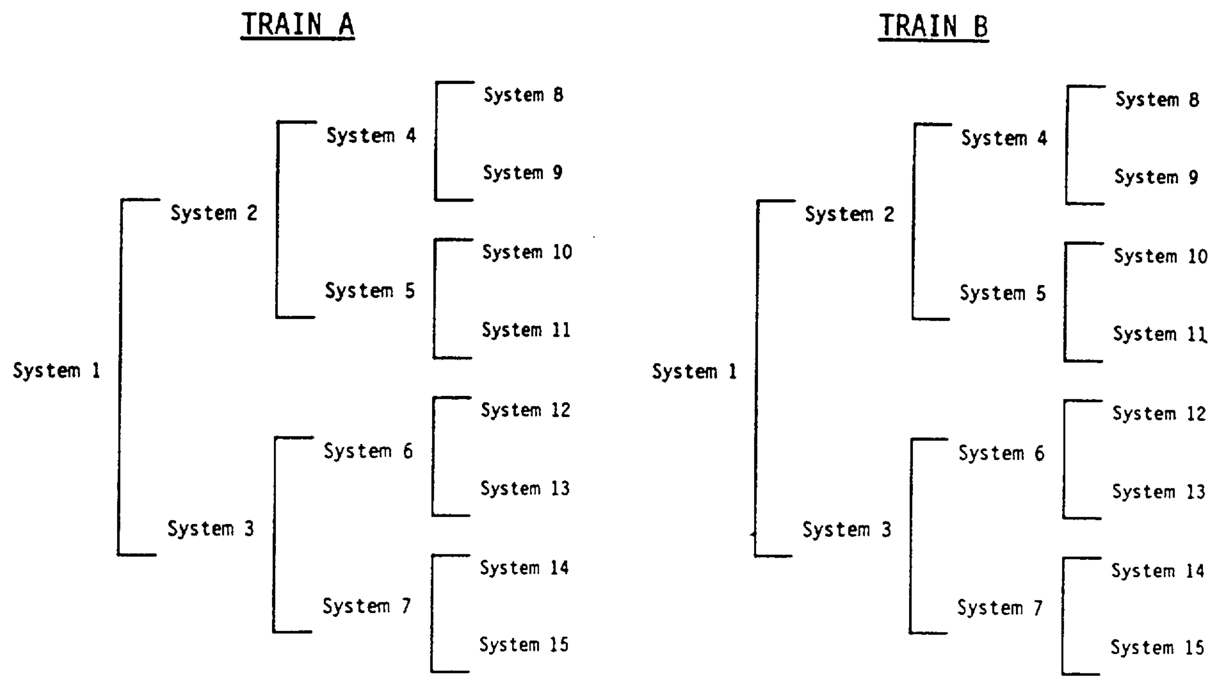

If this evaluation determines that a loss of safety function exists, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

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. Special Test Exception (STE) LCOs 3.1.7, 3.1.8, and 3.4.17 allow specific Technical Specification requirements to be changed to permit performances of these special tests and operations, which otherwise could not be performed if required to comply with these Technical Specifications. Unless otherwise specified, all the other Technical Specification 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 STE LCO represents a condition not necessarily in compliance with the normal requirements of

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the Technical Specification. Compliance with STE LCOs is optional. A special operation may be performed either under the provisions of the appropriate STE LCO or under the other applicable Technical Specification requirements. If it is desired to perform the special operation under the provisions of the STE LCO, the requirements of the STE LCO shall be followed.

B 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

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SRs	Surveillance Requirement 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 3.0.1

Surveillance Requirement 3.0.1 establishes 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 surveillance tests are performed to verify the OPERABILITY of systems and components, and that variables are within specified limits. Failure to meet a SR within the specified Frequency, in accordance with SR 3.0.2, constitutes a failure to meet an LCO.

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:

a. The systems or components are known to be inoperable, although still meeting the SRs; or

b. The requirements of the surveillance test(s) are known to not be met between required surveillance test performances.

Surveillance tests 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 STE are only applicable when the STE is used as an allowable exception to the requirements of a Specification.

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.

Surveillance tests, including surveillance tests invoked by Required Actions, do not have to be performed on inoperable equipment, because the ACTIONS define the applicable remedial measures. Surveillance tests have to be 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 surveillance tests pass 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 pump turbine maintenance during refueling that requires testing at steam pressures > 800 psi. However, if other appropriate testing is satisfactorily completed, the Auxiliary Feedwater 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 maintenance during shutdown that requires system functional tests at a specified pressure. Provided other appropriate testing is satisfactorily completed, startup can proceed with high pressure safety injection 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

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

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

The 25% extension does not significantly degrade the reliability that results from performing the surveillance test at its specified Frequency. This is based on the recognition that the most probable result of any particular surveillance test being performed is the verification of conformance with the SRs. The exceptions to SR 3.0.2 are those surveillance tests for which the 25% extension of the interval specified in the Frequency does not apply. These exceptions are stated in the individual Specifications. An example of where SR 3.0.2 does not apply is the Containment Leakage Rate Testing Program.

As stated in SR 3.0.2, the 25% extension also does not apply to the initial portion of a periodic Completion Time that requires performance on a "once per . . ." basis. The 25% extension applies to each performance after the initial performance. The initial performance of the Required Action, whether it is a particular surveillance test or some other remedial action, is considered a single action with a single Completion Time. One reason for not allowing the 25% 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 test intervals (other than those consistent with refueling intervals) or periodic Completion Time intervals beyond those specified.

SR 3.0.3

Surveillance Requirement 3.0.3 establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a surveillance test 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 less, applies from when it is discovered that the surveillance test has not been performed in accordance with SR 3.0.2, and not at the time that the specified Frequency was not met.

This delay period provides an adequate time to complete missed surveillance tests. This delay period permits the completion of a surveillance test before complying with Required Actions or other remedial measures that might preclude completion of the surveillance test.

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

When a surveillance test with a Frequency based not on time intervals, but upon specified unit conditions or operational situations, is discovered not to have been performed when specified, SR 3.0.3 allows the full delay period of 24 hours to perform the surveillance test.

Surveillance Requirement 3.0.3 also provides a time limit for completion of surveillance tests 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 that is not intended to be used as an operational convenience to extend surveillance test intervals.

If a surveillance test is not completed within the allowed delay period, 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 test fails within the delay period, 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 test.

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

SR 3.0.4

Surveillance Requirement 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 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.

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 required to be performed, per SR 3.0.1, which states surveillance tests 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 test(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.

The provisions of SR 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 SR 3.0.4 shall not prevent changes in MODEs or other specified conditions in the Applicability that result from any unit shutdown.

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 SR, or both. This allows performance of surveillance tests when the prerequisite condition(s) specified in a surveillance test procedure requires entry into the MODE or other specified condition in the Applicability of the associated LCO prior to the performance or completion of a surveillance test. A surveillance test that could not be performed until after entering the LCO Applicability would have its Frequency specified such that it is not "due" until the specific conditions needed are met. Alternately, the surveillance test 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.

Surveillance Requirement 3.0.4 is only applicable when entering MODE 4 from MODE 5, MODE 3 from MODE 4, or MODE 2 from MODE 3, or MODE 1 from MODE 2. Furthermore, SR 3.0.4 is applicable when entering any other specified condition in the Applicability only while operating in MODEs 1, 2, 3, or 4. The requirements of SR 3.0.4 do not apply in MODEs 5 and 6, or in other specified conditions of the Applicability (unless in MODEs 1, 2, 3, or 4), because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.1 SHUTDOWN MARGIN (SDM)

BASES

BACKGROUND

The reactivity control systems must be redundant and capable of holding the reactor core subcritical when shut down under cold conditions, in accordance with Reference 1, Appendix 1C, Criteria 27, 29, and 30. Maintenance of the SDM ensures that postulated reactivity events will not damage the fuel. SHUTDOWN MARGIN 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 of all control element assemblies (CEAs), assuming the single CEA of highest reactivity worth is fully withdrawn.

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 CEAs and soluble boric acid in the Reactor Coolant System (RCS). The CEA 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 CEA 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.

During power operation, SDM control is ensured by operating with the shutdown CEAs fully withdrawn and the regulating CEAs within the limits of Limiting Condition for Operation (LCO) 3.1.6. When the unit is in the shutdown and refueling MODEs, the SDM requirements are met by means of adjustments to the RCS boron concentration.

APPLICABLE SAFETY ANALYSIS

The minimum required SDM is assumed as an initial condition in safety analysis. The safety analysis (Reference 1, Section 3.4) establishes a SDM that ensures specified acceptable fuel design limits (SAFDLs) are not exceeded for normal operation and AOOs, with the assumption of the highest worth CEA stuck out following a reactor trip. For MODE 5, the primary safety analysis that relies on the SDM limit is the boron dilution analysis.

The acceptance criteria for the SDM requirements are that SAFDLs 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 limit AOOs, and an acceptable energy deposition for the CEA ejection accident [Reference 1, Chapter 14]); 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 are based on a main steam line break (MSLB), as described in the accident analysis (Reference 1, Chapter 14). The increased steam flow resulting from a pipe break in the Main Steam System causes an increased energy removal from the affected steam generator, 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 (MTC), 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 guillotine 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 steam generator 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 requirement for MODEs 3 and 4 must also protect against an uncontrolled CEA withdrawal from a hot zero power or low power condition, and a CEA 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.

The withdrawal of CEAs from hot zero power or low power conditions 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 withdrawal of CEAs also produces a time-dependent redistribution of core power.

Depending on the system initial conditions and reactivity insertion rate, the uncontrolled CEA withdrawal transient is terminated by either a High Power trip or a High Pressurizer Pressure trip. In all cases, power level, RCS pressure, linear heat rate (LHR), and the DNBR do not exceed allowable limits.

SHUTDOWN MARGIN satisfies 10 CFR 50.36(c)(2)(ii), Criterion 2.

LCO

The MSLB and the boron dilution accidents (Reference 1, Chapter 14) 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 the acceptance criteria given in Reference 1, Chapter 14. 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. Because both initial RCS level and the dilution flow rate

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also significantly impact the boron dilution event in MODE 5 with pressurizer level < 90 inches from the bottom of the pressurizer, the LCO also includes limits for these parameters during these conditions.

SHUTDOWN MARGIN is a core physics design condition that can be ensured through CEA positioning (regulating and shutdown CEA) in MODEs 1 and 2 and through the soluble boron concentration in all other MODEs.

APPLICABILITY

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 MODEs 1 and 2, SDM is ensured by complying with LCOs 3.1.5 and 3.1.6. In MODE 6, the shutdown reactivity requirements are given in LCO 3.9.1.

ACTIONS

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

With non-borated water sources of > 88 gpm available, while the unit is in MODE 5 with the pressurizer level < 90 inches, the consequences of a boron dilution event may exceed the analysis results. Therefore, action must be initiated immediately to reduce the potential for such an event. To accomplish this, Required Action A.1 requires immediate suspension of positive reactivity additions. However, since Required Action A.1 only reduces the potential for the event and does not eliminate it, immediate action must also be initiated to increase the SDM to compensate for the non-borated water sources (Required Action A.2). Finally, Required Action A.3 requires periodic verification, once per 12 hours, that the SDM increase is maintained sufficient to compensate for the additional sources of non-borated water. Required Action A.1 is modified by a Note indicating that the suspension of positive reactivity additions is not required if SDM has been sufficiently increased to compensate for the additional sources of non-borated water. The immediate Completion Time reflects the urgency of the corrective actions. The periodic Completion Time of 12 hours is considered reasonable, based on other administrative controls available and operating experience.

B.1 and B.2

With the RCS level at or below the bottom of the hot leg nozzles, while the unit is in MODE 5 with the pressurizer level < 90 inches, the consequences of a boron dilution event may exceed the analysis results. Therefore, action must be initiated immediately to reduce the potential for such an event. To accomplish this, Required Action B.1 requires immediate suspension of positive reactivity additions. However, since Required Action B.1 only reduces the potential for the event and does not eliminate it, immediate action must also be initiated to increase the RCS level to above the bottom of the hot leg nozzles (Required Action B.2). The immediate Completion Time reflects the urgency of the corrective actions.

C.1

If the SDM requirements are not met for reasons other than addressed in Condition A or B, boration must be initiated promptly. A Completion Time of immediately is required to meet the assumptions of the safety analysis. 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 storage tank or the refueling water tank. The operator should borate with the best source available for the plant conditions. However, as a minimum, the boration flow rate shall be ≥ 40 gpm and the boron concentration shall be ≥ 2300 ppm boric acid solution or equivalent.

Assuming that a value of 1% $\Delta k/k$ must be recovered and a boration flow rate of 40 gpm from the boric acid storage tank, it is possible to increase the boron concentration of the RCS by 100 ppm in approximately 15 minutes. If an inverse boron worth of 100 ppm/% $\Delta k/k$ is assumed, this combination of parameters will increase the SDM by 1% $\Delta k/k$. These boration parameters of 40 gpm and 100 ppm represent typical values and are provided for the purpose of offering a specific example.

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1

SHUTDOWN MARGIN is verified by performing a reactivity balance calculation, considering the listed reactivity effects:

a. RCS boron concentration;

b. CEA positions;

c. RCS average temperature;

d. Fuel burnup based on gross thermal energy generation;

e. Xenon concentration;

f. Samarium concentration; and

g. Isothermal temperature coefficient.

Using the isothermal temperature coefficient 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 Frequency of 24 hours is based on the generally slow change in required boron concentration, and also allows sufficient time for the operator to collect the required data, which includes performing a boron concentration analysis, and complete the calculation.

SR 3.1.1.2 and SR 3.1.1.3

These Surveillance Requirements (SRs) periodically verify the significant assumptions of a boron dilution event are maintained. A non-borated water source of ≤ 88 gpm allows for only one charging pump to be capable of injection during these conditions since each charging pump is capable of an injection rate of 46 gpm. Each SR is modified by a Note indicating that it is only required when the unit is in MODE 5 with the pressurizer level < 90 inches. Since the applicable conditions for the SR may be attained while already in MODE 5, each SR is provided with a Frequency of once within 1 hour after achieving MODE 5 with pressurizer level < 90 inches. This provides a short period of time to verify compliance after the conditions are attained. Additionally, each SR must be completed once each 12 hours after the initial verification. The Frequency of 12 hours is considered reasonable, in view of other administrative controls available and operating experience.

REFERENCES

1. Updated Final Safety Analysis Report (UFSAR)

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.2 Reactivity Balance

BASES

BACKGROUND

According to Reference 1, Appendix 1C, Criteria 27, 29, and 30, reactivity shall be controllable, such that, subcriticality is maintained under cold conditions, and acceptable fuel design limits are not exceeded during normal operation and AOOs. 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, CEA 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) 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 critical boron curve, which provides an indication of the soluble boron concentration in the RCS versus cycle burnup. Periodic measurement of the RCS boron concentration for comparison with the predicted value with other variables fixed (such as CEA 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.

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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 hot full power, the excess positive reactivity is compensated by burnable absorbers (if any), CEAs, whatever neutron poisons (mainly xenon and samarium) are present in the fuel, and the RCS boron concentration.

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 critical boron curve is based on steady state operation at RATED THERMAL POWER (RTP). Therefore, deviations from the predicted critical boron curve may indicate deficiencies in the design analysis, deficiencies in the calculational models, or abnormal core conditions, and must be evaluated.

APPLICABLE SAFETY ANALYSES

Accurate prediction of core reactivity is either an explicit or implicit assumption in the accident analysis evaluations. Most accident evaluations (Reference 1, Section 14.1) are, therefore, dependent upon accurate evaluation of core reactivity. In particular, SDM and reactivity transients, such as CEA withdrawal accidents or CEA 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 calculational models used to predict core reactivity. If the measured and predicted RCS boron concentrations for identical core conditions at beginning-of-cycle (BOC) do not agree, 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 BOC, the prediction may be normalized to the measured boron concentration. Thereafter, any significant deviations in the measured boron concentration from the predicted critical boron curve that develop during fuel depletion may be an indication that the calculational model is not adequate for core burnups beyond BOC, or that an unexpected change in core conditions has occurred.

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 CEAs in their normal positions for power operation. The normalization is performed at BOC conditions, so that core reactivity relative to predicted values can be continually monitored and evaluated as core conditions change during the cycle.

The reactivity balance satisfies 10 CFR 50.36(c)(2)(ii), Criterion 2.

LCO

The reactivity balance limit is established to ensure plant operation is maintained within the assumptions of the safety analyses. 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

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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.

APPLICABILITY

The limits on core reactivity must be maintained during MODE 1 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 MODE 2 because enough operating margin exists to limit the effects of a reactivity anomaly, and THERMAL POWER is low enough ($\leq$ 5% RTP) such that reactivity anomalies are unlikely to occur. 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) 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, or CEA replacement, or shuffling).

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 to 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, 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, 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, the boron letdown curve may be renormalized, and power operation may continue. If operational restrictions or additional SRs are necessary to ensure the reactor core is acceptable for continued operation, they must be defined.

The required Completion Time of 7 days is adequate for preparing whatever operating restrictions or SRs 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 2 within 6 hours. The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 2 from full power conditions in an orderly manner and without challenging plant systems.

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 CEA position, moderator

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temperature, fuel temperature, fuel depletion, xenon concentration, and samarium concentration. The SR is performed prior to entering MODE 1 as an initial check on core conditions and design calculations at BOC and every 31 days after 60 effective full power days (EFPD). The SR is modified by two Notes. The Note in the SR column indicates that the normalization of predicted core reactivity to the measured value may take place within the first 60 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 of 31 EFPD 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 (e.g., quadrant power tilt ratio, etc.) for prompt indication of an anomaly. The Frequency Note, "only required after 60 EFPD after each fuel loading," is added to the Frequency column to allow this.

REFERENCES

1. UFSAR

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.3 Moderator Temperature Coefficient (MTC)

BASES

BACKGROUND

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 a large 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.

Moderator temperature coefficient 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 BOC MTC is less positive than that allowed by the LCO. 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 (burnable poison) to yield an MTC at the BOC within the range analyzed in the plant accident analysis. The end-of-cycle (EOC) 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 EOC limit.

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 (Reference 1, Section 14.2.2); and

b. The MTC must be such that inherently stable power operations result during normal operation and during accidents, such as overheating and overcooling events.

Reference 2, Section 14.1.2 contains analyses of accidents that result in both overheating and overcooling of the reactor core. Moderator temperature coefficient 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, such as very large soluble boron concentrations, to ensure the accident results are bounding.

Accidents that cause core overheating, either by decreased heat removal or increased power production, must be evaluated for results when the MTC is positive. Reactivity accidents that cause increased power production include the CEA withdrawal transient from either zero or full THERMAL POWER. The limiting overheating event relative to plant response is based on the maximum difference between core power and steam generator heat removal during a transient. The most limiting event with respect to a positive MTC is a CEA withdrawal accident from zero power, also referred to as a startup accident (Reference 1, Section 14.2.2).

Accidents that cause core overcooling must be evaluated for results when the MTC is most negative. The event that produces the most rapid cooldown of the RCS, and is therefore the most limiting event with respect to the negative MTC, is a steam line break (SLB) event. Following the reactor trip for the postulated EOC SLB event, the large moderator temperature reduction combined with the large negative MTC may produce reactivity increases that are as much as the shutdown reactivity. When this occurs, a substantial fraction of core power is produced with all CEAs inserted, except the most reactive one, which is assumed withdrawn. Even if the reactivity increase produces slightly subcritical conditions, a large fraction of core power may be produced through the effects of subcritical neutron multiplication.

Moderator temperature coefficient values are bounded in reload safety evaluations assuming steady state conditions at BOC and EOC. A middle-of-cycle (MOC) measurement is conducted at conditions when the RCS boron concentration reaches approximately 550 ppm. The measured value may be

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extrapolated to project the EOC value, in order to confirm reload design predictions.

The MTC satisfies 10 CFR 50.36(c)(2)(ii), Criterion 2.

LCO

Limiting Condition for Operation 3.1.3 requires the MTC to be within specified limits of the Core Operating Limits Report (COLR), with the maximum positive limit specified in Figure 3.1.3-1, to ensure 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. The limit on a positive MTC ensures that core overheating accidents will not violate the accident analysis assumptions. The negative MTC limit for EOC specified in the COLR ensures that core overcooling accidents will not violate the accident analysis assumptions.

Moderator temperature coefficient is a core physics parameter determined by the fuel and fuel cycle design and cannot be easily controlled once the core design is fixed. During operation, therefore, the LCO can only be ensured through measurement. The surveillance checks at BOC and MOC on an MTC provide confirmation that the MTC is behaving as anticipated, so that the acceptance criteria are met.

APPLICABILITY

In MODE 1, the limits on the 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, the limits must also be maintained to ensure startup accidents, such as the uncontrolled CEA or group withdrawal, will not violate the assumptions of the accident analysis. In MODEs 3, 4, 5, and 6, this LCO is not applicable, since no DBAs using the MTC as an analysis assumption are initiated from these MODEs. However, the variation of the MTC, with temperature in MODEs 3, 4, and 5 for DBAs initiated in MODEs 1 and 2, is accounted for in the accident analysis. The variation of the MTC, with temperature assumed in the safety analysis, is accepted as valid once the BOC and MOC measurements are used for normalization.

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ACTIONS

A.1

Moderator temperature coefficient is a function of the fuel and fuel cycle designs, and cannot be controlled directly once the designs have been implemented in the core. If MTC exceeds its limits, the reactor must be placed in MODE 3. This eliminates the potential for violation of the accident analysis bounds. The associated Completion Time of 6 hours is reasonable, considering the probability of an accident occurring during the time period that would require an MTC value within the LCO limits, and the time for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.3.1 and SR 3.1.3.2

The SRs for measurement of the MTC at the beginning and middle of each fuel cycle provide for confirmation of the limiting MTC values. The MTC changes smoothly from most positive (least negative) to most negative value during fuel cycle operation, as the RCS boron concentration is reduced to compensate for fuel depletion. The requirement for measurement prior to operation > 5% RTP satisfies the confirmatory check on the most positive (least negative) MTC value. The requirement for measurement, within 7 EFPD of initially reaching an equilibrium condition with THERMAL POWER ≥ 90% RTP, and within 7 days after reaching 2/3 core burnup, satisfies the confirmatory check of the most negative MTC value. The EOC measurement is performed at any THERMAL POWER, so that the projected EOC MTC may be evaluated before the reactor actually reaches the EOC condition. Moderator temperature coefficient values may be extrapolated and compensated to permit direct comparison to the specified MTC limits.

Surveillance Requirement 3.1.3.2 is modified by a Note, which indicates that if the extrapolated MTC is more negative than the EOC COLR limit, the SR may be repeated, and that shutdown must occur prior to exceeding the minimum allowable boron concentration at which MTC is projected to exceed the lower limit. An engineering evaluation is performed if the extrapolated value of MTC exceeds the Specification limits.

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REFERENCES 1. UFSAR

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.4 Control Element Assembly (CEA) Alignment

BASES

BACKGROUND

The OPERABILITY (e.g., trippability) of the shutdown and regulating CEAs is an initial assumption in all safety analyses that assume CEA insertion upon reactor trip.

The applicable criteria for these reactivity and power distribution design requirements are found in Reference 1, Appendix 1C, Criteria 6, 27, 29, and 30, and Reference 2.

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

Limits on CEA alignment and OPERABILITY have been established, and all CEA 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.

Control element assemblies are moved by their control element drive mechanisms (CEDMs). Each CEDM moves its CEA one step (approximately 3/4-inch) at a time.

The CEAs are arranged into groups that are radially symmetric. Therefore, movement of the CEA groups do not introduce radial asymmetries in the core power distribution. The shutdown and regulating CEAs provide the required reactivity worth for immediate reactor shutdown upon a reactor trip. The regulating CEAs also provide reactivity (power level) control during normal operation and transients.

The axial position of shutdown and regulating CEAs is indicated by two separate and independent systems, which are

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the Plant Computer CEA Position Indication System and the Reed Switch Position Indication System.

The Plant Computer CEA Position Indication System counts the commands sent to the CEA gripper coils from the CEDM Control System that moves the CEAs. There is a one step counter for each CEA. Individual CEAs in a group all receive the same signal to move and should, therefore, all be at the same position indicated by the group step counter for that group. Plant Computer CEA Position Indication System is considered highly precise (± 1 step or ± 3/4-inch). If a CEA does not move one step for each command signal, the step counter will still count the command and incorrectly reflect the position of the CEA.

The Reed Switch Position Indication System provides a highly accurate indication of actual CEA position, but at a lower precision than the step counters. This system is based on inductive analog signals from a series of reed switches spaced along a tube with a center-to-center distance of 1.5 inches, which is two steps. To increase the reliability of the system, there are redundant reed switches at each position.

APPLICABLE SAFETY ANALYSES

Control element assembly misalignment accidents are analyzed in the safety analysis (Reference 1, Sections 14.2, 14.11, and 14.13). The accident analysis defines CEA misoperation as any event, with the exception of sequential group withdraws, which could result from a single malfunction in the reactivity control systems. For example, CEA misalignment may be caused by a malfunction of the CEDM, CEDM Control System, or by operator error. A stuck CEA may be caused by mechanical jamming of the CEA fingers or of the gripper. A dropped CEA could be caused by an electrical failure in the CEA coil power programmers.

The acceptance criteria for addressing CEA inoperability/ misalignment are that:

a. There shall be no violations of:

    1. SAFDLs, or

    2. RCS pressure boundary integrity; and

b. The core must remain subcritical after accidents or transients.

Two types of misalignment are distinguished in the safety analysis (Reference 1, Appendix 1C). The first type of misalignment occurs if one CEA 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 remaining CEAs to meet the SDM requirement with the maximum worth CEA stuck fully withdrawn. If a CEA is stuck in the fully withdrawn position, its worth is added to the SDM requirement, since the safety analysis does not take two stuck CEAs into account. The second type of misalignment occurs when one CEA drops partially or fully into the reactor core. This event causes an initial power reduction followed by a return toward the original power, due to positive reactivity feedback from the negative MTC. Increased peaking during the power increase may result in excessive local LHRs (Reference 1, Section 14.14).

None of the above CEA misoperations will result in an automatic reactor trip. In the case of the full-length CEA drop, a prompt decrease in core average power and a distortion in radial power are initially produced, which, when conservatively coupled, result in a local power and heat flux increase, and a decrease in DNBR parameters.

The results of the CEA misoperation analysis show that, during the most limiting misoperation events, no violations of the SAFDLs, fuel centerline temperature, or RCS pressure occur.

Control element assembly alignment satisfies 10 CFR 50.36(c)(2)(ii), Criteria 2 and 3.

LCO

The limits on shutdown and regulating CEA alignments ensure that the assumptions in the safety analysis will remain valid. The requirements on OPERABILITY ensure that upon reactor trip, the CEAs will be available and will be inserted to provide enough negative reactivity to shut down the reactor. The OPERABILITY requirements also ensure that

BASES

the CEA banks maintain the correct power distribution and CEA alignment.

The requirement is to maintain the CEA alignment to within 7.5 inches between any CEA and its group.

Failure to meet the requirements of this LCO may produce unacceptable power peaking factors and LHRs, or unacceptable SDMs, all of which may constitute initial conditions inconsistent with the safety analysis.

APPLICABILITY

The requirements on CEA 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 (e.g., trippability) and alignment of CEAs have the potential to affect the safety of the plant. In MODEs 3, 4, 5, and 6, the alignment limits do not apply because the CEAs are bottomed, and the reactor is shut down and not producing fission power. In the shutdown MODEs, the OPERABILITY of the shutdown and regulating CEAs 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 for SDM in MODEs 3, 4, and 5, and LCO 3.9.1 for boron concentration requirements during refueling.

ACTIONS

A.1 and B.1

A CEA may become misaligned, yet remain trippable. In this condition, the CEA can still perform its required function of adding negative reactivity should a reactor trip be necessary.

If one or more regulating or shutdown CEAs are misaligned by > 7.5 inches and ≤ 15 inches but trippable, or one CEA is misaligned by > 15 inches but trippable, continued operation in MODEs 1 and 2 may continue, provided CEA alignment is restored within 1 hour for CEAs misaligned ≤ 15 inches and within the time specified in the COLR for CEAs misaligned > 15 inches. (The maximum time provided in the COLR is 1 hour.)

BASES

Regulating and shutdown CEA alignment is restored by either aligning the misaligned CEA(s) to within 7.5 inches of its group or aligning the misaligned CEAs group to within 7.5 inches of the misaligned CEA.

Xenon redistribution in the core starts to occur as soon as a CEA becomes misaligned. Restoring CEA alignment ensures acceptable power distributions are maintained. For small misalignments (≤ 15 inches) of the CEAs, there is:

a. A small effect on the time-dependent, long-term power distributions relative to those used in generating LCOs and limiting safety system settings setpoints;

b. A negligible effect on the available SDM; and

c. A small effect on the ejected CEA worth used in the accident analysis.

With a large CEA misalignment (> 15 inches), however, this misalignment would cause distortion of the core power distribution. This distortion may, in turn, have a significant effect on the time-dependent, long-term power distributions relative to those used in generating LCOs and limiting safety system settings setpoints.

The effect on the available SDM and the ejected CEA worth used in the accident analysis remains small.

Therefore, this condition is limited to a single CEA misalignment, while still allowing 1 hour for recovery.

In both cases, a 1-hour time period is sufficient to:

a. Identify cause of a misaligned CEA;

b. Take appropriate corrective action to realign the CEAs; and

c. Minimize the effects of xenon redistribution.

If a CEA is untrippable, it is not available for reactivity insertion during a reactor trip. With an untrippable CEA, meeting the insertion limits of LCOs 3.1.5 and 3.1.6 does not ensure that adequate SDM exists. The CEA must be returned to OPERABLE status within 1 hour or Condition C must be entered.

C.1 and C.2

If any CEA is not restored to within its alignment limits within the Completion Time provided in Required Action A.1 or B.1, an additional 2 hours is allowed to restore CEA alignment, provided THERMAL POWER is reduced ≤ 70% RTP. Prompt action must be taken to reduce THERMAL POWER, and the reduction must be completed within 1 hour. Reducing THERMAL POWER ensures acceptable power distributions are maintained during the additional time provided to restore alignment. The Completion Times are acceptable based on the reasons provided in the Bases for Required Actions A.1 and B.1.

D.1, D.2.1, and D.2.2

The CEA motion inhibit permits CEA motion within the requirements of LCO 3.1.6, and prevents regulating CEAs from being misaligned from other CEAs in the group.

Performing SR 3.1.4.1 within 1 hour and every 4 hours thereafter is considered acceptable, in view of other information continuously available to the operator in the Control Room.

With the CEA motion inhibit inoperable, a Completion Time of 6 hours is allowed for restoring the CEA motion inhibit to OPERABLE status, or fully withdrawing the CEAs in groups 3 and 4, and withdrawing all CEAs in group 5 to < 5% insertion.

Withdrawal of the CEAs to the positions required in Required Action D.2.2 provides additional assurance that core perturbations in local burnup, peaking factors, and SDM will not be more adverse than the Conditions assumed in the safety analyses and LCO setpoint determination (Reference 1, Chapter 14).

The 6-hour Completion Time takes into account Required Action D.1, the protection afforded by the CEA deviation circuits, and other information continuously available to the operator in the Control Room, so that during actual CEA motion, deviations can be detected.

Required Action D.2.2 is modified by a Note indicating that performing this Required Action is not required when in conflict with Required Actions A.1, B.1, C.2, or E.1.

E.1

When the CEA deviation circuit is inoperable, performing SR 3.1.4.1 within 1 hour and every 4 hours thereafter ensures improper CEA alignments are identified before unacceptable flux distributions occur. The specified Completion Times take into account other information continuously available to the operator in the Control Room, so that during CEA movement, deviations can be detected, and the protection provided by the CEA inhibit and deviation circuit is not required.

F.1

If any Required Action and associated Completion Time of Condition C, Condition D, or Condition E is not met, one or more regulating or shutdown CEAs are untrippable, two or more CEAs are misaligned by > 15 inches, the unit is required to be brought to MODE 3. By being brought to MODE 3, the unit is brought outside the MODE of applicability. Continued operation is not allowed in the case of more than one CEA misaligned from any other CEA in its group by > 15 inches, or one or more CEAs untrippable. This is because these cases could result in a loss of SDM and power distribution and a loss of safety function, respectively.

When a Required Action cannot be completed within the required Completion Time, a controlled shutdown should be commenced. 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 plant systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.4.1

Verification that individual CEA positions are within 7.5 inches (indicated reed switch positions) of all other CEAs in the group are performed at Frequencies of within 1 hour of any CEA movement of > 7.5 inches and every 12 hours. The CEA position verification after each movement of > 7.5 inches ensure that the CEAs in that group are properly aligned at the time when CEA misalignments are most likely to have occurred. The 12-hour Frequency allows the operator to detect a CEA that is beginning to deviate from its expected position. The specified Frequency takes into account other CEA position information that is continuously available to the operator in the Control Room, so that during CEA movement, deviations can be detected, and protection can be provided by the CEA motion inhibit and deviation circuits.

SR 3.1.4.2

Demonstrating the CEA motion inhibit OPERABLE verifies that the CEA motion inhibit is functional, even if it is not regularly operated. The verification shall ensure that the motion inhibit circuit maintains the CEA group overlap and sequencing requirements of LCO 3.1.6, and prevents any regulating CEA from being misaligned from all other CEAs in its group by > 7.5 inches (indicated position). The 31-day Frequency takes into account other information continuously available to the operator in the Control Room, so that during CEA movement, deviations can be detected, and protection can be provided by the CEA deviation circuits.

SR 3.1.4.3

Demonstrating the CEA deviation circuit is OPERABLE verifies the circuit is functional. The 31-day Frequency takes into account other information continuously available to the operator in the Control Room, so that during CEA movement, deviations can be detected, and protection can be provided by the CEA motion inhibit.

SR 3.1.4.4

Verifying each CEA is trippable would require that each CEA be tripped. In MODEs 1 and 2, tripping each CEA would result in radial or axial power tilts or oscillations. Therefore, individual CEAs are exercised every 92 days to provide increased confidence that all CEAs continue to be trippable, even if they are not regularly tripped. A movement of 7.5 inches is adequate to demonstrate motion without exceeding the alignment limit when only one CEA is being moved. For the purposes of performing the CEA operability test, if the CEA has an inoperable position indicator channel, the alternate indication system (pulse counter or voltage dividing network) will be used to monitor position. The 92-day Frequency takes into consideration other information available to the operator in the Control Room and other SRs being performed more frequently, which add to the determination of OPERABILITY of the CEAs. Between required performances of SR 3.1.4.5, if a CEA(s)is discovered to be immovable, but remains trippable and aligned, the CEA is considered to be OPERABLE. At any time, if a CEA(s) is immovable, a determination of the trippability (OPERABILITY) of the CEA(s) must be made, and appropriate action taken.

SR 3.1.4.5

Performance of a CHANNEL FUNCTIONAL TEST of each reed switch position transmitter channel ensures the channel is OPERABLE and capable of indicating CEA position over the entire length of the CEA's travel. Since this SR must be performed when the reactor is shut down, a 24-month Frequency to be coincident with refueling outages was selected. Operating experience has shown that these components usually pass this SR when performed at a Frequency of once every 24 months. Furthermore, the Frequency takes into account other SRs being performed at shorter Frequencies, which determine the OPERABILITY of the CEA Reed Switch Indication System.

SR 3.1.4.6

Verification of CEA drop times determined that the maximum CEA drop time permitted is consistent with the assumed drop time used in that safety analysis (Reference 1, Chapter 14). Control element assembly drop time is measured from the time when electrical power is interrupted to the CEDM until the CEA reaches its 90% insertion position, from a fully withdrawn position, with $T_{ave} \geq 515°F$ and all reactor coolant

B. . _S

pumps operating. Measuring drop times prior to reactor criticality, after reactor vessel head removal, ensures that reactor internals and CEDM will not interfere with CEA motion or drop time, and that no degradation in these systems has occurred that would adversely affect CEA motion or drop time. Individual CEAs whose drop times are greater than safety analysis assumptions are not OPERABLE. This SR is performed prior to criticality, based on the need to perform this SR under the conditions that apply during a unit outage and because of the potential for an unplanned unit transient if the SR were performed with the reactor at power.

REFERENCES

1. UFSAR
2. 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plants"

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.5 Shutdown Control Element Assembly (CEA) Insertion Limits

BASES

BACKGROUND

The insertion limits of the shutdown CEAs are initial assumptions in all safety analyses that assume CEA insertion upon reactor trip. The insertion limits directly affect core power distributions and assumptions of available SDM, ejected CEA worth, and initial reactivity insertion rate.

The applicable criteria for these reactivity and power distribution design requirements are in Reference 1, Appendix 1C, Criteria 6, 27, 28, 29, and 30, and Reference 2. Limits on shutdown CEA insertion have been established, and all CEA positions are monitored and controlled during power operation to ensure that the reactivity limits, ejected CEA worth, and SDM limits are preserved.

The shutdown CEAs are arranged into groups that are radially symmetric. Therefore, movement of the shutdown CEAs does not introduce radial asymmetries in the core power distribution. The shutdown and regulating CEAs provide the required reactivity worth for immediate reactor shutdown upon a reactor trip.

The design calculations are performed with the assumption that the shutdown CEAs are withdrawn prior to the regulating CEAs. The shutdown CEAs can be fully withdrawn without the core going critical. The shutdown CEAs are controlled manually by the Control Room operator. During normal unit operation, the shutdown CEAs are fully withdrawn. The shutdown CEAs must be completely withdrawn from the core prior to withdrawing any regulating CEAs during an approach to criticality. The shutdown CEAs are left in this position until the reactor is shut down. They affect core power, burnup distribution, and add negative reactivity to shut down the reactor upon receipt of a reactor trip signal.

BASES

APPLICABLE SAFETY ANALYSES

Accident analysis assumes that the shutdown CEAs are fully withdrawn any time the reactor is critical. This ensures that:

a. The minimum SDM is maintained; and

b. The potential effects of a CEA ejection accident are limited to acceptable limits.

Control element assemblies are considered fully withdrawn at 129 inches, since this position places them outside the active region of the core.

On a reactor trip, all CEAs (shutdown and regulating), except the most reactive CEA, are assumed to insert into the core. The shutdown and regulating CEAs shall be at or above their insertion limits and available to insert the required amount of negative reactivity on a reactor trip signal. The regulating CEAs may be partially inserted in the core as allowed by LCO 3.1.6. The shutdown CEA insertion limit is 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) following a reactor trip from full power. The combination of regulating CEAs and shutdown CEAs (less the most reactive CEA, 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 (Reference 1, Sections 3.2 and 3.4). The shutdown CEA insertion limit also limits the reactivity worth of an ejected shutdown CEA.

The acceptance criteria for addressing shutdown CEA, as well as regulating CEA insertion limits and inoperability or misalignment, are that:

a. There be no violation of:

   1. SAFDLs, or

   2. RCS pressure boundary damage; and

b. The core remains subcritical after accident transients.

BASES

As such, the shutdown CEA insertion limits affect safety analyses involving core reactivity, ejected CEA worth, and SDM (Reference 1, Section 14.1.2).

The shutdown CEA insertion limits satisfy 10 CFR 50.36(c)(2)(ii), Criterion 2.

LCO

The shutdown CEAs 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.

APPLICABILITY

The shutdown CEAs must be within their insertion limits, with the reactor in MODEs 1 and 2. The Applicability in MODE 2 begins anytime any regulating CEA is not fully inserted. 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. In MODE 3, 4, 5, or 6, the shutdown CEAs are fully inserted in the core and contribute to the SDM. Refer to LCO 3.1.1 for SDM requirements in MODEs 3, 4, and 5. Limiting Condition for Operation 3.9.1 ensures adequate SDM in MODE 6.

This LCO has been modified by a Note indicating the LCO requirement is suspended during SR 3.1.4.4. This SR verifies the freedom of the CEAs to move, and requires the shutdown CEAs to move below the LCO limits, which would normally violate the LCO.

ACTIONS

A.1

When one shutdown CEA is withdrawn ≥ 121.5 inches and < 129 inches, the accumulated times the shutdown CEAs have been withdrawn within this range must be verified. The Completion Time for this action is once within 4 hours and 24 hours thereafter. Operation is allowed for 7 consecutive days and a total of 14 days per 365 days. The peaking factors may not be outside required limits when one shutdown CEA is misaligned; therefore, continued operation is allowed. Since the power distribution limits are being maintained via the LCOs of Technical Specification Section 3.2, any out-of-limit peaking factor conditions will require entry into the Actions of the appropriate Section 3.2 LCO(s). The limits on consecutive days and total days in this condition reflect that the core may be approaching the acceptable limits placed on operation with flux patterns outside those assumed in the long-term burnup assumptions. Therefore, operation in this condition cannot continue and the CEA is required to be restored per Action B. The accumulated times are required to be verified once within 4 hours to determine which accumulated time limit is more limiting. The periodic Completion Time of 24 hours after the initial completion within 4 hours is adequate to ensure that the accumulated time limits are not exceeded.

B.1

Prior to entering this condition, the shutdown CEAs were fully withdrawn or all but one shutdown CEA was withdrawn ≥ 129 inches. If one shutdown CEA is withdrawn ≥ 121.5 inches and < 129 inches for > 7 days per occurrence or > 14 days per 365 days, or one shutdown CEA withdrawn < 121.5 inches, or two or more shutdown CEAs withdrawn < 129 inches, the out-of-limit CEAs must be restored to within limits within 2 hours. The Completion Time of 2 hours reflects that the power distribution limits may be outside required limits and that the core may be approaching the acceptable limits placed on operation within flux patterns outside those assumed in the long-term burnup assumptions.

The CEA(s) must be restored to within limits within 2 hours. The 2-hour total Completion Time allows the operator adequate time to adjust the CEA(s) in an orderly manner and is consistent with the required Completion Times in LCO 3.1.4.

C.1

When Required Action A.1 or B.1 cannot be met or completed within the required Completion Time, a controlled shutdown should be commenced. 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 plant systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.5.1

Verification that the shutdown CEAs are within their insertion limits prior to an approach to criticality ensures that when the reactor is critical, or being taken critical, the shutdown CEAs 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 CEAs are withdrawn before the regulating CEAs are withdrawn during a unit startup.

Since the shutdown CEAs are positioned manually by the Control Room operator, verification of shutdown CEA position at a Frequency of 12 hours is adequate to ensure that the shutdown CEAs are within their insertion limits. Also, the 12-hour Frequency takes into account other information available to the operator in the Control Room for the purpose of monitoring the status of the shutdown CEAs.

REFERENCES

1. UFSAR
2. 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plants"

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.6 Regulating Control Element Assembly (CEA) Insertion Limits

BASES

BACKGROUND

The insertion limits of the regulating CEAs are initial assumptions in all safety analyses that assume CEA insertion upon reactor trip. The insertion limits directly affect core power distributions, assumptions of available SDM, and initial reactivity insertion rate. The applicable criteria for these reactivity and power distribution design requirements are Reference 1, Appendix 1C, Criteria 27, 29, 30, and 31, and Reference 2.

Limits on regulating CEA insertion have been established, and all CEA positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking, ejected CEA worth, reactivity insertion rate, and SDM limits are preserved.

The regulating CEA groups operate with a predetermined amount of position overlap, in order to approximate a linear relation between CEA worth and CEA position (integral CEA worth). The regulating CEA groups are withdrawn and operate in a predetermined sequence. The group sequence and overlap limits are specified in the COLR. Regulating CEAs are considered to be fully withdrawn when withdrawn to at least 129.0 inches.

The regulating CEAs are used for precise reactivity control of the reactor. The positions of the regulating CEAs are 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 to maintain SAFDLs, including limits that preserve the criteria specified in Reference 2. Together, LCOs 3.1.6, 3.2.4, and LCO 3.2.5 provide limits on control component operation and on monitored process variables to ensure the core operates within the LHR (LCO 3.2.1); Total Planar Radial Peaking Factor (F^T_{xy}) (LCO 3.2.2); and Total Integrated Radial Peaking Factor (F^T_r) (LCO 3.2.3) limits in the COLR. Operation within the LHR limits given in the COLR prevents power peaks that would exceed the loss of coolant accident (LOCA) limits derived by the Emergency Core Cooling System analysis. Operation within the F_{xy}^T and F_r^T limits given in the COLR prevents departure from nucleate boiling (DNB) during a loss of forced reactor coolant flow accident. In addition to the LHR, F_{xy}^T, and F_r^T limits, certain reactivity limits are preserved by regulating CEA insertion limits. The regulating CEA insertion limits also restrict the ejected CEA worth to the values assumed in the safety analysis and preserve the minimum required SDM in MODEs 1 and 2.

The regulating CEA insertion and alignment limits are process variables that together characterize and control the three-dimensional power distribution of the reactor core. Additionally, the regulating bank insertion limits control the reactivity that could be added in the event of a CEA ejection accident, and the shutdown and regulating 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 LOCA, loss of flow, ejected CEA, or other accident requiring termination by a Reactor Protective System trip function.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation (Condition I) and AOOs (Condition II). The acceptance criteria for the regulating CEA insertion, ASI, F_{xy}^T, F_r^T, LHR, and AZIMUTHAL POWER TILT (T_q) LCOs are such as to preclude core power distributions from occurring that would violate the following fuel design criteria:

a. During a large break LOCA, the peak cladding temperature must not exceed a limit of 2200°F (Reference 2);

b. During a loss of forced reactor coolant flow accident, 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 a DNB condition;

c. During an ejected CEA accident, the energy input to the fuel must not exceed accepted limits (Reference 1, Section 14.3); and

d. The CEAs must be capable of shutting down the reactor with a minimum required SDM, with the highest worth CEA stuck fully withdrawn, Reference 1, Appendix 1C, Criterion 29.

Regulating CEA position, ASI, F_{xy}^T, F_r^T, LHR, and T_q are process variables that together characterize and control the three-dimensional power distribution of the reactor core.

Fuel cladding damage does not normally occur when the core is operated outside these LCOs during normal operation. However, fuel cladding damage could result if an accident or AOO occurs with simultaneous violation of one or more of these LCOs. Changes in the power distribution can cause increased power peaking and corresponding increased local LHRs.

The SDM requirement is ensured by limiting the regulating and shutdown CEA insertion limits, so that the allowable inserted worth of the CEAs is such that sufficient reactivity is available to shut down the reactor to hot zero power. SHUTDOWN MARGIN assumes the maximum worth CEA remains fully withdrawn upon trip (Reference 1, Section 3.4).

The most limiting SDM requirements for MODEs 1 and 2 conditions at BOC are determined by the requirements of several transients, e.g., Loss of Flow, Seized Rotor, etc. However, the most limiting SDM requirements for MODEs 1 and 2 at EOC come from just one transient, SLB. The requirements of the SLB event at EOC for both the full power and no load conditions are significantly larger than those of any other event at that time in cycle and, also, considerably larger than the most limiting requirements at BOC.

Although the most limiting SDM requirements at EOC are much larger than those at BOC, the available SDMs obtained via the scramming of the CEAs are also substantially larger due

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to the much lower boron concentration at EOC. To verify that adequate SDMs are available throughout the cycle to satisfy the changing requirements, calculations are performed at both BOC and EOC. It has been determined that calculations at these two times in cycle a are sufficient since the differences between available SDMs and the limiting SDM requirements are the smallest at these times in a cycle. The measurement of CEA bank worth performed as part of the Startup Testing Program demonstrates that the core has the expected shutdown capability. Consequently, adherence to LCOs 3.1.5 and 3.1.6 provides assurance that the available SDM at any time in a cycle will exceed the limiting SDM requirements at that time in a cycle.

Operation at the insertion limits or ASI limits may approach the maximum allowable linear heat generation rate or peaking factor, with the allowed T_q present. Operation at the insertion limit may also indicate the maximum ejected CEA worth could be equal to the limiting value in fuel cycles that have sufficiently high ejected CEA worths.

The regulating and shutdown CEA insertion limits ensure that safety analyses assumptions for reactivity insertion rate, SDM, ejected CEA worth, and power distribution peaking factors are preserved (Reference 1, Section 3.4).

The regulating CEA insertion limits satisfy 10 CFR 50.36(c)(2)(ii), Criterion 2.

LCO

The limits on regulating CEAs 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 CEA worth is maintained, and ensuring adequate negative reactivity insertion on trip. The overlap between regulating banks provides more uniform rates of reactivity insertion and withdrawal and is imposed to maintain acceptable power peaking during regulating CEA motion.

The power-dependent insertion limit (PDIL) alarm circuit is required to be OPERABLE for notification that the CEAs are outside the required insertion limits. The PDIL alarm

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circuit required to be OPERABLE receives its signal from the reed switch position indication system. When the PDIL alarm circuit is inoperable, the verification of CEA positions is increased to ensure improper CEA alignment is identified before unacceptable flux distribution occurs.

APPLICABILITY

The regulating CEA sequence, overlap, and physical insertion limits shall be maintained with the reactor in MODEs 1 and 2. These limits must be maintained, since they preserve the assumed power distribution, ejected CEA worth, SDM, and reactivity rate insertion assumptions. Applicability in MODEs 3, 4, and 5 is not required, since neither the power distribution nor ejected CEA worth assumptions would be exceeded in these MODEs. SHUTDOWN MARGIN is preserved in MODEs 3, 4, and 5 by adjustments to the soluble boron concentration.

This LCO has been modified by a Note indicating the LCO requirement is suspended during SR 3.1.4.4. This SR verifies the freedom of the CEAs to move, and requires the regulating CEAs to move below the LCO limits, which would normally violate the LCO.

ACTIONS

A.1 and A.2

Operation beyond the transient insertion limit may result in a loss of SDM and excessive peaking factors. The transient insertion limit should not be violated during normal operation; this violation, however, may occur during transients when the operator is manually controlling the CEAs in response to changing plant conditions. When the regulating groups are inserted beyond the transient insertion limits, actions must be taken to either withdraw the regulating groups beyond the limits or to reduce THERMAL POWER to less than or equal to that allowed for the actual CEA insertion limit. Two hours provides a reasonable time to accomplish this, allowing the operator to deal with current plant conditions while limiting peaking factors to acceptable levels.

B.1 and B.2

If the CEAs are inserted between the long-term steady state insertion limits and the transient insertion limits for intervals > 4 hours per 24 hour period, and the short-term steady state insertions are exceeded, peaking factors can develop that are of immediate concern (Reference 1, Chapter 14).

Verifying the short-term steady state insertion limits are not exceeded ensures that the peaking factors that do develop are within those allowed for continued operation. Fifteen minutes provides adequate time for the operator to verify if the short-term steady state insertion limits are exceeded.

C.1

With the regulating CEAs inserted between the long-term steady state insertion limit and the transient insertion limit, and with the core approaching the 5 EFPD per 30 EFPD or 14 EFPD per 365 EFPD limits, the CEAs must be returned to within the long-term steady state insertion limits, or the core must be placed in a condition in which the abnormal fuel burnup cannot continue. A Completion Time of 2 hours is allotted to return the CEAs to within the long-term steady state insertion limits.

The required Completion Time of 2 hours from initial discovery of a regulating CEA group outside the limits until its restoration to within the long-term steady state limits, shown on the figures in the COLR, allows sufficient time for borated water to enter the RCS from the chemical addition and makeup systems, and to cause the regulating CEAs to withdraw to the acceptable region. It is reasonable to continue operation for 2 hours after it is discovered that the 5-day or 14-day EFPD limit has been exceeded. This Completion Time is based on limiting the potential xenon redistribution, the low probability of an accident, and the steps required to complete the action.

D.1

When the PDIL alarm circuit is inoperable, performing SR 3.1.6.1 within 1 hour and once per 4 hours thereafter ensures improper CEA alignments are identified before unacceptable flux distributions occur.

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E.1

When a Required Action cannot be completed within the required Completion Time, a controlled shutdown should be commenced. 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 plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1

With the PDIL alarm circuit OPERABLE, verification of each regulating CEA group position every 12 hours is sufficient to detect CEA positions that may approach the acceptable limits, and to provide the operator with time to undertake the Required Action(s) should the sequence or insertion limits be found to be exceeded. The 12-hour Frequency also takes into account the indication provided by the PDIL alarm circuit and other information about CEA group positions available to the operator in the Control Room.

SR 3.1.6.2

Verification of the accumulated time of CEA group insertion between the long-term steady state insertion limits and the transient insertion limits ensures the cumulative time limits are not exceeded. The 24-hour Frequency ensures the operator identifies a time limit that is being approached before it is reached.

SR 3.1.6.3

Demonstrating the PDIL alarm circuit OPERABLE verifies that the PDIL alarm circuit is functional. The 31-day Frequency takes into account other SRs being performed at shorter Frequencies that identify improper CEA alignments.

REFERENCES

1. UFSAR
2. 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plants" 10 CFR 50.46

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.7 Special Test Exception (STE)-SHUTDOWN MARGIN (SDM)

BASES

BACKGROUND

The primary purpose of the SDM STE is to permit relaxation of existing LCOs to allow the performance of certain PHYSICS TESTS. These tests are constructed to determine the CEA worth.

Reference 1, Appendix B, Section XI requires that a test program be established to ensure that structures, systems, and components will perform satisfactorily in service. All functions necessary to ensure that specified design conditions are not exceeded during normal operation and AOOs must be tested. Testing is required as an integral part of the design, fabrication, construction, and operation of the power plant. Requirements for notification of the Nuclear Regulatory Commission, for the purpose of conducting tests and experiments, are specified in Reference 1, 10 CFR 50.59.

The key objectives of a test program (Reference 2) are to:

a. Ensure that the facility has been adequately designed;

b. Validate the analytical models used in design and analysis;

c. Verify assumptions used for predicting plant response;

d. Ensure that installation of equipment in the facility has been accomplished in accordance with the design; and

e. Verify that operating and emergency procedures are adequate.

To accomplish these objectives, testing is required prior to initial criticality, after each refueling shutdown, and during startup, low power operation, power ascension, and at power operation. 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 (Reference 3, Section 13.4).

PHYSICS TESTS' procedures are written and approved in accordance with an established process. The procedures include all information necessary to permit a detailed execution of testing required to ensure that the design intent is met. PHYSICS TESTS are performed in accordance with these procedures, and test results are independently reviewed prior to continued power escalation and long- term power operation. Examples of PHYSICS TESTS include determination of critical boron concentration, CEA group worths, reactivity coefficients, flux symmetry, and core power distribution.

APPLICABLE SAFETY ANALYSES

It is acceptable to suspend certain LCOs for PHYSICS TESTS because fuel damage criteria are not exceeded. Even if an accident occurs during PHYSICS TESTS with one or more LCOs suspended, fuel damage criteria are preserved because adequate limits on power distribution and shutdown capability are maintained during PHYSICS TESTS.

Reference 2 defines the requirements for initial testing of the facility, including PHYSICS TESTS. Requirements for reload fuel cycle PHYSICS TESTS are defined in the UFSAR Reference 3, Section 13.4. Although these PHYSICS TESTS are generally accomplished within the limits of all 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. As long as the LHR remains within its limit, fuel design criteria are preserved.

In this test, the following LCOs are suspended:

a. LCO 3.1.1; and

b. LCO 3.1.6.

Therefore, this LCO places limits on the minimum amount of CEA worth required to be available for reactivity control when CEA worth measurements are performed.

The individual LCOs cited above govern SDM CEA group height, insertion, and alignment. Additionally, the LCOs governing RCS flow, reactor inlet temperature, and pressurizer pressure contribute to maintaining DNB parameter limits. The initial condition criteria for accidents sensitive to core power distribution are preserved by the LHR and DNB parameter limits. The criteria for the LOCA are specified in Reference 2, 10 CFR 50.46. The criteria for the loss of forced reactor coolant flow accident are specified in Reference 3, Chapter 14. Operation within the LHR limit preserves the LOCA criteria; operation within the DNB parameter limits preserves the loss of flow criteria.

Surveillance tests are conducted as necessary to ensure that LHR and DNB parameters remain within limits during PHYSICS TESTS. Performance of these SRs allows PHYSICS TESTS to be conducted without decreasing the margin of safety.

Requiring that shutdown reactivity equivalent to at least the highest estimated CEA worth (of those CEAs actually withdrawn) be available for trip insertion from the OPERABLE CEA provides a high degree of assurance that shutdown capability is maintained for the most challenging postulated accident, a stuck CEA. When LCO 3.1.1 is suspended, there is not the same degree of assurance during this test that the reactor would always be shut down if the highest worth CEA was stuck out and calculational uncertainties or the estimated highest CEA worth was not as expected (the single failure criterion is not met). This situation is judged acceptable, however, because SAFDLs are still met. The risk of experiencing a stuck CEA and subsequent criticality is reduced during this PHYSICS TESTS exception by the Surveillance Requirements; and by ensuring that shutdown reactivity is available, equivalent to the reactivity worth of the estimated highest worth withdrawn CEA (Reference 3, Chapter 3).

PHYSICS TESTS include measurement of core parameters or exercise of control components that affect process variables. Among the process variables involved are total planar radial peaking factor, total integrated radial peaking factor, T_q and ASI, which represent initial condition input (power peaking) to the accident analysis. Also involved are the shutdown and regulating CEAs, which affect power peaking and are required for shut down of 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 STE LCOs is optional and, therefore, no criteria of 10 CFR 50.36(c)(2)(ii) apply. Special 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.

LCO

This LCO provides that a minimum amount of CEA worth is immediately available for reactivity control when CEA worth measurement tests are performed. The STE is required to permit the periodic verification of the actual versus predicted worth of the regulating and shutdown CEAs. The SDM requirements of LCO 3.1.1, the shutdown CEA insertion limits of LCO 3.1.5, and the regulating CEA insertion limits of LCO 3.1.6 may be suspended.

APPLICABILITY

This LCO is applicable in MODEs 2 and 3. Although CEA worth testing is conducted in MODE 2, sufficient negative reactivity is inserted during the performance of these tests to result in temporary entry into MODE 3. Because the intent is to immediately return to MODE 2 to continue CEA worth measurements, the STE allows limited operation to 6 consecutive hours in MODE 3, as indicated by the Note, without having to borate to meet the SDM requirements of LCO 3.1.1.

ACTIONS

A.1

With any CEA not fully inserted and less than the minimum required reactivity equivalent available for insertion, or with all CEAs inserted and the reactor subcritical by less than the reactivity equivalent of the highest worth CEA, restoration of the minimum SDM requirements must be accomplished by increasing the RCS boron concentration. The boration flow rate shall be ≥ 40 gpm and the boron concentration shall be ≥ 2300 ppm boric acid solution or equivalent. The required Completion Time of immediately is required to meet the assumptions of the safety analysis. It is assumed that boration will be continued until the SDM requirements are met.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.7.1

Verification of the position of each partially or fully withdrawn full-length or part-length CEA is necessary to ensure that the minimum negative reactivity requirements for insertion on a trip are preserved. A 2-hour Frequency is sufficient for the operator to verify that each CEA position is within the acceptance criteria.

SR 3.1.7.2

Prior demonstration that each CEA to be withdrawn from the core during PHYSICS TESTS is capable of full insertion, when tripped from at least a 50% withdrawn position, ensures that the CEA will insert on a trip signal. The Frequency ensures that the CEAs are OPERABLE prior to reducing SDM to less than the limits of LCO 3.1.1.

REFERENCES

1. 10 CFR Part 50
2. Regulatory Guide 1.68, Revision 2, "Initial Test Programs for Water-Cooled Nuclear Power Plants," August 1978
3. UFSAR

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.8 Special Test Exceptions (STE)-MODEs 1 and 2

BASES

BACKGROUND

The primary purpose of these MODEs 1 and 2 STEs is to permit relaxation of existing LCOs to allow the performance of certain PHYSICS TESTS. These tests are conducted to determine specific reactor core characteristics.

Reference 1, Appendix B, Section XI requires that a test program be established to ensure that structures, systems, and components will perform satisfactorily in service. All functions necessary to ensure that specified design conditions are not exceeded during normal operation and AOOs must be tested. Testing is required as an integral part of the design, fabrication, construction, and operation of the power plant. Requirements for notification of the Nuclear Regulatory Commission, for the purpose of conducting tests and experiments, are specified in Reference 1, 10 CFR 50.59.

The key objectives of a test program (Reference 2) are to:

a. Ensure that the facility has been adequately designed;

b. Validate the analytical models used in design and analysis;

c. Verify assumptions used for predicting plant response;

d. Ensure that installation of equipment in the facility has been accomplished in accordance with design; and

e. Verify that operating and emergency procedures are adequate.

To accomplish these objectives, testing is required prior to initial criticality, after each refueling shutdown, and during startup, low power operation, power ascension, and at power operation. 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 (Reference 3, Section 13.4).

PHYSICS TESTS procedures are written and approved in accordance with established formats. The procedures include all information necessary to permit a detailed execution of

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testing required to ensure that 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.

Examples of PHYSICS TESTS include determination of critical boron concentration, CEA group worths, reactivity coefficients, flux symmetry, and core power distribution.

APPLICABLE SAFETY ANALYSES

It is acceptable to suspend certain LCOs for PHYSICS TESTS because fuel damage criteria are not exceeded. Even if an accident occurs during a PHYSICS TESTS with one or more LCOs suspended, fuel damage criteria are preserved because the limits on power distribution and shutdown capability are maintained during PHYSICS TESTS.

Reference 3, Section 13.4 defines the requirements for initial testing of the facility, including PHYSICS TESTS. Although these PHYSICS TESTS are generally accomplished within the limits of all LCOs, conditions may occur when one or more LCO 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. As long as the LHR remains within its limit, fuel design criteria are preserved.

In this test, the following LCOs are suspended: LCO 3.1.3; LCO 3.1.4; LCO 3.1.5; LCO 3.1.6; LCO 3.2.2; LCO 3.2.3; and LCO 3.2.4.

The safety analysis (Reference 3, Section 13.4) places limits on allowable THERMAL POWER during PHYSICS TESTS and requires the LHR and the DNB parameter to be maintained within limits.

The individual LCOs governing CEA group height, insertion and alignment, ASI, F^T_{xy}, F^T_r, and T_q preserve the LHR limits. Additionally, the LCOs governing RCS flow, reactor inlet temperature (T_c), and pressurizer pressure contribute to maintaining DNB parameter limits. The initial condition criteria for accidents sensitive to core power distribution are preserved by the LHR and DNB parameter limits. The criteria for the LOCA are specified in Reference 1, 10 CFR 50.46. The criteria for the loss of forced reactor coolant flow accident are specified in Reference 1, 10 CFR 50.46. Operation within the LHR limit preserves the LOCA criteria; operation within the DNB parameter limits preserves the loss of flow criteria.

During PHYSICS TESTS, one or more of the LCOs that normally preserve the LHR and DNB parameter limits may be suspended. The results of the accident analysis are not adversely impacted, however, if LHR and DNB parameters are verified to be within their limits while the LCOs are suspended. Therefore, SRs are placed as necessary to ensure that LHR and DNB parameters remain within limits during PHYSICS TESTS. Performance of these SRs allows PHYSICS TESTS to be conducted without decreasing the margin of safety.

PHYSICS TESTS include measurement of core parameters or exercise of control components that affect process variables. Among the process variables involved are F^T_{xy}, F^T_r, T_q, and ASI, which represent initial condition input (power peaking) to the accident analysis. Also involved are the shutdown and regulating CEAs, which affect power peaking and are required for shut down of 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 STE LCOs is optional and, therefore, no criteria of 10 CFR 50.36(c)(2)(ii) apply. Special 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.

LCO

This LCO permits individual CEAs to be positioned outside of their normal group heights and insertion limits during the performance of PHYSICS TESTS, such as those required to:

a. Measure CEA worth;

b. Determine the reactor stability index and damping factor under xenon oscillation conditions;

c. Determine power distributions for nonnormal CEA configurations;

d. Measure rod shadowing factors; and

e. Measure temperature and power coefficients.

The requirements of LCO 3.1.3, LCO 3.1.4, LCO 3.1.5, LCO 3.1.6, LCO 3.2.2, LCO 3.2.3, and LCO 3.2.4 may be suspended during the performance of PHYSICS TESTS, provided THERMAL POWER is restricted to test power plateau, which shall not exceed 85% RTP.

APPLICABILITY

This LCO is applicable in MODEs 1 and 2 because the reactor must be critical at various THERMAL POWER levels to perform the PHYSICS TESTS described in the LCO section. Limiting the test power plateau to < 85% RTP ensures that LHRs are maintained within acceptable limits.

ACTIONS

A.1

If THERMAL POWER exceeds the test power plateau, THERMAL POWER must be reduced to restore the additional thermal margin provided by the reduction. The 15-minute Completion Time ensures that prompt action shall be taken to reduce THERMAL POWER to within acceptable limits.

B.1 and B.2

If Required Action A.1 cannot be completed within the required Completion Time, PHYSICS TESTS must be suspended within 1 hour, and the reactor must be brought to MODE 3. Allowing 1 hour for suspending PHYSICS TESTS allows the operator sufficient time to change any abnormal CEA configuration back to within the limits of LCO 3.1.4, LCO 3.1.5, and LCO 3.1.6. Bringing the reactor to MODE 3 within 6 hours increases thermal margin and is consistent with the Required Actions of the power distribution LCOs. The required Completion Time of 6 hours is adequate for performing a controlled shutdown from full power conditions in an orderly manner and without challenging plant systems, and is consistent with power distribution LCO Completion Times.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.8.1

Verifying that THERMAL POWER is equal to or less than that allowed by the test power plateau, as specified in the PHYSICS TESTS procedure and required by the safety analysis, ensures that adequate LHR and DNB parameter margins are maintained while LCOs are suspended. The 1-hour Frequency is sufficient, based on the slow rate of power change and increased operational controls in place during PHYSICS TESTS.

REFERENCES

1. 10 CFR Part 50
2. Regulatory Guide 1.68, Revision 2, "Initial Test Programs for Water-Cooled Nuclear Power Plants," August 1978
3. UFSAR

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1 Linear Heat Rate (LHR)

BASES

BACKGROUND

The purpose of this Limiting Condition for Operation (LCO) is to limit the core power distribution to the initial values assumed in the accident analyses. Operation within the limits imposed by this LCO either limits or prevents potential fuel cladding failures that could 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 accident (LOFA), ejected control element assembly (CEA) accident, or other postulated accident requiring termination by a Reactor Protective System trip function. This LCO limits the amount of damage to the fuel cladding during an accident by ensuring that the plant is operating within acceptable bounding conditions at the onset of a transient.

Methods of controlling the power distribution include:

a. Using CEAs to alter the axial power distribution;

b. Decreasing CEA insertion by boration, thereby improving the radial power distribution; and

c. Correcting less than optimum conditions (e.g., a CEA drop or misoperation of the unit) that cause margin degradations.

The core power distribution is controlled so that, in conjunction with other core operating parameters (e.g., CEA insertion and alignment limits), the power distribution satisfies this LCO. The limiting safety system settings (LSSS) and this LCO are based on the accident analyses (Reference 1, Chapter 14), so that specified acceptable fuel design limits (SAFDLs) are not exceeded as a result of anticipated operational occurrences (AOOs), and the limits of acceptable consequences are not exceeded for other postulated accidents.

Limiting power distribution skewing over time also minimizes the xenon distribution skewing, which is a significant factor in controlling the axial power distribution.

Power distribution is a product of multiple parameters, various combinations of which may produce acceptable power distributions. Operation within the design limits of power distribution is accomplished by generating operating limits on linear heat rate (LHR) and departure from nucleate boiling (DNB).

The limits on LHR, Total Planar Radial Peaking Factor (F_{xy}^T), Total Integrated Radial Peaking Factor (F_r^T), AZIMUTHAL POWER TILT (T_q), and AXIAL SHAPE INDEX (ASI) represent limits within which the LHR algorithms are valid. These limits are obtained directly from the core reload analysis.

Either of the two core power distribution monitoring systems, the Excore Detector Monitoring System or the Incore Detector Monitoring System, provides adequate monitoring of the core power distribution and is capable of verifying that the LHR is within its limits. The Excore Detector Monitoring System performs this function by continuously monitoring ASI with the OPERABLE quadrant symmetric excore neutron flux detectors and verifying that the ASI is maintained within the allowable limits specified in the Core Operating Limit Report (COLR).

In conjunction with the use of the Excore Detector Monitoring System and in establishing ASI limits, the following assumptions are made:

a. The CEA insertion limits of LCOs 3.1.5 and 3.1.6 are satisfied;

b. The T_q restrictions of LCO 3.2.4 are satisfied; and

c. F_{xy}^T is within the limits of LCO 3.2.2.

The Incore Detector Monitoring System continuously provides a more direct measure of the peaking factors and alarms that have been established for the individual incore detector segments, ensuring that the peak LHRs are maintained within the limits specified in the COLR. The setpoints for these alarms include tolerances, set in conservative directions, for:

a. A measurement calculational uncertainty factor of 1.062;

b. An engineering uncertainty factor of 1.03;

c. An allowance of 1.002 for axial fuel densification and thermal expansion; and

d. A THERMAL POWER measurement uncertainty factor of 1.02 for THERMAL POWER > 50% RATED THERMAL POWER (RTP) and 1.035 for THERMAL POWER > 20% RTP and ≤ 50% RTP.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and AOOs (Reference 1, Appendix 1C, Criterion 6). The power distribution and CEA insertion and alignment LCOs preclude core power distributions that violate the following fuel design criteria:

a. During a LOCA, peak cladding temperature must not exceed 2200°F (Reference 2);

b. During a LOFA, 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 DNB condition;

c. During an ejected CEA accident, the energy input to the fuel must not exceed the accepted limits (Reference 1, Section 14.13); and

d. The control rods must be capable of shutting down the reactor with a minimum required SHUTDOWN MARGIN (SDM) with the highest worth control rod stuck fully withdrawn (Reference 1, Appendix 1C, Criterion 29).

The power density at any point in the core must be limited to maintain the fuel design criteria (Reference 2). This is accomplished by maintaining the power distribution and reactor coolant conditions so that the peak LHR and DNB parameters are within operating limits supported by accident analyses (Reference 1, Chapter 14), with due regard for the correlations between measured quantities, the power

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distribution, and uncertainties in determining the power distribution.

Fuel cladding failure during a LOCA is limited by restricting the maximum linear heat generation rate (LHGR) so that the peak cladding temperature does not exceed 2200°F (Reference 2). High peak cladding temperatures are assumed to cause severe cladding failure by oxidation due to a Zircaloy-water reaction.

The LCOs governing LHR, ASI, and the Reactor Coolant System (RCS) ensure that these criteria are met as long as the core is operated within the ASI, F_{xy}^T, F_r^T, and T_q limits specified in the COLR. The latter are process variables that characterize the three-dimensional power distribution of the reactor core. Operation within the limits for these variables ensures that their actual values are within the ranges used in the accident analyses.

Fuel cladding damage does not normally occur while the unit is operating at conditions outside the limits of these LCOs during normal operation. Fuel cladding damage could result, however, if an accident or AOO occurs from initial conditions outside the limits of these LCOs. The potential for fuel cladding damage exists because changes in the power distribution can cause increased power peaking and can correspondingly increase local LHR.

The LHR satisfies 10 CFR 50.36(c)(2)(ii), Criterion 2.

LCO

The power distribution LCO limits are based on correlations between power peaking and certain measured variables used as inputs to the LHR and DNB ratio operating limits. The power distribution LCO limits, except T_q, are provided in the COLR. The limitation on the LHR ensures that, in the event of a LOCA, the peak temperature of the fuel cladding does not exceed 2200°F. However, fuel cladding damage does not normally occur when outside the LCO limit if an accident does not occur.

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APPLICABILITY

In MODE 1, power distribution must be maintained within the limits assumed in the accident analysis to ensure that fuel damage does not result following an AOO. In other MODEs, this LCO does not apply because there is not sufficient THERMAL POWER to require a limit on the core power distribution.

ACTIONS

A.1

With the LHR exceeding its limit, excessive fuel damage could occur following an accident. In this Condition, prompt action must be taken to restore the LHR to within the specified limits. One hour to restore the LHR to within its specified limits is reasonable and ensures that the core does not continue to operate in this Condition. The 1-hour Completion Time also allows the operator sufficient time for evaluating core conditions and for initiating proper corrective actions.

B.1

If the LHR cannot be returned to within its specified limits, THERMAL POWER must be reduced. The change to MODE 2 provides reasonable assurance that the core is operating within its thermal limits and places the core in a conservative condition. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 2 from full power MODE 1 conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

A Note was added to the Surveillance Requirements (SRs) to require LHR to be determined by either the Excore Detector Monitoring System or the Incore Detector Monitoring System.

SR 3.2.1.1

The periodic SR to verify the value of F^T_{xy} ensures that the LHR remains within the range assumed in the analysis. Determining the measured F^T_{xy} every 72 hours when the excores are used to monitor LHR ensures the power distribution parameters are within limits when full core mapping is not being used.

Performance of the SR every 72 hours of accumulated operation in MODE 1 provides reasonable assurance that unacceptable changes in the F_{xy}^{T} and LHR are promptly detected.

The SR is modified by a Note that only requires the SR to be performed when the excores are being used to determine LHR. This SR is not required when the LHR is being measured by the incores, which is a more accurate measure of Core Power Distributions.

SR 3.2.1.2

This SR requires verification that the ASI alarm setpoints are within the limits specified in the COLR. Performance of this SR ensures that the Excore Detector Monitoring System can accurately monitor the LHR, and provide alarms when LHR is not within limits. Therefore, this SR is only applicable when the Excore Detector Monitoring System is being used to determine the LHR. The F_{xy}^{T} value determined by SR 3.2.1.1 is used in the derivation of the ASI alarm setpoint specified in the COLR. The 31-day Frequency is appropriate for this SR because it is consistent with the requirements of SR 3.3.1.3 for calibration of the excore detectors using the incore detectors.

The SR is modified by a Note that states that the SR is only applicable when the Excore Detection Monitoring System is being used to determine LHR. The reason for the Note is that the excore detectors input neutron flux information into the ASI calculation.

SR 3.2.1.3 and SR 3.2.1.4

Continuous monitoring of the LHR is provided by the Incore Detector Monitoring System and the Excore Detector Monitoring System. Either of these two core power distribution monitoring systems provides adequate monitoring of the core power distribution and is capable of verifying that the LHR does not exceed its specified limits.

Performance of these SRs verifies that the Incore Detector Monitoring System can accurately monitor LHR. Therefore,

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they are only applicable when the Incore Detector Monitoring System is being used to determine the LHR.

A 31-day Frequency is consistent with the historical testing frequency of the incore detector monitoring system. The SRs are modified by two Notes. Note 1 allows the SRs to be performed only when the Incore Detector Monitoring System is being used to determine LHR. Note 2 states that the SRs are not required to be performed when THERMAL POWER is < 20% RTP. The accuracy of the neutron flux information from the incore detectors is not reliable at THERMAL POWER < 20% RTP.

REFERENCES

1. Updated Final Safety Analysis Report (UFSAR)
2. 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plants"

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.2 Total Planar Radial Peaking Factor (F_{xy}^T)

BASES

BACKGROUND

The purpose of this LCO is to limit the core power distribution to the initial values assumed in the accident analyses. Operation within the limits imposed by this LCO decreases or prevents potential fuel cladding failures that could breach the primary fission product barrier and release fission products to the reactor coolant in the event of a LOCA, LOFA, ejected CEA accident, or other postulated accident requiring termination by a Reactor Protective System trip function. This LCO limits damage to the fuel cladding during an accident by ensuring that the plant is operating within acceptable bounding conditions at the onset of a transient.

Methods of controlling the power distribution include:

a. Using CEAs to alter the axial power distribution;

b. Decreasing CEA insertion by boration, thereby improving the radial power distribution; and

c. Correcting off--optimum conditions (e.g., a CEA drop or misoperation of the unit) that cause margin degradations.

The core power distribution is controlled so that, in conjunction with other core operating parameters (e.g., CEA insertion and alignment limits), the power distribution does not result in violation of this LCO. The LSSS and this LCO are based on accident analyses (Reference 1, Chapter 14), so that SAFDLs are not exceeded as a result of AOOs and the limits of acceptable consequences are not exceeded for other postulated accidents.

Limiting power distribution skewing over time also minimizes the xenon distribution skewing, which is a significant factor in controlling the axial power distribution.

Power distribution is a product of multiple parameters, various combinations of which may produce acceptable power distributions. Operation within the design limits of power distribution is accomplished by generating operating limits on the LHR and DNB.

The limits on LHR, F^T_{xy}, Total Integrated Radial Peaking Factor (F^T_r), T_q, and ASI represent limits within which the LHR algorithms are valid. These limits are obtained directly from the core reload analysis.

Either of the two core power distribution monitoring systems, the Excore Detector Monitoring System or the Incore Detector Monitoring System, provides adequate monitoring of the core power distribution and is capable of verifying that the LHR does not exceed its limits. The Excore Detector Monitoring System performs this function by continuously monitoring the ASI with the OPERABLE quadrant symmetric excore neutron flux detectors and verifying that the ASI is maintained within the allowable limits specified in the COLR.

In conjunction with the use of the Excore Detector Monitoring System and in establishing the ASI limits, the following assumptions are made:

a. The CEA insertion limits of LCOs 3.1.5 and 3.1.6 are satisfied;

b. The T_q restrictions of LCO 3.2.4 are satisfied; and

c. F^T_{xy} does not exceed the limits of this LCO.

The Incore Detector Monitoring System provides a more direct measure of the peaking factors, and the alarms that have been established for the individual incore detector segments ensure that the peak LHRs are maintained within the limits specified in the COLR. The setpoints for these alarms include tolerances, set in conservative directions, for:

a. A measurement calculational uncertainty factor of 1.062;

b. An engineering uncertainty factor of 1.03;

c. An allowance of 1.002 for axial fuel densification and thermal expansion; and

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d. A THERMAL POWER measurement uncertainty factor of 1.02 for THERMAL POWER > 50% RTP and 1.035 for THERMAL POWER > 20% RTP and ≤ 50% RTP.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation or AOOs (Reference 1, Appendix 1C, Criterion 6). The Power Distribution and CEA Insertion and Alignment LCOs preclude core power distributions that violate the following fuel design criteria:

a. During a LOCA, peak cladding temperature must not exceed 2200°F (Reference 2);

b. During a LOFA, 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 DNB condition;

c. During an ejected CEA accident, the energy input to the fuel must not exceed the accepted limits (Reference 1, Section 14.13); 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 (Reference 1, Appendix 1C, Criterion 29).

The power density at any point in the core must be limited to maintain the fuel design criteria (Reference 2). This limiting is accomplished by maintaining the power distribution and reactor coolant conditions such that the peak LHR and DNB parameters are within operating limits supported by the accident analyses (Reference 1, Chapter 14), with due regard for the correlations between measured quantities, the power distribution, and the uncertainties in the determination of power distribution.

Fuel cladding failure during a LOCA is limited by restricting the maximum LHGR so that the peak cladding temperature does not exceed 2200°F (Reference 2). High peak cladding temperatures are assumed to cause severe cladding failure by oxidation due to a Zircaloy-water reaction.

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The LCOs governing LHR, ASI, and the RCS ensure that these criteria are met as long as the core is operated within the ASI, F^T_{xy}, F^T_r, and T_q limits specified in the COLR. The latter are process variables that characterize the three dimensional power distribution of the reactor core. Operation within the limits for these variables ensures that their actual values are within the ranges used in the accident analyses.

Fuel cladding damage does not normally occur while at conditions outside the limits of these LCOs during normal operation. Fuel cladding damage could result, however, should an accident or AOO occur from initial conditions outside the limits of these LCOs. This potential for fuel cladding damage exists because changes in the power distribution can cause increased power peaking and correspondingly increased local LHR.

F^T_{xy} satisfies 10 CFR 50.36(c)(2)(ii), Criterion 2.

LCO

The power distribution LCO limits are based on correlations between power peaking and certain measured variables used as inputs to the LHR and DNB ratio operating limits. The power distribution LCO limits, except T_q, are provided in the COLR. The limitation on LHR ensures that in the event of a LOCA the peak temperature of the fuel cladding does not exceed 2200°F.

APPLICABILITY

In MODE 1, power distribution must be maintained within the limits assumed in the accident analyses to ensure that fuel damage does not result following an AOO. In other MODEs, this LCO does not apply because there is not sufficient THERMAL POWER to require a limit on the core power distribution.

ACTIONS

A.1

The limitations on F^T_{xy} provided in the COLR ensure that the assumptions used in the analysis for establishing the LHR, LCO, and LSSS remain valid during operation at the various allowable CEA group insertion limits. If F^T_{xy} exceeds its basic limitation, six hours is provided to restore F^T_{xy} to within limits. The F^T_{xy} must be restored to within limits by either withdrawing the regulating CEA groups or by reducing THERMAL POWER. Six hours to return F^T_{xy} to within its limit is reasonable and is sufficiently short to minimize the time F^T_{xy} is not within limits.

B.1

If F^T_{xy} cannot be returned to within its limit, THERMAL POWER must be reduced. A change to MODE 2 provides reasonable assurance that the core is operating within its thermal limits and places the core in a conservative condition. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 2 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1

The periodic SR to determine the calculated F^T_{xy} ensures that F^T_{xy} remains within the range assumed in the analysis throughout the fuel cycle. Determining the measured F^T_{xy} after each fuel loading prior to the reactor exceeding 70% RTP ensures that the core is properly loaded.

Performance of the SR every 31 days of accumulated operation in MODE 1 provides reasonable assurance that unacceptable changes in the F^T_{xy} are promptly detected.

The power distribution map can only be obtained after THERMAL POWER exceeds 20% RTP because the incore detectors are not reliable below 20% RTP.

The SR is modified by a Note that requires the incore detectors to be used to determine F^T_{xy} by using them to obtain a power distribution map with all full length CEAs above the long term steady state insertion limits, as specified in the COLR. This determination is limited to core planes between 15% and 85% of full core height inclusive and still exclude regions influenced by grid effects.

BASES

REFERENCES

1. UFSAR

2. 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plants"

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3 Total Integrated Radial Peaking Factor (F_r^T)

BASES

BACKGROUND

The purpose of this LCO is to limit the core power distribution to the initial values assumed in the accident analyses. Operation within the limits imposed by this LCO either limits or prevents potential fuel cladding failures that could breach the primary fission product barrier and release fission products to the reactor coolant in the event of a LOCA, LOFA, ejected control element assembly (CEA) accident, or other postulated accident requiring termination by a Reactor Protective System trip function. This LCO limits the amount of damage to the fuel cladding during an accident by ensuring that the plant is operating within acceptable bounding conditions at the onset of a transient.

Methods of controlling the power distribution include:

a. The use of CEAs to alter the axial power distribution;

b. Decreasing CEA insertion by boration, thereby improving the radial power distribution; and

c. Correcting off-optimum conditions (e.g., a CEA drop or misoperation of the unit) that cause margin degradations.

The core power distribution is controlled so that, in conjunction with other core operating parameters (e.g., CEA insertion and alignment limits), the power distribution does not result in violation of this LCO. The LSSS and this LCO are based on the accident analyses (Reference 1, Chapter 14), so that SAFDLs are not exceeded as a result of AOOs, and the limits of acceptable consequences are not exceeded for other postulated accidents.

Limiting power distribution skewing over time also minimizes the xenon distribution skewing, which is a significant factor in controlling the axial power distribution.

Power distribution is a product of multiple parameters, various combinations of which may produce acceptable power distributions. Operation within the design limits of power distribution is accomplished by generating operating limits on the LHR and DNB.

The limits on LHR, F_{xy}^T, F_r^T, T_q, and ASI represent limits within which the LHR algorithms are valid. These limits are obtained directly from the core reload analysis.

Either of the two core power distribution monitoring systems, the Excore Detector Monitoring System or the Incore Detector Monitoring System, provide adequate monitoring of the core power distribution and are capable of verifying that the LHR does not exceed its limits. The Excore Detector Monitoring System performs this function by continuously monitoring the ASI with the OPERABLE quadrant symmetric excore neutron flux detectors and verifying that the ASI is maintained within the allowable limits specified in the COLR.

In conjunction with the use of the Excore Detector Monitoring System and in establishing the ASI limits, the following conditions are assumed:

a. The CEA insertion limits of LCOs 3.1.5 and 3.1.6 are satisfied;

b. The T_q restrictions of LCO 3.2.4 are satisfied; and

c. F_{xy}^T does not exceed the limits of LCO 3.2.2.

The Incore Detector Monitoring System continuously provides a more direct measure of the peaking factors, and the alarms established for the individual incore detector segments ensure that the peak LHRs are maintained within the limits specified in the COLR. The setpoints for these alarms include tolerances, set in conservative directions, for:

a. A measurement calculational uncertainty factor of 1.062;

b. An engineering uncertainty factor of 1.03;

c. An allowance of 1.002 for axial fuel densification and thermal expansion; and

d. A THERMAL POWER measurement uncertainty factor of 1.02 for THERMAL POWER > 50% RTP and 1.035 for THERMAL POWER > 20% RTP and ≤ 50% RTP.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and AOOs (Reference 1, Appendix 1C, Criterion 6). The power distribution and CEA insertion and alignment LCOs preclude core power distributions that violate the following fuel design criteria:

a. During a LOCA, peak cladding temperature must not exceed 2200°F (Reference 2);

b. During a LOFA, 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 DNB condition;

c. During an ejected CEA accident, the energy input to the fuel must not exceed the accepted limits (Reference 1, Section 14.13); 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 (Reference 1, Appendix 1C, Criterion 29).

The power density at any point in the core must be limited to maintain the fuel design criteria (Reference 2). This is accomplished by maintaining the power distribution and reactor coolant conditions so that the peak LHR and DNB parameters are within operating limits supported by the accident analyses (Reference 1, Chapter 14), with due regard for the correlations between measured quantities, the power distribution, and uncertainties in the determination of power distribution.

Fuel cladding failure during a LOCA is limited by restricting the maximum LHGR so that the peak cladding temperature does not exceed 2200°F (Reference 2). High peak cladding temperatures are assumed to cause severe cladding failure by oxidation due to a Zircaloy-water reaction.

The LCOs governing LHR, ASI, and the RCS ensure that these criteria are met as long as the core is operated within the

BASES

ASI, F_{xy}^T, and F_r^T limits specified in the COLR, and within the T_q limits. The latter are process variables that characterize the three-dimensional power distribution of the reactor core. Operation within the limits for these variables ensures that their actual values are within the range used in the accident analysis.

Fuel cladding damage does not normally occur while at conditions outside the limits of these LCOs during normal operation. Fuel cladding damage could result, however, if an accident or AOO occurs from initial conditions outside the limits of these LCOs. This potential for fuel cladding damage exists because changes in the power distribution cause increased power peaking and correspondingly increased local LHR.

F_r^T satisfies 10 CFR 50.36(c)(2)(ii), Criterion 2.

LCO

The LCO limits for power distribution are based on correlations between power peaking and measured variables used as inputs to LHR and DNB ratio operating limits. The LCO limits for power distribution, except T_q, are provided in the COLR. The limitation on the LHR ensures that, in the event of a LOCA, the peak temperature of the fuel cladding does not exceed 2200°F.

APPLICABILITY

In MODE 1, power distribution must be maintained within the limits assumed in the accident analysis to ensure that fuel damage does not result following an AOO. In other MODEs, this LCO does not apply because there is not sufficient THERMAL POWER to require a limit on the core power distribution.

ACTIONS

A.1

The limitations on F_r^T provided in the COLR ensure that the assumptions used in the analysis for establishing the ASI, LCO, and LSSS remain valid during operation at the various allowable CEA group insertion limits. If F_r^T exceeds its basic limitation, 6 hours is provided to restore F_r^T to within limits. The F_r^T must be restored to within limits by either withdrawing the regulating CEA groups or by reducing

BASES

THERMAL POWER. Six hours to return F_r^T to within its limits is reasonable and is sufficiently short to minimize the time F_r^T is not within limits.

B.1

If F_r^T cannot be returned to within its limit, THERMAL POWER must be reduced. A change to MODE 2 provides reasonable assurance that the core is operating within its thermal limits and places the core in a conservative condition. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 2 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.3.1

The periodic SR to determine the calculated F_r^T ensures that F_r^T remains within the range assumed in the analysis throughout the fuel cycle. Determining the measured F_r^T once after each fuel loading prior to exceeding 70% RTP ensures that the core is properly loaded.

Performance of the SR every 31 days of accumulated operation in MODE 1 provides reasonable assurance that unacceptable changes in the F_r^T are promptly detected.

The power distribution map can only be obtained after THERMAL POWER exceeds 20% RTP because the incore detectors are not reliable below 20% RTP.

The SR is modified by a Note that requires the incore detectors to be used to determine F_{xy}^T by using them to obtain a power distribution map with all full length CEAs above the long-term steady state insertion limits, as specified in the COLR.

REFERENCES

1. UFSAR
2. 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plants"

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 AZIMUTHAL POWER TILT (T_q)

BASES

BACKGROUND

The purpose of this LCO is to limit the core power distribution to the initial values assumed in the accident analyses. Operation within the limits imposed by this LCO limits or prevents potential fuel cladding failures that could breach the primary fission product barrier and release fission products to the reactor coolant in the event of a LOCA, LOFA, ejected control element assembly (CEA) accident, or other postulated accident requiring termination by a Reactor Protective System trip function. This LCO limits the amount of damage to the fuel cladding during an accident by ensuring that the plant is operating within acceptable bounding conditions at the onset of a transient.

Methods of controlling the power distribution include:

a. Using CEAs to alter the axial power distribution;

b. Decreasing CEA insertion by boration, thereby improving the radial power distribution; and

c. Correcting off-optimum conditions (e.g., a CEA drop or misoperation of the unit) that cause margin degradations.

The core power distribution is controlled so that, in conjunction with other core operating parameters (e.g., CEA insertion and alignment limits), the power distribution does not result in violation of this LCO. The LSSS and this LCO are based on the accident analyses (Reference 1, Chapter 14), so that SAFDLs are not exceeded as a result of AOOs, and the limits of acceptable consequences are not exceeded for other postulated accidents.

Limiting power distribution skewing over time also minimizes the xenon distribution skewing, which is a significant factor in controlling the axial power distribution.

Power distribution is a product of multiple parameters, various combinations of which may produce acceptable power distributions. Operation within the design limits of power distribution is accomplished by generating operating limits for LHR and DNB.

BASES

The limits on LHR, F_{xy}^T, F_r^T. T_q, and ASI represent limits within which the LHR algorithms are valid. These limits are obtained directly from the core reload analysis.

Either of the two core power distribution monitoring systems, the Excore Detector Monitoring System or the Incore Detector Monitoring System, provides adequate monitoring of the core power distribution and is capable of verifying that the LCO limits are not exceeded. The Excore Detector Monitoring System performs this function by continuously monitoring ASI with OPERABLE quadrant symmetric excore neutron detectors and by verifying ASI is maintained within the limits specified in the COLR.

In conjunction with the use of the Excore Detector Monitoring System and in establishing the ASI limits, the following assumptions are made:

a. The CEA insertion limits of LCOs 3.1.5 and 3.1.6 are satisfied;

b. The T_q restrictions of LCO 3.2.4 are satisfied; and

c. F_{xy}^T does not exceed the limits of LCO 3.2.2.

The Incore Detector Monitoring System continuously provides a more direct measure of the peaking factors, and the alarms that have been established for the individual incore detector segments ensure that the peak LHRs are maintained within the limits specified in the COLR. The setpoints for these alarms include tolerances, set in conservative directions, for:

a. A measurement calculational uncertainty factor of 1.062;

b. An engineering uncertainty factor of 1.03;

c. An allowance of 1.002 for axial fuel densification and thermal expansion; and

d. A THERMAL POWER measurement uncertainty factor of 1.02 for THERMAL POWER > 50% RTP and 1.035 for THERMAL POWER > 20% RTP and ≤ 50% RTP.

BASES

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation or AOOs (Reference 1, Appendix 1C, Criterion 6). The power distribution and CEA insertion and alignment LCOs preclude core power distributions that violate the following fuel design criteria:

a. During a LOCA, peak cladding temperature must not exceed 2200°F (Reference 2);

b. During a LOFA, 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 DNB condition;

c. During an ejected CEA accident, the energy input to the fuel must not exceed the accepted limits (Reference 1, Section 14.13); 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 (Reference 1, Appendix 1C, Criterion 29).

The power density at any point in the core must be limited to maintain the fuel design criteria (Reference 2). This process is accomplished by maintaining the power distribution and reactor coolant conditions so that the peak LHR and DNB parameters are within operating limits supported by the accident analysis (Reference 1, Chapter 14), with due regard for the correlations between measured quantities, the power distribution, and uncertainties in determining the power distribution.

Fuel cladding failure during a LOCA is limited by restricting the maximum LHGR so that the peak cladding temperature does not exceed 2200°F (Reference 2). High peak cladding temperatures are assumed to cause severe cladding failure by oxidation due to a Zircaloy-water reaction.

The LCOs governing LHR, ASI, and the RCS ensure that these criteria are met as long as the core is operated within the ASI, F_{xy}^T, and F_r^T limits specified in the COLR, and within the T_q limits. The latter are process variables that characterize the three-dimensional power distribution of the reactor core. Operation within the limits for these variables ensures that their actual values are within the range used in the accident analyses.

Fuel cladding damage does not normally occur while the reactor is operating at conditions outside these LCOs during otherwise normal operation. Fuel cladding damage could result, however, if an accident or AOO occurs from initial conditions outside the limits of these LCOs. Changes in the power distribution cause increased power peaking and correspondingly increased local LHRs.

The T_q satisfies 10 CFR 50.36(c)(2)(ii), Criterion 2.

LCO

The power distribution LCO limits are based on correlations between power peaking and the measured variables used as inputs to the LHR and DNB ratio operating limits. The power distribution LCO limits, except T_q, are provided in the COLR. The limits on LHR ensure that in the event of a LOCA, the peak temperature of the fuel cladding does not exceed 2200°F.

APPLICABILITY

In MODE 1 with THERMAL POWER > 50% RTP, T_q must be maintained within the limits assumed in the accident analysis to ensure that fuel damage does not result following an AOO. In other MODEs, this LCO does not apply because THERMAL POWER is not sufficient to require a limit on T_q.

ACTIONS

A.1 and A.2

If the measured T_q is > 0.03 and < 0.10, the calculation of T_q may be nonconservative. T_q must be restored within 4 hours, or F^T_{xy} and F^T_r must be determined to be within the limits of LCOs 3.2.2 and 3.2.3 within 4 hours, and determined to be within these limits every 8 hours thereafter, as long as T_q is out-of-limits. Four hours is sufficient time to allow the operator to reposition CEAs, and significant radial xenon redistribution cannot occur within this time. The 8 hour Completion Time ensures changes in F^T_{xy} and F^T_r can be identified before the limits of LCOs 3.2.2 and 3.2.3, respectively, are exceeded.

B.1

With $T_q > 0.10$, it must be restored to ≤ 0.10 with 2 hours. F_{xy}^T and F_r^T must be verified to be within their specified limits to ensure that acceptable flux peaking factors are maintained. Operation may proceed for a total of 2 hours, after the Condition is entered, while attempts are made to restore T_q to within its limit.

If the tilt is generated due to a CEA misalignment, operating at ≤ 50% RTP allows for the recovery of the CEA. Except as a result of CEA misalignment, $T_q > 0.10$ is not expected; if it occurs, continued operation of the reactor may be necessary to discover the cause of the tilt. If this procedure is followed, operation is restricted to only those conditions required to identify the cause of the tilt. It is necessary to account explicitly for power asymmetries because the radial power peaking factors used in core power distribution calculations are based on an untilted power distribution.

If T_q is not restored to within its limits, the reactor continues to operate with an axial power distribution mismatch. Continued operation in this configuration may induce an axial xenon oscillation that causes increased LHRs when the xenon redistributes. If T_q cannot be restored to within its limits within 2 hours, reactor power must be reduced.

C.1

If Required Actions and associated Completion Times of Condition A or B are not met, THERMAL POWER must be reduced to ≤ 50% RTP. This requirement provides conservative protection from increased peaking due to potential xenon redistribution and provides reasonable assurance that the core is operating within its thermal limits and places the core in a conservative condition. Four hours is a reasonable time to reach 50% RTP in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS	<u>SR 3.2.4.1</u> T_q must be calculated at 12-hour intervals. T_q is determined using the incore and excore detectors. When one excore channel is inoperable and THERMAL POWER is > 75% RTP, the incore detectors shall be used. The 12-hour Frequency prevents significant xenon redistribution between surveillance tests.
REFERENCES	1. UFSAR 2. 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plants"

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.5 AXIAL SHAPE INDEX (ASI)

BASES

BACKGROUND

The purpose of this LCO is to limit the core power distribution to the initial values assumed in the accident analysis. Operation within the limits imposed by this LCO either limits or prevents potential fuel cladding failures that could breach the primary fission product barrier and release fission products to the reactor coolant in the event of a LOCA, LOFA, ejected control element assembly (CEA) accident, or other postulated accident requiring termination by a Reactor Protective System trip function. This LCO limits the amount of damage to the fuel cladding during an accident by ensuring that the plant is operating within acceptable bounding conditions at the onset of a transient.

Methods of controlling the power distribution include:

a. Using CEAs to alter the axial power distribution;

b. Decreasing CEA insertion by boration, thereby improving the radial power distribution; and

c. Correcting off optimum conditions (e.g., a CEA drop or misoperation of the unit) that cause margin degradations.

The core power distribution is controlled so that, in conjunction with other core operating parameters (e.g., CEA insertion and alignment limits), the power distribution does not result in violation of this LCO. The LSSS and this LCO are based on the accident analyses (Reference 1, Chapter 14), so that SAFDLs are not exceeded as a result of AOOs, and the limits of acceptable consequences are not exceeded for other postulated accidents.

Limiting power distribution skewing over time also minimizes the xenon distribution skewing, which is a significant factor in controlling the axial power distribution.

Power distribution is a product of multiple parameters, various combinations of which may produce acceptable power distributions. Operation within the design limits of power distribution is accomplished by generating operating limits on LHR and DNB.

BASES

The limits on LHR, F_{xy}^T, F_r^T, T_q, and ASI represent limits within which the LHR algorithms are valid. These limits are obtained directly from the core reload analysis.

Either of the two core power distribution monitoring systems, the Excore Detector Monitoring System and the Incore Detector Monitoring System, provide adequate monitoring of the core power distribution and are capable of verifying that the LHR does not exceed its limits. The Excore Detector Monitoring System performs this function by continuously monitoring the ASI with the OPERABLE quadrant symmetric excore neutron flux detectors and verifying that the ASI is maintained within the allowable limits specified in the COLR.

In conjunction with the use of the Excore Detector Monitoring System and in establishing the ASI limits, the following conditions are assumed:

a. The CEA insertion limits of LCOs 3.1.5 and 3.1.6 are satisfied;

b. The T_q restrictions of LCO 3.2.4 are satisfied; and

c. F_{xy}^T does not exceed the limits of LCO 3.2.2.

The Incore Detector Monitoring System continuously provides a more direct measure of the peaking factors, and the alarms that have been established for the individual incore detector segments ensure that the peak LHR is maintained within the limits specified in the COLR. The setpoints for these alarms include tolerances, set in conservative directions, as follows:

a. A measurement calculational uncertainty factor of 1.062;

b. An engineering uncertainty factor of 1.03;

c. An allowance of 1.002 for axial fuel densification and thermal expansion; and

d. A THERMAL POWER measurement uncertainty factor of 1.02 for THERMAL POWER > 50% RTP and 1.035 for THERMAL POWER > 20% RTP and ≤ 50% RTP.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation or AOOs (Reference 1, Appendix 1C, Criterion 6). The power distribution and CEA insertion and alignment LCOs prevent core power distributions from reaching levels that violate the following fuel design criteria:

a. During a LOCA, peak cladding temperature must not exceed 2200°F (Reference 2);

b. During a LOFA, 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 DNB condition;

c. During an ejected CEA accident, the energy input to the fuel must not exceed the acceptable limits (Reference 1, Section 14.13); 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 (Reference 1, Appendix 1C, Criterion 29).

The power density at any point in the core must be limited to maintain the fuel design criteria (Reference 2). This limitation is accomplished by maintaining the power distribution and reactor coolant conditions so that the peak LHR and DNB parameters are within operating limits supported by the accident analyses (Reference 1, Chapter 14), with due regard for the correlations among measured quantities, the power distribution, and uncertainties in the determination of power distribution.

Fuel cladding failure during a LOCA is limited by restricting the maximum LHGR so that the peak cladding temperature does not exceed 2200°F (Reference 2). High peak cladding temperatures are assumed to cause severe cladding failure by oxidation due to a Zircaloy-water reaction.

The LCOs governing LHR, ASI, and the RCS ensure that these criteria are met as long as the core is operated within the ASI, F_{xy}^{T}, and F_{r}^{T} limits specified in the COLR, and within the T_q limits. The latter are process variables that characterize the three-dimensional power distribution of the reactor core. Operation within the limits for these variables ensures that their actual values are within the ranges used in the accident analyses.

Fuel cladding damage does not normally occur while the reactor is operating at conditions outside these LCOs during normal operation. Fuel cladding damage results, however, when an accident or AOO occurs from initial conditions outside the limits of these LCOs. This potential for fuel cladding damage exists because changes in the power distribution can cause increased power peaking and correspondingly increased local LHRs.

The ASI satisfies 10 CFR 50.36(c)(2)(ii), Criterion 2.

LCO

The power distribution LCO limits are based on correlations between power peaking and certain measured variables used as inputs to the LHR and DNB ratio operating limits. These power distribution LCO limits, except T_q, are provided in the COLR. The limitation on LHR ensures that in the event of a LOCA, the peak temperature of the fuel cladding does not exceed 2200°F.

The limitation on ASI, along with the limitations of LCO 3.3.1, represents a conservative envelope of operating conditions consistent with the assumptions that have been analytically-demonstrated adequate for maintaining an acceptable minimum DNB ratio throughout all AOOs. Of these, the loss of flow transient is the most limiting. Operation of the core with conditions within the specified limits ensures that an acceptable minimum margin from DNB conditions is maintained in the event of any AOO, including a loss of flow transient.

APPLICABILITY

In MODE 1 with THERMAL POWER > 20% RTP, power distribution must be maintained within the limits assumed in the accident analyses to ensure that fuel damage does not result following an AOO. In other MODEs, this LCO does not apply because THERMAL POWER is not sufficient to require a limit on the core power distribution. Below 20% RTP, the incore detector accuracy is not reliable.

BASES

ACTIONS

A.1

Operating the core within ASI limits specified in the COLR and within the limits of LCO 3.3.1 ensures an acceptable margin for DNB and for maintaining local power density in the event of an AOO. Maintaining ASI within limits also ensures that the limits of Reference 2 are not exceeded during accidents. The Required Actions to restore ASI must be completed within 2 hours to limit the duration the plant is operated outside the initial conditions assumed in the accident analyses. In addition, this Completion Time is sufficiently short that the xenon distribution in the core cannot change significantly.

B.1

If the ASI cannot be restored to within its specified limits, or ASI cannot be determined because of Excore Detector Monitoring System inoperability, core power must be reduced. Reducing THERMAL POWER to ≤ 20% RTP provides reasonable assurance that the core is operating farther from thermal limits and places the core in a conservative condition. Four hours is a reasonable amount of time, based on operating experience, to reduce THERMAL POWER to ≤ 20% RTP in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.5.1

Verifying that the ASI is within the specified limits provides reasonable assurance that the core is not approaching DNB conditions. A Frequency of 12 hours is adequate for the operator to identify trends in conditions that result in an approach to the ASI limits, because the mechanisms that affect the ASI, such as xenon redistribution or CEA drive mechanism malfunctions, cause the ASI to change slowly and should be discovered before the limits are exceeded.

REFERENCES

1. UFSAR
2. 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plants"

B 3.3 INSTRUMENTATION

B 3.3.1 Reactor Protective System (RPS) Instrumentation-Operating

BASES

BACKGROUND

The RPS initiates a reactor trip to protect against violating the core specified acceptable fuel design limits and breaching the reactor coolant pressure boundary during anticipated operational occurrences (AOOs). By tripping the reactor, the RPS also assists the Engineered Safety Features (ESF) systems in mitigating accidents.

The protective systems have been designed to ensure safe operation of the reactor. This is achieved by specifying limiting safety system settings (LSSS) in terms of parameters directly monitored by the RPS, as well as Limiting Conditions for Operation (LCOs) on other reactor system parameters and equipment performance.

The LSSS, defined in this Specification as the Allowable Value, in conjunction with the LCOs, establishes the threshold for protective system action to prevent exceeding acceptable limits during Design Basis Accidents (DBAs).

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

- The departure from nucleate boiling ratio (DNBR) shall be maintained above the Safety Limit (SL) value to prevent departure from nucleate boiling;
- Fuel centerline melting shall not occur; and
- The Reactor Coolant System (RCS) pressure SL of 2750 psia shall not be exceeded.

Maintaining the parameters within the above values ensures that the offsite dose will be within Reference 2, 10 CFR Parts 50 and 100, criteria during AOOs.

Accidents are events that are analyzed even though they are not expected to occur during the plant life. The acceptable limit during accidents is that the offsite dose shall be maintained within the acceptance criteria given in the Reference 1, Chapter 14.

The RPS is segmented into four interconnected modules. These modules are:

- Measurement channels;
- Bistable trip units;
- RPS logic; and
- Reactor trip circuit breakers (RTCBs).

This LCO addresses measurement channels and bistable trip units. It also addresses the automatic bypass removal channel for those trips with operating bypasses. The RPS logic and RTCBs are addressed in LCO 3.3.3.

An instrument channel consists of the measurement channel and bistable trip unit for one channel of one Function.

The role of each of these modules in the RPS, including those associated with the logic and RTCBs, is discussed below.

Measurement Channels

Measurement channels, consisting of field transmitters or process sensors and associated instrumentation, provide a measurable electronic signal based upon the physical characteristics of the parameter being measured.

The Power Range excore nuclear instrumentation drawers, Thermal Margin/Low Pressure (TM/LP) trip calculators, and Axial Power Distribution (APD) trip calculators, are considered components in the measurement channels. The power range nuclear instruments (NIs) provide average power and subchannel deviation signals. The wide range NIs provide a Rate of Change of Power-High trip. Two decades of overlap are provided between the power range NIs and the wide range NIs. Three RPS trip functions use a power level designated as Q power as an input. Q power is the higher of NI power and primary calorimetric power (ΔT power) based on RCS hot leg and cold leg temperatures. Trip functions using Q power as an input include the Power Level-High, TM/LP, and the APD trips.

BASES

The TM/LP and APD trip calculators provide the complex signal processing necessary to calculate the TM/LP trip setpoint, Asymmetric Steam Generator Transient (ASGT) trip setpoint, APD trip setpoint, Power Level-High trip setpoint, and Q power calculation.

The excore NI drawers (wide range and power range) and the TM/LP and APD trip calculators are mounted in the RPS cabinet, with one channel of each in each of the four RPS bays.

Four measurement channels with electrical and physical separation are provided for each parameter used in the direct generation of trip signals. These are designated Channels A through D. Measurement channels provide input to one or more RPS bistables within the same RPS channel. In addition, some measurement channels may also be used as inputs to Engineered Safety Features Actuation System (ESFAS) sensor modules, and most provide indication in the Control Room. Measurement channels used as an input to the RPS are never used for control functions.

When a measurement channel monitoring a parameter exceeds a predetermined setpoint, indicating an unsafe condition, the bistable in the bistable trip unit monitoring the parameter in that measurement channel will trip. Tripping two or more bistable trip units monitoring the same parameter de-energizes matrix logic, which in turn de-energizes the trip path logic. This causes all eight RTCBs to open, interrupting power to the control element assemblies (CEAs), allowing them to fall into the core.

Three of the four instrument channels are necessary to meet the redundancy and testability as described in Reference 1, Appendix 1C. The fourth channel provides additional flexibility by allowing one channel to be removed from service (trip bypass) for maintenance or testing, while still maintaining a minimum two-out-of-three logic. Thus, even with a channel inoperable, no single additional failure in the RPS can either cause an inadvertent trip or prevent a required trip from occurring.

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Since no single failure will either cause or prevent a protective system actuation, and no protective channel feeds a control channel, this arrangement meets the requirements of Reference 1, Section 7.2.2 and Reference 3.

Many of the RPS Function trips are generated by comparing a single measurement to a fixed bistable setpoint. Certain Functions, however, make use of more than one measurement to provide a trip. The following trips use multiple measurement channel inputs:

- Steam Generator Level-Low Trip

  This trip uses the lower of the two steam generator levels as an input to a common bistable.

- Steam Generator Pressure-Low Trip

  This trip uses the lower of the two steam generator pressures as an input to a common bistable.

- Power Level-High Trip

  The Power Level-High trip uses Q power as its only input. Q power is the higher of NI power and ΔT power. Q power has a trip setpoint that tracks power levels downward so that the trip setpoint is always within a fixed increment above current power, subject to a minimum value.

  On power increases, the trip setpoint remains fixed unless manually reset, at which point the trip setpoint increases to the new setpoint, which is a fixed increment above Q power at the time of reset, and the trip setpoint is subject to a maximum value. Thus, during power escalation, the trip setpoint must be repeatedly reset to avoid a reactor trip.

- TM/LP and ASGT Trip

  Q power is only one of several inputs to the TM/LP trip. Other inputs include internal AXIAL SHAPE INDEX (ASI) and cold leg temperature based on the higher of two cold leg resistance temperature detectors. The TM/LP trip setpoint is a complex function of these inputs and represents a minimum acceptable RCS pressure to be compared to actual RCS pressure in the TM/LP trip unit.

Steam generator pressure is also an indirect input to the TM/LP trip via the ASGT. This Function provides a reactor trip when the secondary pressure in either steam generator exceeds that of the other generator by greater than a fixed amount. The trip is implemented by biasing the TM/LP trip setpoint upward so as to ensure TM/LP trip if an ASGT is detected.

- <u>APD-High Trip</u>

  Q power and subchannel deviation are inputs to the APD trip. The APD trip setpoint is a function of Q power, being more restrictive at higher power levels. It provides a reactor trip if actual ASI exceeds the APD trip setpoint.

<u>Bistable Trip Units</u>

Bistable trip units, mounted in the RPS cabinet, receive an analog input from the measurement channels, compare the analog input to trip setpoints, and provide contact output to the matrix logic. They also provide local trip indication and remote annunciation.

There are four channels of bistable trip units, designated A through D, for each RPS Function, one for each measurement channel. Bistable output relays de-energize when a trip occurs.

The contacts from these bistable relays are arranged into six coincidence matrices, comprising the matrix logic. If bistables monitoring the same parameter in at least two bistable trip unit channels trip, the matrix logic will generate a reactor trip (two-out-of-four logic).

Some of the RPS measurement channels provide contact outputs to the RPS, so the comparison of an analog input to a trip setpoint is not necessary. In these cases, the bistable trip unit is replaced with an auxiliary trip unit. The auxiliary trip units provide contact multiplication so the single input contact opening can provide multiple contact outputs to the matrix logic, as well as trip indication and annunciation.

Trip Functions employing auxiliary trip units include the Loss of Load trip and the APD trip.

The APD trip, described above, is a complex function in which the actual trip comparison is performed within the APD calculator. Therefore the APD trip unit employs a contact input from the APD calculator.

All RPS trips, with the exception of the Loss of Load trip, generate a pretrip alarm as the trip setpoint is approached.

The trip setpoints used in the bistable trip units are based on the analytical limits stated in Reference 1, Chapter 14, except for the APD and Loss of Load Functions, which are not credited in safety analyses. The selection of these trip setpoints is such that adequate protection is provided when all sensor and processing time delays are taken into account in the respective analytical limits. To allow for calibration tolerances, instrumentation uncertainties, instrument channel drift, and severe environment errors (for those RPS channels that must function in harsh environments, as defined by Reference 2, 10 CFR 50.49) RPS trip setpoints are conservatively adjusted with respect to the analytical limits. A detailed description of the methodology used to calculate the trip setpoints, including their explicit uncertainties, is provided in Reference 4. The nominal trip setpoint entered into the bistable is more conservative than that specified by the Allowable Value. A channel is inoperable if its actual setpoint is not within its required Allowable Value.

Setpoints in accordance with the Allowable Value will ensure that SLs of Chapter 2.0 are not violated during AOOs and the consequences of DBAs will be acceptable, providing the plant is operated from within the LCOs at the onset of the AOO or DBA and the equipment functions as designed.

Note that in the accompanying LCO 3.3.1, the Allowable Values of Table 3.3.1-1 are the LSSS.

RPS Logic

The RPS logic, addressed in LCO 3.3.3, consists of both matrix and trip path logic and employs a scheme that provides a reactor trip when bistables in any two of the four channels sense the same input parameter trip signal. This is called a two-out-of-four trip logic. This logic and the RTCB configuration are shown in Figure B 3.3.1-1.

Bistable relay contact outputs from the four bistable trip unit channels are configured into six logic matrices. Each logic matrix checks for a coincident trip in the same parameter in two bistable trip unit channels. The matrices are designated the AB, AC, AD, BC, BD, and CD matrices to reflect the bistable trip unit channels being monitored. Each logic matrix contains four normally energized matrix relays. When a coincidence is detected, consisting of a trip in the same Function in the two channels being monitored by the logic matrix, all four matrix relays de-energize.

The logic matrix relay contacts are arranged into trip paths, with one of the four matrix relays in each matrix opening contacts in one of the four trip paths. Each trip path provides power to one of the four normally energized RTCB control relays (K1, K2, K3, and K4). Thus, the trip paths each have six contacts in series, one from each matrix, performing a logical OR function by opening the RTCBs if any one or more of the six logic matrices indicate a coincidence condition.

Each trip path is responsible for opening one set of two of the eight RTCBs. When de-energized, the RTCB control relays (K-relays) interrupt power to the breaker undervoltage trip coils and simultaneously apply power to the shunt trip coils on each of the two breakers. Actuation of either the undervoltage or shunt trip coil is sufficient to open the RTCB and interrupt power from the motor generator (MG) sets to the control element drive mechanisms (CEDMs).

When a coincidence occurs in two RPS instrument channels from one Function, all four matrix relays in the affected matrix de-energize. This, in turn, de-energizes all four RTCB control relays, which simultaneously de-energize the undervoltage and energize the shunt trip coils in all eight RTCBs, tripping them open.

Matrix logic refers to the matrix power supplies, trip channel bypass contacts, and interconnecting matrix wiring between bistable and auxiliary trip units, up to but not including the matrix relays. Contacts in the bistable and auxiliary trip units are excluded from the matrix logic definition, since they are addressed separately.

The trip path logic consists of the trip path power source, matrix relays and their associated contacts, and all interconnecting wiring through the K-relay contacts in the RTCB control circuitry.

It is possible to change the two-out-of-four RPS logic to a two-out-of-three logic for a given input parameter, in one channel at a time, by trip bypassing select portions of the matrix logic. Trip bypassing a bistable trip unit effectively shorts the bistable relay contacts in the three matrices associated with that instrument channel. Thus, the bistables will function normally, producing normal trip indication and annunciation, but a reactor trip will not occur unless two additional instrument channels indicate a trip condition. Trip bypassing can be simultaneously performed on any number of parameters in any number of Functions, providing each parameter is bypassed in only one instrument channel per function at a time. Administrative controls prevent simultaneous trip bypassing of the same parameter in more than one instrument channel. Trip bypassing is normally employed during maintenance or testing.

In addition to the trip bypasses, there are also operating bypasses on select RPS trips. Some of these operating bypasses are enabled manually, others automatically, in all four RPS instrument channels for a Function when plant conditions do not warrant the specific trip function protection. All operating bypasses are automatically removed when enabling bypass conditions are no longer satisfied. Trip Functions with operating bypasses include

Rate of Change of Power-High, Reactor Coolant Flow-Low, Steam Generator Pressure-Low, APD-High, TM/LP, and Steam Generator Pressure Difference trips. The Loss-of-Load, Rate of Change of Power-High, and APD-High trips' operating bypasses are automatically enabled and disabled.

RTCBs

The reactor trip switchgear, addressed in LCO 3.3.3 and shown in Figure B 3.3.1-1, consists of eight RTCBs, which are operated in four sets of two breakers (four RTCB channels, including shunt trip coils and undervoltage coils). Power input to the reactor trip switchgear comes from two full capacity MG sets operated in parallel such that the loss of either MG set does not de-energize the CEDMs. There are two separate CEDM power supply buses, each bus powering half of the CEDMs. Power is supplied from the MG sets to each bus via two redundant trip paths. This ensures that a fault or the opening of a breaker in one trip path (i.e., for testing purposes) will not interrupt power to the CEDM buses.

Each of the four trip paths consists of two RTCBs in series. The two RTCBs within a trip path are actuated by separate trip paths.

The eight RTCBs are operated as four sets of two breakers (four RTCB channels, including shunt trip coils and undervoltage coils). Each set of two RTCB's is opened by the same K-relay. This arrangement ensures that power is interrupted to both CEDM buses, thus preventing trip of only half of the CEAs (a half trip). Any one inoperable RTCB in a RTCB channel (set of two breakers) will make the entire RTCB channel inoperable.

Each set of RTCBs is operated by either a manual trip push button or an RPS actuated K-relay. There are four manual trip push buttons, arranged in two sets of two, as shown in Figure B 3.3.1-1. Depressing both push buttons in either set will result in a reactor trip.

When a manual trip is initiated using the control room push buttons, the RPS trip paths and K-relays are bypassed, and the RTCB undervoltage and shunt trip coils are actuated independent of the RPS.

A manual trip channel includes the push button and interconnecting wiring to both RTCBs necessary to actuate both the undervoltage and shunt trip coils but excludes the K-relay contacts and their interconnecting wiring to the RTCBs, which are considered part of the trip path logic.

Functional testing of the RPS instrument and logic channels, from bistable input through the opening of individual sets of RTCBs, can be performed either at power or shutdown and is normally performed on a quarterly basis. Reference 1, Section 7.2 explains RPS testing in more detail.

APPLICABLE SAFETY ANALYSES

Most of the analyzed accidents and transients can be detected by one or more RPS Functions. The accident analysis contained in Reference 1, Chapter 14 takes credit for most RPS trip Functions. Some Functions not specifically credited in the accident analysis are part of the Nuclear Regulatory Commission (NRC)-approved licensing basis for the plant. These Functions may provide protection for conditions that do not require dynamic transient analysis to demonstrate Function performance. Other Functions, such as the Loss of Load trip, are purely equipment protective, and their use minimizes the potential for equipment damage.

The specific safety analyses applicable to each protective Function are identified below:

1. Power Level-High Trip

   The Power Level-High trip provides reactor core protection against positive reactivity excursions that are too rapid for a Pressurizer Pressure-High or TM/LP trip to protect against. The following events require Power Level-High trip protection:

   - Uncontrolled CEA withdrawal event;
   - Excess load; and
   - CEA ejection event.

The first two events are AOOs, and fuel integrity is maintained. The third is an accident, and limited fuel damage may occur.

2. Rate of Change of Power-High Trip

The Rate of Change of Power-High trip is used to trip the reactor when excore logarithmic power, measured by the wide range logarithmic neutron flux monitors, indicates an excessive rate of change. The Rate of Change of Power-High trip Function minimizes transients for events such as a born dilution event, continuous CEA withdrawal, or CEA ejection from subcritical conditions. Because of this Function, such events are assured of having much less severe consequences than events initiated from critical conditions. The trip is automatically bypassed when NUCLEAR INSTRUMENT POWER is < 1E-4% RTP, when poor counting statistics may lead to erroneous indication. It is also bypassed at > 12% RTP, where other RPS trips provide protection from these events. With the RTCBs open, the Rate of Change of Power-High trip is not required to be OPERABLE; however, at least two wide range logarithmic neutron flux monitor channels are required by LCO 3.3.12 to be OPERABLE. Limiting Condition for Operation 3.3.12 ensures the wide range logarithmic neutron flux monitor channels are available to detect and alert the operator to a boron dilution event.

3. Reactor Coolant Flow-Low Trip

The Reactor Coolant Flow-Low trip provides protection during the following events:

- Loss of RCS flow;
- Loss of non-emergency AC power; and
- Reactor coolant pump (RCP) seized rotor.

The loss of RCS flow and of non-emergency AC power events are AOOs where fuel integrity is maintained. The RCP seized rotor is an accident where fuel damage may result.

4. Pressurizer Pressure-High Trip

The Pressurizer Pressure-High trip, in conjunction with pressurizer safety valves and main steam safety valves, provides protection against overpressure conditions in the RCS during the following events:

- Loss of Load; and
- Feedwater Line Break (FWLB).

5. Containment Pressure-High Trip

The Containment Pressure-High trip prevents exceeding the containment design pressure during certain loss of coolant accidents (LOCAs) or FWLB accidents. It ensures a reactor trip prior to, or concurrent with, a LOCA, thus assisting the ESFAS in the event of a LOCA or Main Steam Line Break (MSLB). Since these are accidents, SLs may be violated. However, the consequences of the accident will be acceptable.

6. Steam Generator Pressure-Low Trip

The Steam Generator Pressure-Low trip provides protection against an excessive rate of heat extraction from the steam generators, which would result in a rapid uncontrolled cooldown of the RCS. This trip is needed to shut down the reactor and assist the ESFAS in the event of an MSLB. Since these are accidents, SLs may be violated. However, the consequences of the accident will be acceptable.

7. Steam Generator 1 and 2 Level-Low Trip

The Steam Generator 1 Level-Low and Steam Generator 2 Level-Low trips are required for the loss of normal feedwater and ASGT events.

The Steam Generator Level-Low trip ensures that low DNBR, high local power density, and the RCS pressure SLs are maintained during normal operation and AOOs, and, in conjunction with the ESFAS, the consequences of the Feedwater System pipe break accident will be acceptable.

8. <u>APD-High Trip</u>

The APD-High trip ensures that excessive axial peaking, such as that due to axial xenon oscillations, will not cause fuel damage. It ensures that neither a DNBR less than the SL, nor a peak linear heat rate that corresponds to the temperature for fuel centerline melting, will occur. This trip is the primary protection against fuel centerline melting.

9. <u>Thermal Margin</u>

a. <u>TM/LP Trip</u>

The TM/LP trip prevents exceeding the DNBR SL during AOOs and aids the ESFAS during certain accidents. The following events require TM/LP trip protection:

- RCS depressurization (inadvertent safety or power-operated relief valves opening);
- Steam generator tube rupture; and
- LOCA accident.

The first event is an AOOs, and fuel integrity is maintained. The second and third events are accidents, and limited fuel damage may occur, although only the LOCA is expected to result in fuel damage. The trip is initiated whenever the RCS pressure signal drops below a minimum value (P_{min}) or a computed value (P_{var}) as described below, whichever is higher. The setpoint is a Function of Q power, ASI, reactor inlet (cold leg) temperature, and the number of RCPs operating.

The minimum value of reactor coolant flow rate, the maximum AZIMUTHAL POWER TILT (T_q), and the maximum CEA deviation permitted for continuous operation are assumed in the generation of this trip signal. In addition, CEA group sequencing in accordance with LCO 3.1.7 is assumed. Finally, the maximum insertion of CEA banks that can occur during any AOO prior to a Power Level-High trip is assumed.

b. ASGT

The ASGT provides protection for those AOOs associated with secondary system malfunctions that result in asymmetric primary coolant temperatures. The most limiting event is closure of a single main steam isolation valve (MSIV). Asymmetric Steam Generator Transient is provided by comparing the secondary pressure in both steam generators in the TM/LP trip calculator. If the pressure in either exceeds that in the other by the trip setpoint, a TM/LP trip will result.

10. Loss of Load

The Loss of Load trip causes a trip when operating above 15% of RTP. This trip provides turbine protection, reduces the severity of the ensuing transient, and helps avoid the lifting of the main steam safety valves during the ensuing transient, thus extending the service life of these valves. No credit was taken in the accident analyses for operation of this trip. Its functional capability is required to enhance overall plant equipment service life and reliability.

Operating Bypasses

The operating bypasses are addressed in footnotes to Table 3.3.1-1. They are not otherwise addressed as specific table entries.

The automatic bypass removal features must function as a backup to manual actions for all trips credited in safety analyses to ensure the trip Functions are not operationally bypassed when the safety analysis assumes the Functions are not bypassed. The RPS operating bypasses are:

Zero power mode bypass (ZPMB) removal of the TM/LP, ASGT, and reactor coolant low flow trips when NUCLEAR INSTRUMENT POWER is < 1E-4% RTP. This bypass is manually enabled below the specified setpoint to permit low power testing. The wide range NI Level 1 bistable in the wide range drawer provides a signal to auxiliary logic, which then permits manual bypassing below the setpoint and removes the bypass above the setpoint.

Power rate of change bypass removal — The Rate of Change of Power-High trip is automatically bypassed at < 1E-4% RTP, as sensed by the wide range NI Level 1 bistable, and at > 12% RTP by the linear range NI Level 1 bistable, mounted in their respective NI drawers (Reference 5). Automatic bypass removal is also effected by these bistables when conditions are no longer satisfied.

Loss of Load and APD-High trip bypass removal — The Loss of Load and APD-High trips are automatically bypassed when at < 15% RTP as sensed by the linear range NI Level 1 bistable. The bypass is automatically removed by this bistable above the setpoint. This same bistable is used to bypass the Rate of Change of Power-High trip.

Steam Generator Pressure-Low trip bypass removal. The Steam Generator Pressure-Low trip is manually enabled below the pretrip setpoint. The permissive signal is removed, and the bypass automatically removed, when the Steam Generator Pressure-Low trip is above the pretrip setpoint.

The RPS instrumentation satisfies 10 CFR 50.36(c)(2)(ii), Criterion 3.

LCO

The LCO requires all instrumentation performing an RPS Function to be OPERABLE. Failure of any required portion of the instrument channel renders the affected channel(s) inoperable and reduces the reliability of the affected Functions. The specific criteria for determining channel OPERABILITY differ slightly between Functions. These criteria are discussed on a Function-by-Function basis below.

Actions allow trip channel bypass of individual instrument channels, but administrative controls prevent operation with a second channel in the same Function bypassed. Plants are restricted to 48 hours in a trip bypass condition before either restoring the Function to four channel operation (two-out-of-four logic) or placing the channel in trip (one-out-of-three logic).

Only the Allowable Values are specified for each RPS trip Function in the LCO. Nominal trip setpoints are established for the Functions via the plant-specific procedures. The nominal setpoints are selected to ensure the plant parameters do not exceed the Allowable Value if the bistable trip unit is performing as required. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable, provided that operation and testing are consistent with the assumptions of the plant--specific setpoint calculations. Each nominal trip setpoint is more conservative than the analytical limit assumed in the safety analysis in order to account for instrument channel uncertainties appropriate to the trip Function. These uncertainties are defined in Reference 4. The nominal trip setpoint entered into a bistable is more conservative than that specified by the Allowable Value. A channel is inoperable if its actual setpoint is not within its required Allowable Value.

The following Bases for each trip Function identify the above RPS trip Function criteria items that are applicable to establish the trip Function OPERABILITY.

1. Power Level-High Trip

   This LCO requires all four instrument channels of the Power Level-High trip to be OPERABLE in MODEs 1 and 2.

   The Allowable Value is high enough to provide an operating envelope that prevents unnecessary Power Level-High trips during normal plant operations. The Allowable Value is low enough for the system to maintain a margin to unacceptable fuel cladding damage should a CEA ejection accident occur.

   The Power Level-High trip setpoint is operator adjustable and can be set at a fixed increment above the indicated THERMAL POWER level. Operator action is required to increase the trip setpoint as THERMAL POWER is increased. The trip setpoint is automatically decreased as THERMAL POWER decreases.

The trip setpoint has a maximum and a minimum setpoint.

Adding to this maximum value the possible variation in trip setpoint due to calibration and instrument errors, the maximum actual steady state THERMAL POWER level at which a trip would be actuated is 109% RTP, which is the value used in the safety analyses.

To account for these errors, the safety analysis minimum value is 40% RTP. The 10% step increase in trip setpoint is a maximum value assumed in the safety analysis. There is no uncertainty applied to the step in the safety analyses.

2. Rate of Change of Power-High Trip

This LCO requires four instrument channels of Rate of Change of Power-High trip to be OPERABLE in MODEs 1 and 2.

The high power rate of change trip serves as a backup to the administratively-enforced startup rate limit. The Function is not credited in the accident analyses; therefore, the Allowable Value for the trip is not derived from analytical limits.

3. Reactor Coolant Flow-Low Trip

This LCO requires four instrument channels of Reactor Coolant Flow-Low trip to be OPERABLE in MODEs 1 and 2.

The trip may be manually bypassed when NUCLEAR INSTRUMENT POWER falls below 1E-4% RTP. This operating bypass is part of the ZPMB circuitry, which also bypasses the TM/LP trip and provides a ΔT power block signal to the Q power select logic. The ZPMB allows low power physics testing at reduced RCS temperatures and pressures. It also allows heatup and cooldown with shutdown CEAs withdrawn.

This trip is set high enough to maintain fuel integrity during a loss of flow condition. The setting is low enough to allow for normal operating fluctuations from offsite power. To account for analysis uncertainty, the value in the safety analysis is 93% of design flow. Reactor Coolant System flow is maintained above design flow by LCO 3.4.1.

4. Pressurizer Pressure-High Trip

This LCO requires four instrument channels of Pressurizer Pressure-High trip to be OPERABLE in MODEs 1 and 2.

The Allowable Value is set high enough to allow for pressure increases in the RCS during normal operation (i.e., plant transients) not indicative of an abnormal condition. The setting is below the lift setpoint of the pressurizer safety valves and low enough to initiate a reactor trip when an abnormal condition is indicated. The analysis setpoint includes allowance for harsh environment, where appropriate.

The Pressurizer Pressure-High trip concurrent with power-operated relief valve operation avoids unnecessary operation of the pressurizer safety valves (Reference 5).

5. Containment Pressure-High Trip

This LCO requires four instrument channels of Containment Pressure-High trip to be OPERABLE in MODEs 1 and 2.

The Allowable Value is high enough to allow for small pressure increases in Containment, expected during normal operation (i.e., plant heatup) that are not indicative of an abnormal condition. The setting is low enough to initiate a reactor trip to prevent containment pressure from exceeding design pressure following a DBA.

6. Steam Generator Pressure-Low Trip

This LCO requires four instrument channels of Steam Generator Pressure-Low trip per steam generator to be OPERABLE in MODEs 1 and 2.

The Allowable Value is sufficiently below the full load operating value for steam pressure so as not to interfere with normal plant operation, but still high enough to provide the required protection in the event of excessive steam demand. Since excessive steam demand causes the RCS to cool down, resulting in positive reactivity addition to the core in the presence of a negative moderator temperature coefficient, a reactor trip is required to offset that effect.

The analysis setpoint value includes harsh environment uncertainties, where appropriate.

The Function may be manually bypassed as steam generator pressure is reduced during controlled plant shutdowns. This operating bypass is permitted at a preset steam generator pressure. The bypass, in conjunction with the ZPMB, allows testing at low temperatures and pressures, and heatup and cooldown with the shutdown CEAs withdrawn. From a bypass condition, the trip will be automatically reinstated as steam generator pressure increases above the preset pressure.

7. <u>Steam Generator Level-Low Trip</u>

This LCO requires four instrument channels of Steam Generator Level - Low per steam generator to be OPERABLE in MODEs 1 and 2.

The Allowable Value is sufficiently below the normal operating level for the steam generators so as not to cause a reactor trip during normal plant operations. The trip setpoint is high enough to ensure a reactor trip signal is generated to prevent operation with the steam generator water level below the minimum volume required for adequate heat removal capacity, and ensures that the pressure of the RCS will not exceed its SL. The specified setpoint, in combination with the Auxiliary Feedwater Actuation System (AFAS), ensures that sufficient water inventory exists in both

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steam generators to remove decay heat following a Loss of Main Feedwater Flow event.

8. APD-High Trip

This LCO requires four instrument channels of APD-High trip to be OPERABLE in MODE 1, NUCLEAR INSTRUMENT POWER ≥ 15% RTP.

The Allowable Value curve was derived from an analysis of many axial power shapes with allowances for instrumentation inaccuracies and the uncertainty associated with the excore to incore ASI relationship.

The APD-High trip is automatically bypassed at < 15% RTP, as measured by the NIs, where it is not required for reactor protection (Reference 5).

9. Thermal Margin

a. TM/LP Trip

This LCO requires four instrument channels of TM/LP trip to be OPERABLE in MODEs 1 and 2.

The Allowable Value includes allowances for equipment response time, measurement uncertainties, processing error, and a further allowance to compensate for the time delay associated with providing effective termination of the occurrence that exhibits the most rapid decrease in margin to the SLs.

This trip may be manually bypassed when NUCLEAR INSTRUMENT POWER falls below 1E-4% RTP. This operating bypass is part of the ZPMB circuitry, which also bypasses the Reactor Coolant Flow-Low trip and provides a ΔT power block signal to the Q power select logic (Reference 5). The ZPMB allows low power physics testing at reduced RCS temperatures and pressures. It also allows heatup and cooldown with shutdown CEAs withdrawn.

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b. ASGT

This LCO requires four instrument channels of ASGT to be OPERABLE in MODEs 1 and 2.

The Allowable Value is high enough to avoid trips caused by normal operation and minor transients, but ensures DNBR protection in the event of DBAs. The difference between the Allowable Value and the analysis setpoint allows for instrument uncertainty.

The trip may be manually bypassed when NUCLEAR INSTRUMENT POWER falls below 1E-4% RTP as part of the ZPMB circuitry operating bypass. The Steam Generator Pressure Difference is subject to the ZPMB, since it is an input to the TM/LP trip and is not required for protection at low power levels (Reference 5).

10. Loss of Load

The LCO requires four Loss of Load instrument channels to be OPERABLE in MODE 1, NUCLEAR INSTRUMENT POWER ≥ 15% RTP.

The Loss of Load trip is automatically bypassed when NUCLEAR INSTRUMENT POWER falls below 15%, as measured by NIs, to allow loading the turbine.

Bypasses

The LCO on automatic bypass removal features requires that the automatic bypass removal feature of all four operating bypass channels be OPERABLE for each RPS Function with an operating bypass in the MODEs addressed in the specific LCO for each Function. All four automatic bypass removal features must be OPERABLE to ensure that none of the four RPS instrument channels are inadvertently bypassed.

The LCO applies to the automatic bypass removal feature only. If the bypass channel is failed so as to prevent entering a bypass condition, operation may continue.

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APPLICABILITY

This LCO is applicable in accordance with Table 3.3.3-1. Most RPS trip functions are required to be OPERABLE in MODEs 1 and 2 because the reactor is critical in these MODEs. The trips are designed to take the reactor subcritical, maintaining the SLs during AOOs and assisting the ESFAS in providing acceptable consequences during accidents. Exceptions are addressed in footnotes to the table. Exceptions to this APPLICABILITY are:

- The APD-High and Loss-of-Load trips are only applicable in MODE 1, NUCLEAR INSTRUMENT POWER $\geq$ 15% RTP because they are automatically bypassed at < 15% RTP, as measured by NIs, where they are no longer needed.
- The Rate of Change of Power-High trip, RPS logic, RTCBs, and manual trip are also required in MODEs 3, 4, and 5, with the RTCBs closed, to provide protection for boron dilution and CEA withdrawal events. The Rate of Change of Power-High trip in these lower MODEs is addressed in LCO 3.3.2. The RPS logic in MODEs 1, 2, 3, 4, and 5 is addressed in LCO 3.3.3.

Most trip functions are not required to be OPERABLE in MODEs 3, 4, and 5. In MODEs 3, 4, and 5, the emphasis is placed on return to power events. The reactor is protected in these MODEs by ensuring adequate SHUTDOWN MARGIN (SDM).

ACTIONS

The most common causes of instrument channel inoperability are outright failure or drift of the bistable trip unit or measurement channel sufficient to exceed the tolerance allowed by Reference 4. Typically, the drift is found to be small which, at worst, results in a delay of actuation rather than a total loss of Function. This determination is generally made during the performance of a CHANNEL CALIBRATION when the process instrument is set up for adjustment to bring it to within specification. Sensor Drift could also be identified during the CHANNEL CHECKS. CHANNEL FUNCTIONAL TESTs identify bistable trip unit drift. If the trip setpoint is less conservative than the Allowable Value in Table 3.3.1-1, the instrument channel is declared inoperable immediately, and the appropriate Condition(s) must be entered immediately.

In the event that either an instrument channel's trip setpoint is found nonconservative with respect to the Allowable Value or the transmitter, instrument loop, signal processing electronics, RPS bistable trip unit, or applicable automatic bypass removal feature when bypass is in effect, is found inoperable, then all affected Functions provided by that channel must be declared inoperable, and the plant must enter the Condition for the particular protection Function affected.

When the number of inoperable instrument channels in a trip Function exceeds that specified in any related Condition associated with the same trip Function, the plant is outside the safety analysis. Therefore, LCO 3.0.3 is immediately entered, if applicable, in the current MODE of operation.

A Note has been added to the ACTIONS to clarify the application of the Completion Time rules. The Conditions of this Specification may be entered independently for each Function. The Completion Times of each inoperable Function will be tracked separately for each Function, starting from the time the Condition was entered.

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

Condition A applies to the failure of a single instrument channel in any RPS automatic trip Function. Reactor Protective System coincidence logic is normally two-out-of-four.

If one RPS bistable trip unit or associated measurement channel is inoperable, startup or power operation is allowed to continue, providing the inoperable bistable trip unit is placed in bypass or trip within 1 hour (Required Action A.1).

The Completion Time of 1 hour allotted to restore, bypass, or trip the instrument channel is sufficient to allow the operator to take all appropriate actions for the failed channel, while ensuring that the risk involved in operating with the failed channel is acceptable.

BASES

The failed instrument channel is restored to OPERABLE status or is placed in trip within 48 hours (Required Action A.2.1 or Required Action A.2.2). Required Action A.2.1 restores the full capability of the Function.

Required Action A.2.2 places the Function in a one-out-of-three configuration. In this configuration, common cause failure of dependent channels cannot prevent a trip.

The Completion Time of 48 hours is based on operating experience, which has demonstrated that a random failure of a second instrument channel occurring during the 48-hour period is a low probability event.

B.1 and B.2

Condition B applies to the failure of two instrument channels in any RPS automatic trip Function.

The Required Action is modified by a Note stating that LCO 3.0.4 is not applicable. The Note was added to allow the changing of MODEs, even though two instrument channels are inoperable, with one channel bypassed and one tripped. MODE changes in this configuration are allowed to permit maintenance and testing on one of the inoperable channels. In this configuration, the protective system is in a one-out-of-two logic, and the probability of a common cause failure affecting both of the OPERABLE channels during the 48 hours permitted is remote.

Required Action B.1 provides for placing one inoperable channel in bypass and the other channel in trip within the Completion Time of 1 hour. This Completion Time is sufficient to allow the operator to take all appropriate actions for the failed channels, while ensuring that the risk involved in operating with the failed channels is acceptable. With one channel of protective instrumentation bypassed, the RPS Function is in a two-out-of-three logic; but with another channel failed, the RPS Function may be operating in a two-out-of-two logic. This is outside the assumptions made in the analyses and should be corrected. To correct the problem, the second channel is placed in trip. This places the RPS Function in a one-out-of-two logic. If any of the other OPERABLE channels receives a trip signal, the reactor will trip.

One instrument channel should be restored to OPERABLE status within 48 hours for reasons similar to those stated under Condition A. After one channel is restored to OPERABLE status, the provisions of Condition A still apply to the remaining inoperable channel. Therefore, the channel that is still inoperable after completion of Required Action B.2 must be placed in trip if more than 48 hours have elapsed since the initial channel failure.

C.1 and C.2

The excore detectors are used to generate the internal ASI used as an input to the TM/LP and APD-High trips. Incore detectors provide a more accurate measurement of ASI. If one or more excore channels cannot be calibrated to match incore detectors, power is restricted or reduced during subsequent operations because of increased uncertainty associated with using uncalibrated excore channels.

The Completion Time of 24 hours is adequate to perform the Surveillance Requirement (SR) while minimizing the risk of operating in an unsafe condition.

D.1, D.2.1, D.2.2.1, and D.2.2.2

Condition D applies to one automatic bypass removal feature inoperable. If the automatic bypass removal feature for any operating bypass channel cannot be restored to OPERABLE status, the associated RPS channel may be considered OPERABLE only if the bypass is not in effect. Otherwise, the affected RPS channel must be declared inoperable, as in Condition A, and the bypass either removed or the automatic bypass removal feature repaired. The Bases for Required Actions and Completion Times are the same as discussed for Condition A.

E.1, E.2.1, and E.2.2

Condition E applies to two inoperable automatic bypass removal features. If the automatic bypass removal features cannot be restored to OPERABLE status, the associated RPS channel may be considered OPERABLE only if the bypasses are not in effect. Otherwise, the affected RPS channels must be declared inoperable, as in Condition B, and the bypasses either removed or the automatic bypass removal features repaired. Also, Required Action E.2.2 provides for the restoration of the one affected RPS channel to OPERABLE status within the rules of Completion Time specified under Condition B. Completion Times are consistent with Condition B.

The Required Action is modified by a Note stating that LCO 3.0.4 is not applicable. The Note was added to allow the changing of MODEs, even though two automatic bypass removal features are inoperable, with one bistable trip unit bypassed and one tripped. MODE changes in this configuration are allowed to permit maintenance and testing on one of the inoperable automatic bypass removal features. In this configuration, the Function is in a one-out-of-two logic, and the probability of a common cause failure affecting both of the OPERABLE automatic bypass removal features during the 48 hours permitted is remote.

F.1

Condition F is entered when the Required Action and associated Completion Time of Condition A, B, C, D, or E are not met for the APD-High trip and Loss-of-Load trip Functions.

If the Required Actions associated with these Conditions cannot be completed within the required Completion Times, the reactor must be brought to a MODE in which the Required Actions do not apply. The allowed Completion Time of 6 hours to reduce THERMAL POWER to < 15% RTP is reasonable, based on operating experience, to decrease power to < 15% RTP from full power conditions in an orderly manner and without challenging plant systems.

G.1

Condition G is entered when the Required Action and associated Completion Time of Condition A, B, C, D, or E are not met except for the APD-High trip and Loss-of-Load trip Functions.

BASES

If the Required Actions associated with these Conditions cannot be completed within the required Completion Times, the reactor must be brought to a MODE in which the Required Actions do not apply. The allowed Completion Time of 6 hours to be in MODE 3 is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

The SRs for any particular RPS Function are found in the SR column of Table 3.3.1-1 for that Function. Most Functions are subject to CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION.

SR 3.3.1.1

Performance of the CHANNEL CHECK once every 12 hours ensures that gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one instrument 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 channel 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.

Agreement criteria are determined by the plant staff based on a qualitative assessment of the instrument channel combined with the instrument channel uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the transmitter or the signal processing equipment has drifted outside its limits. CHANNEL CHECKS are performed on the wide range logarithmic neutron flux monitor for the Rate of Change of Power-High trip Function.

The Frequency, about once every shift, is based on operating experience that demonstrates the rarity of instrument channel failure. Since the probability of two random failures in redundant channels in any 12-hour period is extremely low, the CHANNEL CHECK minimizes the chance of loss of RPS Function due to failure of redundant channels. The CHANNEL CHECK supplements less formal, but more frequent, checks of the channel during normal operational use of the displays.

SR 3.3.1.2

A daily calibration (heat balance) is performed when THERMAL POWER is ≥ 15%. The daily calibration shall consist of adjusting the "nuclear power calibrate" potentiometers to agree with the calorimetric calculation if the absolute difference is > 1.5%. The "ΔT power calibrate" potentiometers are then used to null the "nuclear power-ΔT power" indicators on the RPS Calibration and Indication Panel. Performance of the daily calibration ensures that the two inputs to the Q power measurement are indicating accurately with respect to the much more accurate secondary calorimetric calculation. The heat balance addresses overall gain of the instruments and does not include ASI.

The Frequency of 24 hours is based on plant operating experience and takes into account indications and alarms located in the Control Room to detect deviations in channel outputs. The Frequency is modified by a Note indicating this Surveillance must be performed until 12 hours after THERMAL POWER is ≥ 15% RTP. The secondary calorimetric is inaccurate at lower power levels. The 12 hours allows time for plant stabilization, data taking, and instrument calibration.

A second Note indicates the daily calibration may be suspended during PHYSICS TESTS. This ensures that calibration is proper both preceding and following physics testing at each plateau, recognizing that during testing, changes in power distribution and RCS temperature may render the calibration inaccurate.

SR 3.3.1.3

It is necessary to calibrate the excore power range channel upper and lower subchannel amplifiers such that the internal

ASI used in the TM/LP trip and APD-High trip Functions reflects the true core power distribution as determined by the incore detectors. The SR is not required to be performed until 12 hours after THERMAL POWER is ≥ 20% RTP and required to be performed prior to operation above 90% RTP. Uncertainties in the excore and incore measurement process make it impractical to calibrate when THERMAL POWER is < 20% RTP. The Completion Time of 12 hours allows time for plant stabilization, data taking, and instrument calibration. The Frequency requires the SR be performed every 31 days after the initial performance prior to operation above 90% RTP. Requiring the SR prior to operations above 90% RTP is because of the increased uncertainties associated with using uncalibrated excore detectors. If the excore channels are not properly calibrated to agree with the incore detectors, power is restricted during subsequent operations because of increased uncertainty associated with using uncalibrated excore channels. The 31-day Frequency is adequate, based on operating experience of the excore linear amplifiers and the slow burnup of the detectors. The excore readings are a strong function of the power produced in the peripheral fuel bundles and do not represent an integrated reading across the core. Slow changes in neutron flux during the fuel cycle can also be detected at this Frequency.

SR 3.3.1.4

A CHANNEL FUNCTIONAL TEST is performed on each RPS instrument channel, except Loss of Load and Rate of Change of Power, every 92 days to ensure the entire channel will perform its intended function when needed.

In addition to reference voltage power supply tests, the RPS CHANNEL FUNCTIONAL TEST consists of three overlapping tests as described in Reference 1, Section 7.2. These tests verify that the RPS is capable of performing its intended function, from bistable input through the RTCBs. They include:

Bistable Tests

The bistable setpoint must be found to trip within the Allowable Values specified in the LCO and left set consistent with the assumptions of Reference 4. As-found values must also be recorded and reviewed for consistency with the assumptions of the frequency extension analysis. The requirements for this review are outlined in Reference 8.

A test signal is substituted as the input in one instrument channel at a time to verify that the bistable trip unit trips within the specified tolerance around the setpoint. This is done with the affected RPS channel bistable trip unit bypassed. Any setpoint adjustment shall be consistent with the assumptions of Reference 4.

Matrix Logic Tests

Matrix logic tests are addressed in LCO 3.3.3. This test is performed one matrix at a time. It verifies that a coincidence in the two instrument channels for each Function removes power from the matrix relays. During testing, power is applied to the matrix relay test coils and prevents the matrix relay contacts from assuming their de-energized state. This test will detect any short circuits around the bistable contacts in the coincidence logic, such as may be caused by faulty bistable relay or trip bypass contacts.

Trip Path Tests

Trip path logic tests are addressed in LCO 3.3.3. These tests are similar to the matrix logic tests, except that test power is withheld from one matrix relay at a time, allowing the trip path circuit to de-energize, opening the affected set of RTCBs. The RTCBs must then be closed prior to testing the other three trip path circuits, or a reactor trip may result.

The Frequency of 92 days is based on the reliability analysis presented in Reference 6.

SR 3.3.1.5

A CHANNEL CALIBRATION of the excore power range channels every 92 days ensures that the channels are reading accurately and within tolerance. The SR verifies that the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift between successive calibrations to ensure that the channel remains operational between successive tests. CHANNEL CALIBRATIONS must be performed consistent with the plant-specific SRs.

The as-found and as-left values must also be recorded and reviewed for consistency with the assumptions of the Frequency extension analysis. The requirements for this review are outlined in Reference 8.

A Note is added stating that the neutron detectors are excluded from CHANNEL CALIBRATION because they are passive devices with minimal drift and because of the difficulty of simulating a meaningful signal (Reference 7). Slow changes in detector sensitivity are compensated for by performing the daily calorimetric calibration (SR 3.3.1.2) and the monthly linear subchannel gain check (SR 3.3.1.3). In addition, associated control room indications are continuously monitored by the operators.

The Frequency of 92 days is acceptable, based on plant operating experience, and takes into account indications and alarms available to the operator in the Control Room.

SR 3.3.1.6

A CHANNEL FUNCTIONAL TEST on the Loss of Load, and Rate of Change of Power channels is performed prior to a reactor startup to ensure the entire channel will perform its intended function if required. The Loss of Load sensor cannot be tested during reactor operation without causing reactor trip. The Power Rate of Change-High trip Function is required during startup operation and is bypassed when shut down or > 12% RTP.

SR 3.3.1.7

Surveillance Requirement 3.3.1.7 is a CHANNEL FUNCTIONAL TEST similar to SR 3.3.1.4, except SR 3.3.1.7 is applicable only to Functions with automatic bypass removal features. Proper operation of operating bypasses are critical during plant startup because the bypasses must be in place to allow startup operation and must be removed at the appropriate points during power ascent to enable certain reactor trips. A 24-month SR Frequency is adequate to ensure proper automatic bypass removal feature operation as described in Reference 5. Once the operating bypasses are removed, the bypasses must not fail in such a way that the associated trip Function gets inadvertently bypassed. This feature is verified by the trip Function CHANNEL FUNCTIONAL TEST, SR 3.3.1.4. Therefore, further testing of the automatic bypass removal feature after startup is unnecessary.

SR 3.3.1.8

Surveillance Requirement 3.3.1.8 is the performance of a CHANNEL CALIBRATION every 24 months.

CHANNEL CALIBRATION is a check of the instrument channel, including the sensor. The SR verifies that the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument channel drift between successive calibrations to ensure that the channel remains operational between successive tests. CHANNEL CALIBRATIONS must be performed consistent with Reference 4.

The as-found and as-left values must also be recorded and reviewed for consistency with the assumptions of the frequency extension analysis. The requirements for this review are outlined in Reference 6.

The Frequency is based upon the assumption of a 24-month calibration interval for the determination of the magnitude of equipment drift.

The SR is modified by a Note to indicate that the neutron detectors are excluded from CHANNEL CALIBRATION because they are passive devices with minimal drift, and because of the difficulty of simulating a meaningful signal. Slow changes in detector sensitivity are compensated for by performing the daily calorimetric calibration (SR 3.3.1.2) and the monthly linear subchannel gain check (SR 3.3.1.3).

SR 3.3.1.9

This SR ensures that the RPS RESPONSE TIMES are verified to be less than or equal to the maximum values assumed in the safety analysis. Individual component response times are

BASES

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 RTCBs open. Response times are conducted on a 24-month STAGGERED TEST BASIS. Response time testing acceptance criteria are included in Reference 1, Section 7.2. This results in the interval between successive SRs of a given channel of n x 24 months, where n is the number of channels in the function. The Frequency of 24 months is based upon operating experience, which has shown that random failures of instrumentation components causing serious response time degradation, but not channel failure, are infrequent occurrences. Also, response times cannot be determined at power since equipment operation is required. Testing may be performed in one measurement or in overlapping segments, with verification that all components are tested.

A Note is added to indicate that the neutron detectors are excluded from RPS RESPONSE TIME testing because they are passive devices with minimum drift, and because of the difficulty of simulating a meaningful signal. Slow changes in detector sensitivity are compensated for by performing the daily calorimetric calibration (SR 3.3.1.3).

REFERENCES

1. Updated Final Safety Analysis Report
2. Title 10 Code of Federal Regulations
3. Institute of Electrical and Electronic Engineers (IEEE) No. 279, "Proposed IEEE Criteria for Nuclear Power Plant Protection Systems," August 1968
4. CCNPP Setpoint File
5. Letter from Mr. R. E. Denton (BGE) to NRC Document Control Desk, dated June 5, 1995, "Response to NRC Request for Review & Comment on Review of Preliminary Accident Precursor Analysis of Trip; Loss of 13.8 kV Bus; Short-Term Saltwater Cooling System Unavailability, CCNPP Unit 2"
6. Combustion Engineering Topical Report CEN-327, "RPS/ESFAS Extended Test Interval Evaluation" dated June 2, 1986, including Supplement 1, March 3, 1989

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7. Letter from Mr. D. G. McDonald (NRC) to Mr. R. E. Denton (BGE), dated October 19, 1995, "Issuance of Amendments for Calvert Cliffs Nuclear Power Plant, Unit No. 1 (TAC No. M92479) and Unit No. 2 (TAC No. M92480)

8. Calvert Cliffs Procedure EN-4-104, "Surveillance Testing"

BASES

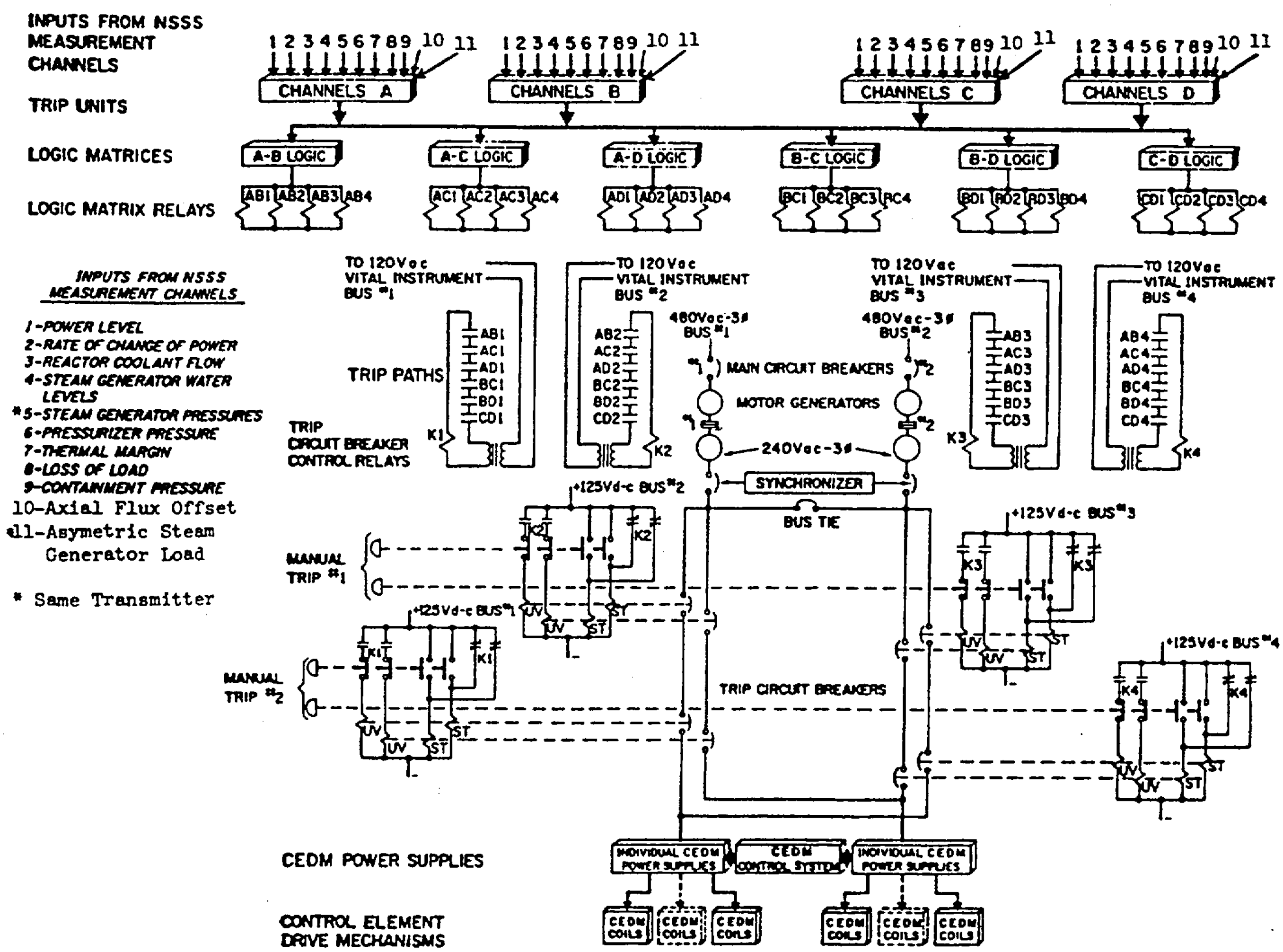

Figure B 3.3.1-1
Functional Diagram of the Two-Out-of-Four Logic and RTCB Configuration

B 3.3 INSTRUMENTATION

B 3.3.2 Reactor Protective System (RPS) Instrumentation-Shutdown

BASES

BACKGROUND

The RPS initiates a reactor trip to protect against violating the core specified acceptable fuel design limits and reactor coolant pressure boundary integrity during AOOs. By tripping the reactor, the RPS also assists the ESF systems in mitigating accidents.

The protective systems have been designed to ensure safe operation of the reactor. This is achieved by specifying LSSS in terms of parameters directly monitored by the RPS, as well as LCOs on other reactor system parameters and equipment performance.

The LSSS, defined in this Specification as the Allowable Value, in conjunction with the LCOs, establish the threshold for protective system action to prevent exceeding acceptable limits during DBAs.

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

- The DNBR shall be maintained above the SL value to prevent departure from nucleate boiling;
- Fuel centerline melting shall not occur; and
- The RCS pressure SL of 2750 psia shall not be exceeded.

Maintaining the parameters within the above values ensures that the offsite dose will be within the Reference 1 criteria during AOOs.

Accidents are events that are analyzed even though they are not expected to occur during the plant life. The acceptable limit during accidents is that the offsite dose shall be maintained within the acceptance criteria given in Reference 2.

Meeting the acceptable dose limit for an accident category is considered having acceptable consequences for that event.

BASES

The RPS is segmented into four interconnected modules. These modules are:

- Measurement channels;
- Bistable trip units;
- RPS logic; and
- RTCBs.

This LCO applies only to the Rate of Change of Power-High trip Functions and associated instrument channels in MODEs 3, 4, and 5 with any of the RTCBs closed and any CEA capable of being withdrawn. In MODEs 1 and 2, this trip Function is addressed in LCO 3.3.1. Limiting Condition for Operation 3.3.12 applies when the RTCBs are open or CEDM System is not capable of CEA withdrawal. In the case of LCO 3.3.12, the wide range logarithmic neutron flux channels are required for monitoring neutron flux, although the trip Function is not required.

Measurement Channels and Bistable Trip Units

The measurement channels providing input to the Rate of Change of Power-High trip Function consist of wide range NI channels using neutron flux leakage from the reactor vessel.

Other aspects of the Rate of Change of Power-High trip are similar to the other RPS measurement channels and bistable trip units. These are addressed in the Background section of LCO 3.3.1.

APPLICABLE SAFETY ANALYSES

Most of the analyzed accidents and transients can be detected by one or more RPS Functions. The accident analysis contained in Reference 2 takes credit for most RPS trip Functions. Some Functions not specifically credited in the accident analysis were qualitatively credited in the safety analysis and the NRC staff-approved licensing basis for the plant. These Functions may provide protection for conditions that do not require dynamic transient analysis to demonstrate Function performance. Other Functions, such as the Loss of Load trip, are purely equipment protective, and their use minimizes the potential for equipment damage.

BASES

The Rate of Change of Power-High trip is used to trip the reactor when excore wide range power indicates an excessive rate of change.

The Rate of Change of Power-High trip serves as a backup to the administratively-enforced startup rate limit.

The Rate of Change of Power-High trip Function minimizes transients for events such as a continuous CEA withdrawal or a boron dilution event from low power levels. The Rate of Change of Power-High trip is automatically bypassed at < 1E-4% RTP, as sensed by the wide range NI flux trip bistable, when poor counting statistics may lead to erroneous indication. It is also bypassed at > 12% RTP, where moderator temperature coefficient and fuel temperature coefficient make high rate of change of power unlikely. This bypass is effected by the power range NI Level 1 bistable. Automatic bypass removal is also effected by these bistables. With the RTCBs open, the Rate of Change of Power-High trip is not required to be OPERABLE; however, the indication and alarm Functions of at least two wide range channels are required to be OPERABLE. Limiting Condition for Operation 3.3.12 ensures the wide range channels are available to detect and alert the operator to a boron dilution event, when LCOs 3.3.1 and 3.3.2 are not applicable.

The RPS instrumentation satisfies 10 CFR 50.36(c)(2)(ii), Criterion 3.

LCO

The LCO requires all instrumentation performing an RPS Function to be OPERABLE. Failure of any required portion of the instrument channel renders the affected channel(s) inoperable and reduces the reliability of the affected Functions.

Actions allow trip bypass of individual instrument channels, but administrative controls prevent operation with a second channel in the same Function bypassed. Plants are in a trip bypass condition before either restoring the Function to four channel operation (two-out-of-four logic) or placing the channel in trip (one-out-of-three logic).

BASES

This LCO requires four instrument channels and automatic bypass removal features of Rate of Change of Power-High trip to be OPERABLE in MODEs 3, 4, and 5, when the RTCBs are closed and the CEDM System is capable of CEA withdrawal. MODE 1 and 2 requirements are addressed in LCO 3.3.1. This trip is not credited in the safety analysis. Therefore, the Allowable Value is not derived from an analytical limit.

APPLICABILITY

This LCO is applicable to the Rate of Change of Power-High trip in MODEs 3, 4, and 5. MODEs 1 and 2 are addressed in LCO 3.3.1.

The power rate of change trip is required in MODEs 3, 4, and 5, with the RTCBs closed and a CEA capable of being withdrawn to provide backup protection for boron dilution and CEA withdrawal events. The power rate of change trip is not credited in the safety analysis, but is part of the NRC-approved licensing basis for the plant.

The power rate of change trip has operating bypasses discussed in the LCO section. In MODEs 3, 4, and 5, the emphasis is placed on return to power events. The reactor is protected in these MODEs by ensuring adequate SDM.

ACTIONS

The most common causes of instrument channel inoperability are outright failure or drift of the bistable trip unit or measurement channel sufficient to exceed the tolerance allowed by Reference 3. Typically, the drift is found to be small, which at worst results in a delay of actuation rather than a total loss of Function. This determination is generally made during the performance of a CHANNEL CALIBRATION when the process instrument is set up for adjustment to bring it to within specification. Sensor drift could also be identified during the CHANNEL CHECKS. CHANNEL FUNCTIONAL TESTS identify bistable trip unit drift. If the trip setpoint is less conservative than the Allowable Value in Table 3.3.1-1, the instrument channel is declared inoperable immediately, and the appropriate Condition(s) must be entered immediately.

In the event that either an instrument channel's trip setpoint is found nonconservative with respect to the Allowable Value, or the transmitter, instrument loop, signal processing electronics, RPS bistable trip unit, or automatic bypass removal feature when bypass is in effect, is found inoperable, then the Rate of Change of Power-High trip Function provided by that instrument channel must be declared inoperable and the plant must enter the Condition for the particular RPS Function affected.

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

Condition A applies to the failure of a single instrument channel of the Rate of Change of Power-High trip RPS automatic trip Function.

Reactor Protective System coincidence logic is normally two-out-of-four. If one RPS bistable trip unit or associated measurement channel is inoperable, startup or power operation is allowed to continue, providing the inoperable bistable trip unit is placed in bypass or trip within 1 hour (Required Action A.1).

The Completion Time of 1 hour allotted to restore, bypass, or trip the instrument channel is sufficient to allow the operator to take all appropriate actions for the failed channel, while ensuring that the risk involved in operating with the failed channel is acceptable.

The failed instrument channel is restored to OPERABLE status or is placed in trip within 48 hours (Required Action A.2.1 or Required Action A.2.2). Required Action A.2.1 restores the full capability of the Function. Required Action A.2.2 places the Function in a one-out-of-three coincidence logic. In this coincidence logic, common cause failure of dependent channels cannot prevent trip.

The 48-hour Completion Time is based on operating experience, which has demonstrated that a random failure of a second instrument channel occurring during the 48-hour period is a low probability event.

B.1 and B.2

Condition B applies to the failure of two instrument channels in the Rate of Change of Power-High trip RPS automatic trip Function.

BASES

Required Action B.1 provides for placing one inoperable instrument channel in bypass and the other channel in trip within the Completion Time of 1 hour. This Completion Time is sufficient to allow the operator to take all appropriate actions for the failed channels, while ensuring the risk involved in operating with the failed channels is acceptable. With one instrument channel bypassed, the RPS Function is in a two-out-of-three logic; but with another channel failed, the RPS Function may be operating in a two-out-of-two logic. This is outside the assumptions made in the analyses and should be corrected. To correct the problem, the second channel is placed in trip. This places the RPS Function in a one-out-of-two logic. If any of the other OPERABLE channels receives a trip signal, the reactor will trip.

The bypassed instrument channel should be restored to OPERABLE status within 48 hours for reasons similar to those stated under Condition A. After one channel is restored to OPERABLE status, the provisions of Condition A still apply to the remaining inoperable channel. Therefore, the channel that is still inoperable after completion of Required Action B.2 shall be placed in trip if more than 48 hours have elapsed since the initial channel failure.

The Required Action is modified by a Note stating that LCO 3.0.4 is not applicable. The Note was added to allow the changing of MODEs, even though two instrument channels are inoperable, with one channel bypassed and one tripped. MODE changes in this configuration are allowed to permit maintenance and testing on one of the inoperable channels. In this configuration, the protective system is in a one-out-of-two logic, and the probability of a common cause failure affecting both of the OPERABLE channels during the 48 hours permitted is remote.

<u>C.1, C.2.1, C.2.2.1, and C.2.2.2</u>

Condition C applies to one automatic bypass removal feature inoperable. If the automatic bypass removal feature cannot be restored to OPERABLE status, the associated Rate of Change of Power-High trip RPS channel may be considered

OPERABLE only if the bypass is not in effect. Otherwise, the affected RPS channel must be declared inoperable, as in Condition A, and the bypass either removed or the automatic bypass removal feature repaired. The Bases for the Required Actions and Completion Times are the same as discussed for Condition A.

D.1, D.2.1, and D.2.2

Condition D applies to two inoperable automatic bypass removal features. If the automatic bypass removal features cannot be restored to OPERABLE status, the associated Rate of Change of Power-High trip RPS channel may be considered OPERABLE only if the bypasses are not in effect. Otherwise, the affected RPS channels must be declared inoperable, as in Condition B, and the bypasses either removed or the automatic bypass removal features repaired. Also, Required Action D.2.2 provides for the restoration of the one affected automatic trip channel to OPERABLE status within the rules of Completion Time specified under Condition B. Completion Times are consistent with Condition B.

The Required Action is modified by a Note stating that LCO 3.0.4 is not applicable. The Note was added to allow the changing of MODEs, even though two instrument channels are inoperable, with one channel bypassed and one tripped. MODE changes in this configuration are allowed to permit maintenance and testing on one of the inoperable channels. In this configuration, the Function is in a one-out-of-two logic, and the probability of a common cause failure affecting both of the OPERABLE channels during the 48 hours permitted is remote.

E.1

Condition E is entered when the Required Actions and associated Completion Times of Condition A, B, C, or D are not met.

If Required Actions associated with these Conditions cannot be completed within the required Completion Time, opening the RTCBs brings the reactor to a MODE where the LCO does not apply and ensures no CEA withdrawal will occur. The basis for the Completion Time of 6 hours is that it is adequate to complete the Required Actions without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.3.2.1

Performance of the CHANNEL CHECK on each wide range channel once every 12 hours ensures that gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one instrument channel to a similar parameter on another channel. 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 channel 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.

Agreement criteria are determined by the plant staff based on a qualitative assessment of the instrument channel that considers instrument channel uncertainties, including isolation, indication, and readability. If a channel is outside the criteria, it may be an indication that the transmitter or the signal processing equipment has drifted outside its limits.

The Frequency, once every shift, is based on operating experience that demonstrates the rarity of instrument channel failure. Since the probability of two random failures in redundant channels in any 12-hour period is extremely low, the CHANNEL CHECK minimizes the chance of loss of RPS Function due to failure of redundant channels. The CHANNEL CHECK supplements less formal, but more frequent, checks of the channel during normal operational use of the displays.

SR 3.3.2.2

A CHANNEL FUNCTIONAL TEST on the power rate of change channels is performed once within 7 days prior to each reactor startup to ensure the entire instrument channel will perform its intended function if required. The Rate of Change of Power-High trip Function is required during startup operation and is bypassed when shut down or > 12% RTP. Additionally, operating experience has shown that these components usually pass the SR when performed at a Frequency of once within 7 days prior to each reactor startup.

Only the Allowable Values are specified for each RPS trip Function in the SR. Nominal trip setpoints are established for the Functions via the plant-specific procedures. The nominal setpoints are selected to ensure the plant parameters do not exceed the Allowable Value if the bistable trip unit is performing as required. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable, provided that operation and testing are consistent with the assumptions of the plant-specific setpoint calculations. Each nominal trip setpoint is more conservative than the analytical limit assumed in the safety analysis in order to account for instrument channel uncertainties appropriate to the trip Function. These uncertainties are defined in Reference 3.

SR 3.3.2.3

Surveillance Requirement 3.3.2.3 is a CHANNEL FUNCTIONAL TEST similar to SR 3.3.2.2, except SR 3.3.2.3 is applicable only to bypass Functions and is performed once every 24 months.

Proper operation of operating bypasses is critical during plant startup because the bypasses must be in place to allow startup operation and must be removed at the appropriate points during power ascent to enable certain reactor trips. A 24-month SR Frequency is adequate to ensure proper automatic bypass removal feature operation as described in Reference 5. Once the operating bypasses are removed, the bypasses must not fail in such a way that the associated trip Function gets inadvertently bypassed. This feature is verified by SR 3.3.2.2. Therefore, further testing of the automatic bypass removal feature after startup is unnecessary.

BASES

SR 3.3.2.4

Surveillance Requirement 3.3.2.4 is the performance of a CHANNEL CALIBRATION every 24 months.

CHANNEL CALIBRATION is a check of the instrument channel including the sensor. The SR verifies that the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift between successive calibrations to ensure that the channel remains operational between successive tests. CHANNEL CALIBRATIONS must be performed consistent with Reference 3.

The as-found and as-left values must also be recorded and reviewed for consistency with the assumptions of the SR interval extension analysis. The requirements for this review are outlined in Reference 4.

The Frequency is based upon the assumption of a 24-month calibration interval in the determination of the magnitude of equipment drift.

The SR is modified by a Note to indicate that the neutron detectors are excluded from CHANNEL CALIBRATION because they are passive devices with minimal drift (Reference 5).

REFERENCES

1. 10 CFR Parts 50, "Domestic Licensing of Production and Utilization Facilities," and 100, "Reactor Site Criteria"
2. Updated Final Safety Analysis Report (UFSAR), Chapter 14, "Safety Analysis"
3. CCNPP Setpoint File
4. Combustion Engineering Topical Report CEN-327, "RPS/ESFAS Extended Test Interval Evaluation" dated June 2, 1986, including Supplement 1, March 3, 1989
5. Letter from Mr. R. E. Denton (BGE) to NRC Document Control Desk, dated June 6, 1995, "License Amendment Request; Extension of Instrument Surveillance Intervals"

B 3.3 INSTRUMENTATION

B 3.3.3 Reactor Protective System (RPS) Logic and Trip Initiation

BASES

BACKGROUND

The RPS initiates a reactor trip to protect against violating the core specified acceptable fuel design limits and reactor coolant pressure boundary integrity during AOOs. By tripping the reactor, the RPS also assists the ESF systems in mitigating accidents.

The protective systems have been designed to ensure safe operation of the reactor. This is achieved by specifying LSSS in terms of parameters directly monitored by the RPS, as well as LCOs on other reactor system parameters and equipment performance.

The LSSS, defined in this Specification as the Allowable Value, in conjunction with the LCOs, establish the threshold for protective system action to prevent exceeding acceptable limits during DBAs.

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

- The DNBR shall be maintained above the SL value to prevent departure from nucleate boiling;
- Fuel centerline melting shall not occur; and
- The RCS pressure SL of 2750 psia shall not be exceeded.

Maintaining the parameters within the above values ensures that the offsite dose will be within the Reference 2 criteria during AOOs.

Accidents are events that are analyzed even though they are not expected to occur during the plant life. The acceptable limit during accidents is that the offsite dose shall be maintained within the acceptance criteria given in Reference 1, Chapter 14.

The RPS is segmented into four interconnected modules. These modules are:

- Measurement channels;
- Bistable trip units;

- RPS logic; and
- RTCBs.

This LCO addresses the RPS logic and RTCBs, including manual trip capability. Limiting Condition for Operation 3.3.1 provides a description of the role of this equipment in the RPS. This is summarized below:

RPS Logic

The RPS logic, consisting of matrix and trip path logic, employs a scheme that provides a reactor trip when bistable trip units in any two of the four instrument channels sense the same input parameter trip. This is called a two-out-of-four trip logic. This logic and the RTCB configuration are shown in Figure B 3.3.1-1.

Bistable relay contact outputs from the four bistable trip unit channels are configured into six logic matrices. Each logic matrix checks for a coincident trip in the same parameter in two bistable trip unit channels. The matrices are designated the AB, AC, AD, BC, BD, and CD matrices to reflect the bistable trip unit channels being monitored. Each logic matrix contains four normally energized matrix relays. When a coincidence is detected, consisting of a trip in the same Function in the two channels being monitored by the logic matrix, all four matrix relays de-energize.

The logic matrix relay contacts are arranged into trip paths, with one of the four matrix relays in each matrix opening contacts in one of the four trip paths. Each trip path provides power to one of the four normally energized RTCB control relays (K1, K2, K3, and K4). Thus, the trip paths each have six contacts in series, one from each matrix, and perform a logical OR function, opening the RTCBs if any one or more of the six logic matrices indicate a coincidence condition.

Each trip path is responsible for opening one set of two of the eight RTCBs. The RTCB control relays (K-relays), when de-energized, interrupt power to the breaker undervoltage trip coils and simultaneously apply power to the shunt trip coils on each of the two breakers. Actuation of either the undervoltage or shunt trip coil is sufficient to open the RTCB and interrupt power from the MG sets to the CEDMs.

When a coincidence occurs in two RPS channels from one Function, all four matrix relays in the affected matrix de-energize. This in turn de-energizes all four breaker control relays, which simultaneously de-energize the undervoltage and energize the shunt trip coils in all eight RTCBs, tripping them open.

The trip path logic consists of the trip path power source, matrix relays and their associated contacts, and all interconnecting wiring, through the K-relay contacts in the RTCB control circuitry.

It is possible to change the two-out-of-four RPS logic to a two-out-of-three logic for a given input parameter in one instrument channel at a time by trip bypassing select portions of the matrix logic. Trip bypassing a bistable effectively shorts the bistable relay contacts in the three matrices associated with that channel. Thus, the bistables will function normally, producing normal trip indication and annunciation, but a reactor trip will not occur unless two additional instrument channels indicate a trip condition. Trip bypassing can be simultaneously performed on any number of parameters in any number of Functions, providing each parameter is bypassed in only one instrument channel per Function at a time. Administrative controls prevent simultaneous trip bypassing of the same parameter in more than one instrument channel. Trip bypassing is normally employed during maintenance or testing.

<u>RTCBs</u>

The reactor trip switchgear, shown in Figure B 3.3.1-1, consists of eight RTCBs, which are operated in four sets of two breakers (four RTCB channels including the shunt trip coils and undervoltage coils). Power input to the reactor trip switchgear comes from two full capacity MG sets operated in parallel such that the loss of either MG set does not de-energize the CEDMs. There are two separate CEDM power supply buses, each bus powering half of the CEDMs.

BASES

Power is supplied from the MG sets to each bus via two redundant trip legs. This ensures that a fault or the opening of a breaker in one trip leg (i.e., for testing purposes) will not interrupt power to the CEDM buses.

Each of the four trip paths consists of two RTCBs in series. The two RTCBs within a trip path are actuated by separate trip paths.

The eight RTCBs are operated as four sets of two breakers (four RTCB channels including the shunt trip coils and undervoltage coils). Each set of two RTCBs is opened by the same K-relay. This arrangement ensures that power is interrupted to both CEDM buses, thus preventing trip of only half of the CEAs (a half trip). Any one inoperable RTCB in a RTCB channel (set of two breakers) will make the entire RTCB channel inoperable.

Each set of RTCBs is operated by either a manual trip push button or an RPS actuated K-relay. There are four manual trip push buttons, arranged in two sets of two, as shown in Figure B 3.3.1-1. Depressing both push buttons in either set will result in a reactor trip.

When a manual trip is initiated using the control room push buttons, the RPS trip paths and K-relays are bypassed, and the RTCB undervoltage and shunt trip coils are actuated independent of the RPS.

A manual trip channel includes the push button and interconnecting wiring to both RTCBs necessary to actuate both the undervoltage and shunt trip coils, but excludes the K-relay contacts and their interconnecting wiring to the RTCBs, which are considered part of the trip path logic.

Functional testing of the entire RPS instrument and logic channels, from bistable input through the opening of individual sets of RTCBs, can be performed either at power or shut down, and is normally performed on a quarterly basis. Reference 1, Section 7.2 explains RPS testing in more detail.

BASES

APPLICABLE SAFETY ANALYSES

RPS Logic

The RPS logic provides for automatic trip initiation to maintain the SLs during AOOs and assist the ESF systems in ensuring acceptable consequences during accidents. All transients and accidents that call for a reactor trip assume the RPS logic is functioning as designed.

RTCBs

All of the transient and accident analyses that call for a reactor trip assume that the RTCBs operate and interrupt power to the CEDMs.

Manual Trip

There are no accident analyses that take credit for the manual trip; however, the manual trip is part of the RPS circuitry. It is used by the operator to shut down the reactor whenever any parameter is rapidly trending toward its trip setpoint. A manual trip accomplishes the same results as any one of the automatic trip Functions.

The RPS logic and initiation satisfy 10 CFR 50.36(c)(2)(ii), Criterion 3.

LCO

RPS Logic

Failures of individual bistable relays and their contacts are addressed in LCO 3.3.1. This Specification addresses failures of the matrix logic not addressed in the above, such as the failure of matrix relay power supplies or the failure of the trip bypass contact in the bypass condition.

Loss of a single vital bus will de-energize one of the two power supplies in each of three matrices. This will result in two sets of two RTCBs opening; however, the remaining two sets of two closed RTCBs will prevent a reactor trip. For the purposes of this LCO, de-energizing up to three matrix power supplies due to a single failure is to be treated as a single channel failure, providing the affected matrix relays de-energize as designed, opening the affected RTCBs.

BASES

Each of the four trip path logic channels opens one set of RTCBs if any of the six logic matrices de-energize their associated matrix relays. Thus, they perform a logical <u>OR</u> function. Each trip path logic channel has its own power supply and is independent of the others. A trip path logic channel includes the matrix relay through to the K-relay contacts, which open the RTCB.

It is possible for two trip path logic channels affecting the same trip leg to de-energize if a matrix power supply or vital instrument bus fails. This will result in opening the two affected sets of two RTCBs.

If one set of RTCBs has been opened in response to a single RTCB channel, trip path logic channel, or manual trip channel failure, the affected set of RTCBs may be closed for up to 1 hour for surveillance on the OPERABLE trip path logic, RTCB, and manual trip channels. In this case, the redundant set of RTCBs will provide protection if a trip should be required. It is unlikely that a trip will be required during the SR, coincident with a failure of the remaining series RTCB channel. If a single matrix power supply or vital bus failure has opened two sets of RTCBs, manual trip and RTCB testing on the closed breakers cannot be performed without causing a trip.

1. <u>Matrix Logic</u>

   This LCO requires six channels of matrix logic to be OPERABLE in MODEs 1 and 2, and in MODEs 3, 4, and 5 when any RTCB is closed and any CEA is capable of being withdrawn.

2. <u>Trip Path Logic</u>

   This LCO requires four channels of trip path logic to be OPERABLE in MODEs 1 and 2, and in MODEs 3, 4, and 5 when any RTCB is closed and any CEA is capable of being withdrawn.

3. <u>RTCBs</u>

   The LCO requires four RTCB channels to be OPERABLE in MODEs 1 and 2, as well as in MODEs 3, 4, and 5 when any

RTCB is closed and any CEA is capable of being withdrawn.

Each RTCB channel consists of two breakers operated in a single set by the trip path logic or manual trip circuitry. This ensures that power is interrupted at identical locations in the trip paths for both CEDM buses, thus preventing power removal to only one CEDM bus (a half trip).

Failure of a single breaker affects the entire RTCB channel, and both breakers in the set must be opened. Without reliable RTCBs and associated support circuitry, a reactor trip cannot occur whether initiated automatically or manually.

Each channel of RTCBs starts at the contacts actuated by the K-relay, and the contacts actuated by the manual trip, for each set of breakers. The K-relay actuated contacts and the upstream circuitry are considered to be RPS logic. Manual trip contacts and upstream circuitry are considered to be part of the manual trip channels.

A Note associated with the ACTIONS states that if one set of RTCBs has been opened in response to a single RTCB channel, trip path logic channel, or manual trip channel failure, the affected set of RTCBs may be closed for up to 1 hour for a surveillance test on the OPERABLE trip path logic, RTCB, and manual trip channels. In this case, the redundant set of RTCBs will provide protection. If a single matrix power supply or vital bus failure has opened two sets of RTCBs, manual trip and RTCB testing on the closed breakers cannot be performed without causing a trip. This Note is not applicable to Condition A, with one matrix logic channel inoperable.

4. <u>Manual Trip</u>

The LCO requires all four manual trip channels to be OPERABLE in MODEs 1 and 2, and MODEs 3, 4, and 5 when

BASES

any RTCB is closed and any CEA is capable of being withdrawn.

Two independent sets of two adjacent push buttons are provided at separate locations. Each push button is considered a channel and operates two of the eight RTCBs. Depressing both push buttons in either set will cause an interruption of power to the CEDMs, allowing the CEAs to fall into the core. This design ensures that no single failure in any push button channel can either cause or prevent a reactor trip.

APPLICABILITY

The RPS matrix logic, RTCBs, and manual trip are required to be OPERABLE in any MODE when any CEA is capable of being withdrawn from the core (i.e., RTCBs closed and power available to the CEDMs). This ensures the reactor can be tripped when necessary, but allows for maintenance and testing when the reactor trip is not needed.

In MODEs 3, 4, and 5, with all the RTCBs open, the CEAs are not capable of withdrawal and these Functions do not have to be OPERABLE. However, two wide range logarithmic neutron flux monitor channels must be OPERABLE to ensure proper indication of neutron population and to indicate a boron dilution event. This is addressed in LCO 3.3.12.

ACTIONS

When the number of inoperable RPS logic or trip initiation channels exceeds that specified in any related Condition, the plant is outside the safety analysis. Therefore, LCO 3.0.3 is immediately entered if applicable in the current MODE of operation.

A.1

Condition A applies if one matrix logic channel is inoperable or three logic matrices channels are inoperable due to a common power source failure de-energizing three matrix power supplies in any applicable MODE.

The matrix logic channel must be restored to OPERABLE status within 48 hours. The Completion Time of 48 hours provides the operator time to take appropriate actions and still ensures that any risk involved in operating with a failed channel is acceptable. Operating experience has demonstrated that the probability of a random failure of a second matrix logic channel is low during any given 48-hour interval. If the channel cannot be restored to OPERABLE status within 48 hours, Condition E is entered.

B.1

Condition B applies to one trip path logic channel, RTCB channel, or manual trip channel in MODEs 1 and 2, since they have the same actions. MODEs 3, 4, and 5, with the RTCBs shut, are addressed in Condition C. These Required Actions require opening the affected RTCBs. This removes the need for the affected channel by performing its associated safety function. With the RTCB open, the affected Functions are in one-out-of-two logic, which meets redundancy requirements, but testing on the OPERABLE channels cannot be performed without causing a reactor trip unless the RTCBs in the inoperable channels are closed to permit testing. Limiting Condition for Operation 3.0.5 allows the RTCBs associated with the inoperable channel to be closed to perform testing.

Required Action B.1 provides for opening the RTCBs associated with the inoperable channel within a Completion Time of 1 hour. This Required Action is conservative, since depressing the manual trip push button associated with either set of breakers in the other trip leg will cause a reactor trip. With this configuration, a single channel failure will not prevent a reactor trip. The allotted Completion Time is adequate to open the affected RTCBs, while maintaining the risk of having them closed at an acceptable level.

C.1

Condition C applies to the failure of one trip path logic channel, RTCB channel, or manual trip channel in MODE 3, 4, or 5 with the RTCBs closed. The channel must be restored to OPERABLE status within 48 hours. If the inoperable channel cannot be restored to OPERABLE status within 48 hours, all RTCBs must be opened, placing the plant in a MODE in which the LCO does not apply and ensuring no CEA withdrawal occurs.

BASES

The Completion Time of 48 hours is consistent with that of other RPS instrumentation and should be adequate to repair most failures.

Testing on the OPERABLE channels cannot be performed without causing a reactor trip unless the RTCBs in the inoperable channels are closed to permit testing. Limiting Condition for Operation 3.0.5 allows the RTCBs associated with the inoperable channel to be closed to perform testing.

D.1

Condition D applies to the failure of both trip path logic channels affecting the same trip leg. Since this will open two channels of RTCBs, this Condition is also applicable to the two affected channels of RTCBs. This Condition allows for loss of a single vital instrument bus or matrix power supply, which will de-energize both trip path logic channels in the same trip leg. This will open both sets of RTCBs in the affected trip leg, satisfying the Required Action of opening the affected channels of RTCBs.

Of greater concern is the failure of the trip path circuit in a nontrip condition (e.g., due to two trip path K-relay failures). With only one trip path logic channel failed in a nontrip condition, there is still the redundant set of RTCBs in the trip leg. With both failed in a nontrip condition, the reactor will not trip automatically when required. In either case, the affected RTCBs must be opened immediately by using the appropriate manual trip push buttons, since each of the four push buttons opens one set of RTCBs, independent of the trip path circuitry. Caution must be exercised, since depressing the wrong push buttons may result in a reactor trip.

If the affected RTCB(s) cannot be opened, Condition E is entered. This would only occur if there is a failure in the manual trip channel or the RTCB(s).

E.1 and E.2

Condition E is entered if Required Actions associated with Condition A, B, or D are not met within the required Completion Time or if one or more Functions with more than one manual trip, matrix logic, trip path logic, or RTCB channel is inoperable for reasons other than Condition A or D.

If the RTCBs associated with the inoperable channel cannot be opened, the reactor must be shut down within 6 hours and all the RTCBs opened. A Completion Time of 6 hours is reasonable, based on operating experience, to reach the required MODE from full power conditions in an orderly manner, without challenging plant systems, and to open RTCBs. All RTCBs should then be opened, placing the plant in a MODE where the LCO does not apply and ensuring no CEA withdrawal occurs.

SURVEILLANCE REQUIREMENTS

SR 3.3.3.1

A CHANNEL FUNCTIONAL TEST is performed on each RTCB channel every 31 days. This verifies proper operation of each RTCB. The RTCB must then be closed prior to testing the other RTCBs, or a reactor trip may result. The frequency of 31 days is based on the reliability analysis presented in Reference 3.

SR 3.3.3.2

A CHANNEL FUNCTIONAL TEST on each RPS logic channel is performed every 92 days to ensure the entire channel will perform its intended function when needed.

In addition to reference voltage tests, the RPS CHANNEL FUNCTIONAL TEST consists of three overlapping tests as described in Reference 1, Section 7.2. These tests verify that the RPS is capable of performing its intended function, from bistable input through the RTCBs. The first test, the instrument channel test, is addressed by SR 3.3.1.4 in LCO 3.3.1.

This SR addresses the two tests associated with the RPS logic: matrix logic and trip path logic.

Matrix Logic Tests

These tests are performed one matrix at a time. They verify that a coincidence in the two instrument channels for each

BASES

Function removes power from the matrix relays. During testing, power is applied to the matrix relay test coils and prevents the matrix relay contacts from assuming their de-energized state. The matrix logic tests will detect any short circuits around the bistable contacts in the coincidence logic such as may be caused by faulty bistable relay or trip bypass contacts.

Trip Path Tests

These tests are similar to the matrix logic tests, except that test power is withheld from one matrix relay at a time, allowing the trip path circuit to de-energize, opening the affected set of RTCBs. The RTCBs must then be closed prior to testing the other three trip path circuits, or a reactor trip may result.

The Frequency of 92 days is based on the reliability analysis presented in Reference 3.

SR 3.3.3.3

A CHANNEL FUNCTIONAL TEST on the manual trip channels is performed prior to a reactor startup to ensure the entire channel will perform its intended function if required. The manual trip Function can be tested either at power or shut down. However, the simplicity of this circuitry and the absence of drift concern makes this Frequency adequate. Additionally, operating experience has shown that these components usually pass the SR when performed once within 7 days prior to each reactor startup.

REFERENCES

1. UFSAR
2. 10 CFR Part 100, "Reactor Site Criteria"
3. Combustion Engineering Topical Report CEN-327, "RPS/ESFAS Extended Test Interval Evaluation" dated June 2, 1986, including Supplement 1, March 3, 1989

B 3.3 INSTRUMENTATION

B 3.3.4 Engineered Safety Features Actuation System (ESFAS) Instrumentation

BASES

BACKGROUND

The ESFAS actuates necessary safety systems, based upon the values of selected unit parameters to mitigate accidents in order to protect the public and plant personnel from the accidental release of radioactive fission products.

The ESFAS contains devices and circuitry that generate the following signals when the monitored variables reach levels that are indicative of conditions requiring protective action:

1. Safety Injection Actuation Signal (SIAS);
2. Containment Spray Actuation Signal (CSAS);
3. Containment Isolation Signal (CIS);
4. Steam Generator Isolation Signal (SGIS);
5. Recirculation Actuation Signal (RAS) for the Containment Sump; and
6. AFAS Signal.

Equipment actuated by each of the above signals is identified in the Reference 1, Section 7.3.

Each of the above ESFAS actuation systems is segmented into four sensor channel and two actuation logic channels. Each sensor channel includes measurement channels and bistables (sensor modules). The actuation logic channels include two sets of logic circuitry (actuation logic modules) and actuation relay equipment. The actuation logic channels actuate ESFAS equipment trains that are sequentially loaded on the diesel generators (DGs).

Each of the four sensor modules monitors redundant and independent process measurement channels. Each sensor is monitored by at least one sensor module. The sensor module associated with each ESFAS sensor channel will trip when the monitored variable exceeds the trip setpoint. When tripped, the sensor channels provide outputs to the two actuation logic channels.

BASES

The two independent actuation logic channels compare the four sensor channel outputs. If a trip occurs in the same parameter in two or more sensor channels, the two-out-of-four logic in each actuation logic channel will initiate the associated train of ESFAS. Each train can provide protection to the public in the case of a DBA. Actuation logic is addressed in LCO 3.3.5.

Each of the four sensor channels is mounted in a separate cabinet, excluding the sensors and field wiring.

The role of the sensor channel (measurement channels and sensor modules) is discussed below; actuation logic channels are discussed in LCO 3.3.5.

Measurement Channels

Measurement channels, consisting of field transmitters or process sensors and associated instrumentation, provide a measurable electronic signal based upon the physical characteristics of the parameter being measured.

Four measurement channels with electrical and physical separation are provided for each parameter used in the generation of actuation signals. These are designated Channels ZD through ZG. Measurement channels provide input to ESFAS sensor modules within the same ESFAS channel. In addition, some measurement channels may also be used as inputs to Reactor Protective System (RPS) bistable trip units, and most provide indication in the Control Room. Measurement channels used as an input to the RPS or ESFAS are not used for control functions.

When a measurement channel monitoring a parameter indicates an unsafe condition, the sensor module monitoring the parameter in that channel will trip. Tripping two or more channels of sensor modules monitoring the same parameter will de-energize both channels of actuation logic of the associated ESF equipment.

Three of the four sensor channels are necessary to meet the redundancy and testability requirements of Reference 1, Appendix 1C. The fourth channel provides additional flexibility by allowing one channel to be removed from service (maintenance bypass) for maintenance or testing while still maintaining a minimum two-out-of-three logic.

Since no single failure will either cause or prevent a protective system actuation and no protective channel feeds a control channel, this arrangement meets the requirements of proposed Reference 2.

Sensor Modules

Sensor modules receive an analog input (digital for RAS) from the measurement channels, compare the input to trip setpoints, and provide contact output to the actuation logic channels (Reference 3). They also provide local trip indication and remote annunciation.

There are four channels of sensor modules, designated ZD through ZG, for each ESF Function, one for each measurement channel.

The trip setpoints and Allowable Values used in the sensor modules are based on the analytical limits used in Reference 1, Chapter 14. The selection of these trip setpoints is such that adequate protection is provided when all sensor and processing time delays are taken into account in the respective analytical limits. To allow for calibration tolerances, instrumentation uncertainties, sensor channel drift, and severe environment errors, (for those ESFAS channels that must function in harsh environments, where appropriate, as defined by Reference 4. Engineered Safety Features Actuation System sensor modules trip setpoints are conservatively adjusted with respect to the analytical limits. A detailed description of the method used to calculate the trip setpoints, including their explicit uncertainties, is provided in Reference 5. The actual nominal trip setpoint entered into the sensor module is more conservative than that specified by the Allowable Value. If the measured setpoint does not exceed the Allowable Value, the sensor module is considered OPERABLE.

Setpoints in accordance with the Allowable Value will ensure that the consequences of AOOs and DBAs will be acceptable, providing the plant is operated from within the LCOs at the onset of the AOO or DBA and the equipment functions as designed.

ESFAS Logic

It is possible to change the two-out-of-four ESFAS logic to a two-out-of-three logic for a given input parameter in one sensor channel at a time by disabling one sensor channel input to the logic. Thus, the sensor modules will function normally, producing normal trip indication and annunciation, but ESFAS actuation will not occur since the bypassed channel is effectively removed (blocked) from the coincidence logic. Sensor channel bypassing can be simultaneously performed on any number of parameters in any number of Functions, providing each parameter is bypassed in only one sensor channel per Function at a time. Sensor channel bypassing is normally employed during maintenance or testing.

Engineered Safety Features Actuation System logic is addressed in LCO 3.3.5.

APPLICABLE SAFETY ANALYSES

Most 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. Functions such as manual actuation, not specifically credited in the accident analysis, serve as backups to Functions and are part of the NRC approved licensing basis for the plant.

ESFAS protective Functions are as follows:

1. SIAS

   The SIAS ensures acceptable consequences during LOCA events, including steam generator tube rupture, and other DBAs. To provide the required protection, either a high containment pressure or a low pressurizer pressure signal will actuate SIAS. The SIAS actuates the Emergency Core Cooling System (ECCS), control room isolation, and performs other functions, such as starting the DGs.

2. CSAS

The CSAS actuates containment spray, preventing containment overpressurization during a LOCA or MSLB. Both a high containment pressure signal and a SIAS have to actuate to provide the required protection. This configuration reduces the likelihood of inadvertent containment spray.

3. CIS

The CIS actuates the Containment Isolation System, ensuring acceptable consequences during LOCAs and other DBAs (inside Containment). A high containment pressure signal will actuate CIS.

4. SGIS

The SGIS ensures acceptable consequences during an excessive loss of steam from the Main Steam System by isolating both steam generators if either generator indicates a low steam generator pressure. The SGIS, concurrent with or following a reactor trip, minimizes the rate of heat extraction and subsequent cooldown of the RCS during these events.

5. RAS

At the end of the injection phase of a LOCA, the refueling water tank (RWT) 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. Switchover from RWT to the containment sump must occur before the RWT empties to prevent damage to the ECCS 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 pump suction. Furthermore, early switchover must not occur so sufficient borated water is injected from the RWT to ensure the reactor remains shut down in the recirculation mode. An RWT

Level-Low trip signal, generated by a level switch, actuates the RAS.

6. <u>AFAS Signal</u>

An AFAS Signal actuates feedwater flow to both steam generators if a low level is indicated in either steam generator, unless the generator is ruptured.

The AFAS Signal maintains a steam generator heat sink during the following events:

- MSLB;
- FWLB; and
- Loss of feedwater.

A low steam generator water level signal will actuate auxiliary feed to both steam generators.

Secondary steam generator differential pressure (SG-1 > SG-2) or (SG-2 > SG-1) blocks auxiliary feedwater (AFW) to a generator identified as being ruptured. This input to the AFAS logic prevents loss of the intact generator while preventing feeding a ruptured generator during MSLBs and FWLBs. This prevents containment overpressurization and/or excessive RCS cooldown during these events.

The ESFAS satisfies 10 CFR 50.36(c)(2)(ii), Criterion 3.

LCO

The LCO requires all sensor channel components necessary to provide an ESFAS actuation to be OPERABLE.

The Bases for the LCO on ESFAS Functions are:

1. <u>SIAS</u>

   a. <u>Containment Pressure-High Trip</u>

   This LCO requires four sensor channels of SIAS Containment Pressure-High trip to be OPERABLE in MODEs 1, 2, and 3.

The Allowable Value for this trip is set high enough to allow for small pressure increases in Containment expected during normal operation (i.e., plant heatup) and is not indicative of an offnormal condition. The setting is low enough to initiate the ESF Functions when a LOCA or other DBA condition is indicated. This allows the ESF systems to perform as expected in the accident analyses to mitigate the consequences of the analyzed accidents.

b. Pressurizer Pressure-Low Trip

This LCO requires four sensor channels of SIAS Pressurizer Pressure-Low trip to be OPERABLE in MODEs 1, 2, and 3.

The Allowable Value for this trip is set low enough to prevent actuating the SIAS during normal plant operation and pressurizer pressure transients. The setting is high enough that with a LOCA or some other DBA it will actuate to perform as expected, mitigating the consequences of the accidents.

The Pressurizer Pressure-Low trip may be blocked when pressurizer pressure is reduced during controlled plant shutdowns. This block is permitted below 1800 psia, and block permissive responses are annunciated in the Control Room. This allows for a controlled depressurization of the RCS, while maintaining administrative control of ESF protection. From a blocked condition, the block will be automatically removed as pressurizer pressure increases above 1800 psia, as sensed by two of the four sensor channels, in accordance with the block philosophy of removing blocks when the enabling conditions are no longer satisfied.

This LCO requires four channels of the automatic block removal features for SIAS Pressurizer Pressure-Low trip to be OPERABLE in MODEs 1, 2, and 3.

The block permissive channels consist of four sensor channels and two actuation sensor block modules. This LCO applies to failures in the four sensor channels, including measurement channels and sensor block modules. Failures in the actuation logic channels, including the manual bypass key switches, are considered actuation logic failures and are addressed in LCO 3.3.5.

This LCO applies to the automatic block removal feature, not the sensor block modules. If the block enable Function is failed so as to prevent entering a block condition, operation may continue.

The block permissive is set low enough so as not to be enabled during normal plant operation, but high enough to allow blocking prior to reaching the trip setpoint.

2. CSAS

The CSAS is initiated either manually or automatically. It is also necessary to have an automatic or manual SIAS for complete actuation. The CSAS opens the containment spray valves, where as SIAS actuates other related components. The SIAS requirement should always be satisfied on a legitimate CSAS, since the Containment Pressure-High trip signal field setpoint used in the SIAS is the same or below the setpoint used in the CSAS.

a. Containment Pressure-High Trip

This LCO requires four sensor channels of CSAS Containment Pressure-High trip to be OPERABLE in MODEs 1, 2, and 3.

The Allowable Value is set high enough to allow for small pressure increases in Containment expected during normal operation (i.e., plant heatup) and is not indicative of an offnormal condition. The setting is low enough to initiate the ESF Functions when an offnormal condition is indicated. This allows the ESF systems to perform as expected in the accident analyses to mitigate the consequences of the analyzed accidents.

The Containment Pressure-High trip setpoint is the same in the SIAS (Function 1), CIS (Function 3), and is a different setpoint for CSAS (Function 2). However, different logic is used in each of these Functions.

3. CIS

a. Containment Pressure-High Trip

This LCO requires four sensor channels of CIS Containment Pressure-High trip to be OPERABLE in MODEs 1, 2, and 3.

The Allowable Value is set high enough to allow for small pressure increases in Containment expected during normal operation (i.e., plant heatup) and is not indicative of an offnormal condition. The setting is low enough to initiate the ESF Functions when an offnormal condition is indicated. This allows the ESF systems to perform as expected in the accident analyses to mitigate the consequences of the analyzed accidents.

The Containment Pressure-High trip setpoint is the same in the SIAS (Function 1) and CIS (Function 3), and is a different setpoint for CSAS (Function 2). However, different logic is used in each of these Functions.

4. SGIS

The SGIS is required to be OPERABLE in MODEs 1, 2, and 3 except when all associated valves are closed and de-activated. De-activated means valve operating power is removed.

a. Steam Generator Pressure-Low Trip

This LCO requires four sensor channels of SGIS Steam Generator Pressure-Low trip for each steam generator to be OPERABLE in MODEs 1, 2, and 3.

The Allowable Value is set below the full load operating value for steam pressure so as not to interfere with normal plant operation. However, the setting is high enough to provide the required protection for excessive steam demand. An excessive steam demand causes the RCS to cool down, resulting in a positive reactivity addition to the core. An SGIS is required to prevent the excessive cooldown.

This Function may be manually blocked when steam generator pressure is reduced during controlled plant cooldowns. The block is permitted below 785 psia, and block permissive responses are annunciated in the Control Room. This allows a controlled depressurization of the secondary system, while maintaining administrative control of ESF protection. From a blocked condition, the block will be removed automatically as steam generator pressure increases above 785 psia, as sensed by two of the four sensor channels, in accordance with the block philosophy of removing blocks when the enabling conditions are no longer satisfied.

This LCO requires four channels per steam generator of the automatic block removal for SGIS Steam Generator Pressure-Low trip to be OPERABLE in MODEs 1, 2, and 3.

The automatic block removal features consist of four sensor channels and two actuation logic channels. This LCO applies to failures in the four sensor channels, including measurement channels and sensor block modules. Failures in the actuation logic channels, including the manual bypass key switches, are considered actuation logic failures and are addressed in LCO 3.3.5.

This LCO applies to the automatic block removal feature only. If the block enable Function is failed so as to prevent entering a block condition, operation may continue.

The block permissive is set low enough so as not to be enabled during normal plant operation, but high enough to allow blocking prior to reaching the trip setpoint.

5. <u>RAS for the Containment Sump</u>

   a. <u>RWT Level-Low Trip</u>

   This LCO requires four sensor channels of RWT Level-Low trip to be OPERABLE in MODEs 1, 2, and 3. The signal provided is a level indication from a level switch, not an analog signal.

   The upper limit on the Allowable Value for this trip is set low enough to ensure RAS does not actuate before sufficient water is transferred to the containment sump. Premature recirculation could impair the reactivity control Function of safety injection by limiting the amount of boron injection. Premature recirculation could also damage or disable the recirculation system if recirculation begins before the sump has enough water to prevent air entrainment in the suction. The lower limit on the RWT Level-Low trip Allowable Value is high enough to transfer suction to the containment sump prior to emptying the RWT.

6. <u>AFAS Signal</u>

   The AFAS logic actuates AFW to a steam generator on low level in that generator unless it has been identified as being ruptured.

   A low level in either generator, as sensed by a two-out-of-four coincidence of four wide range sensors for any generator, will generate an AFAS start signal, which starts both trains of AFW pumps, operates other equipment, and feeds both steam generators. The AFAS also monitors the secondary differential pressure in both steam generators and actuates an AFAS block signal to a ruptured generator, if the pressure in that generator is lower than that in the other generator by the differential pressure setpoint.

a. Steam Generator 1/2 Level-Low Trip

This LCO requires four sensor channels for each steam generator of Steam Generator Level-Low trip to be OPERABLE in MODEs 1, 2, and 3.

The Allowable Value ensures adequate time exists to initiate AFW, while the steam generators can function as a heat sink.

b. Steam Generator Pressure Difference-High Trip (SG-1 > SG-2) or (SG-1 > SG-2)

This LCO requires four sensor channels per steam generator of Steam Generator Pressure Difference-High trip to be OPERABLE in MODEs 1, 2, and 3.

The Allowable Value for this trip is high enough to allow for small pressure differences and normal instrumentation errors between the steam generator channels during normal operation without an actuation. The setting is low enough to detect and block feeding of a ruptured steam generator in the event of an MSLB or FWLB, while permitting the feeding of the intact steam generator.

APPLICABILITY

All ESFAS Functions are required to be OPERABLE in MODEs 1, 2, and 3. In MODEs 1, 2, and 3, there is sufficient energy in the primary and secondary systems to warrant automatic ESF system responses to:

- Close the MSIVs to preclude a positive reactivity addition;
- Actuate AFW to preclude the loss of the steam generators as a heat sink (in the event the normal feedwater system is not available);

- Actuate ESF systems to prevent or limit the release of fission product radioactivity to the environment by isolating Containment and limiting the containment pressure from exceeding the containment design pressure during a design basis LOCA or other DBAs; and
- Actuate ESF systems to ensure sufficient borated inventory to permit adequate core cooling and reactivity control during a design basis LOCA or other DBAs.

In MODEs 4, 5, and 6, automatic actuation of ESFAS Functions is not required because adequate time is available for plant operators to evaluate plant conditions and respond by manually operating the ESF components, if required, as addressed by LCO 3.3.5. In LCO 3.3.5, manual capability is required for Functions other than AFAS in MODE 4, even though automatic actuation is not required. Because of the large number of components actuated on each ESFAS, actuation is simplified by the use of the manual actuation push buttons. Manual start of AFAS is not required in MODE 4 because AFW or shutdown cooling will already be in operation or available in this MODE.

The ESFAS actuation logic must be OPERABLE in the same MODEs as the automatic and manual actuation. In MODE 4, only the portion of the ESFAS logic responsible for the required manual actuation must be OPERABLE.

In MODEs 5 and 6, ESFAS actuated systems are either reconfigured or disabled for shutdown cooling operation. Accidents in these MODEs are slow to develop and would be mitigated by manual operation of individual components.

The most common cause of sensor channel inoperability is outright failure or drift of the sensor module or measurement channel sufficient to exceed the tolerance allowed by Reference 5.

Typically, the drift is small which, at worst, results in a delay of actuation rather than a total loss of Function. Determination of setpoint drift is generally made during the performance of a CHANNEL CALIBRATION when the process instrument is set up for adjustment to bring it to within specification. Sensor drift could also be identified during the CHANNEL CHECKS. CHANNEL FUNCTIONAL TESTS identify sensor module drift. If the actual trip setpoint is not within the Allowable Value in Table 3.3.4-1, the sensor channel is inoperable and the appropriate Condition(s) are entered.

In the event that either a sensor channel's trip setpoint is found nonconservative with respect to the Allowable Value in Table 3.3.4-1, or the sensor, instrument loop, signal processing electronics, ESFAS sensor module or applicable automatic block removal feature when block is in effect is found inoperable, all affected Functions provided by that sensor channel must be declared inoperable and the plant must enter the Condition statement for the particular protection Function affected.

When the number of inoperable sensor channels in an ESFAS Function exceeds those specified in any related Condition associated with the same ESFAS Function, the plant is outside the safety analysis. Therefore, LCO 3.0.3 should be immediately entered if applicable in the current MODE of operation.

A Note has been added to clarify the application of the Completion Time rules. The Conditions of this Specification may be entered independently for each Function in Table 3.3.4-1. Completion Times for the inoperable channel of a Function will be tracked separately.

<u>A.1, A.2.1, and A.2.2</u>

Condition A applies to the failure of a single channel (measurement channel or sensor module) of one or more input parameters in the following ESFAS Functions:

1. <u>SIAS</u>

   Containment Pressure-High Trip
   Pressurizer Pressure-Low Trip

2. <u>CSAS</u>

   Containment Pressure-High Trip

3. CIS

   Containment Pressure-High Trip

4. SGIS

   Steam Generator Pressure-Low Trip

5. RAS for the Containment Sump

   RWT Level-Low Trip

6. AFAS Signal

   Steam Generator Level-Low Trip
   Steam Generator Pressure Difference-High Trip

Engineered Safety Features Actuation System coincidence logic is normally two-out-of-four. If one ESFAS sensor channel is inoperable, startup or power operation is allowed to continue as long as action is taken to restore the design level of redundancy.

If one ESFAS sensor channel is inoperable, startup or power operation is allowed to continue, providing the inoperable channel is placed in bypass or trip within 1 hour (Required Action A.1).

The Completion Time of 1 hour allotted to bypass or trip the sensor channel is sufficient to allow the operator to take all appropriate actions for the failed channel, and still ensures that the risk involved in operating with the failed channel is acceptable.

One failed sensor channel is restored to OPERABLE status or is placed in trip within 48 hours (Required Action A.2.1 or A.2.2). Required Action A.2.1 restores the full capability of the function. Required Action A.2.2 places the function in a one-out-of-three configuration. In this configuration, common cause failure of the dependent channel cannot prevent ESFAS actuation. The 48-hour Completion Time is based upon operating experience, which has demonstrated that a random failure of a second channel occurring during the 48-hour period is a low probability event.

B.1 and B.2

Condition B applies to the failure of two sensor channels in any of the following ESFAS functions:

1. SIAS

   Containment Pressure-High Trip
   Pressurizer Pressure-Low Trip

2. CSAS

   Containment Pressure-High Trip

3. CIS

   Containment Pressure-High Trip

4. SGIS

   Steam Generator Pressure-Low Trip

5. RAS for the Containment Sump

   RWT Level-Low Trip

6. AFAS Signal

   Steam Generator Level-Low Trip
   Steam Generator Pressure Difference-High Trip

With two inoperable sensor channels, one channel should be placed in bypass, and the other channel should be placed in trip within the 1-hour Completion Time. With one channel of protective instrumentation bypassed, the ESFAS Function is in two-out-of-three logic; but with another channel failed, the ESFAS may be operating with a two-out-of-two logic. This is outside the assumptions made in the analyses and should be corrected. To correct the problem, the second channel is placed in trip. This places the ESFAS in a one-out-of-two logic. If any of the other OPERABLE channels receive a trip signal, ESFAS actuation will occur.

One of the failed sensor channels should be restored to OPERABLE status within 48 hours. After one channel is restored to OPERABLE status, the provisions of Condition A still apply to the remaining inoperable channel. Therefore, the channel that is still inoperable after completion of Required Action B.2 must be placed in trip if more than 48 hours has elapsed since the initial channel failure.

The Required Action is modified by a Note stating that LCO 3.0.4 is not applicable. The Note was added to allow the changing of MODEs, even though two sensor channels are inoperable, with one channel bypassed and one tripped. MODE changes in this configuration are allowed, to permit maintenance and testing on one of the inoperable channels. In this configuration, the protective system is in a one-out-of-two logic, and the probability of a common cause failure affecting both of the OPERABLE channels during the 48 hours permitted is remote.

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

Condition C applies to the failure of one automatic block removal feature when the block is in effect.

The automatic block removal features are incorporated into the four sensor block modules (per steam generator for SGIS) and two block logic modules. Condition C applies to failures in the automatic block removal feature of one of the four sensor block modules. Failures in the block logic modules, including the block logic manual bypass key switches, are considered actuation logic failures and are addressed in LCO 3.3.5.

In Condition C, it is permissible to continue operation with the automatic block removal feature in one sensor block module failed, providing the sensor block module is disabled (Required Action C.1). This can be accomplished by adjusting the sensor block module setpoint, which disables the sensor block modules to both block logic modules. Therefore, a block permissive signal is not produced by the sensor block module.

Placing a sensor module in bypass defeats the block permissive input in one of the four channels to the two-out-of-four block removal logic, placing the automatic block removal feature in one-out-of-three logic. Thus, any of the remaining three channels is capable of removing the block feature when the block enable conditions are no longer valid.

In this configuration, common cause failure of the dependent channel cannot prevent block removal.

D.1, D.2.1, and D.2.2

Condition D applies to two inoperable automatic block removal features. The automatic block removal features consist of four sensor block modules (per steam generator for SGIS) and two actuation logic channels. This Condition applies to failures in two of the four sensor block modules. With two of the four sensor block modules failed in a nonconservative direction (enabling the block feature), the automatic block removal feature is in two-out-of-two logic. Failures in the actuation logic channels, including the manual bypass key switches, are considered actuation logic failures and are addressed in LCO 3.3.5.

In Condition D, it is permissible to continue operation with two automatic block removal features failed, providing the sensor block modules are disabled in a similar manner as discussed for Condition C.

If the failed sensor block modules cannot be disabled, actions to address the inoperability of the affected sensor block modules must be taken. Required Action D.2.1 and Required Action D.2.2 are equivalent to the Required Actions for a two sensor channel failure (Condition B). Also similar to Condition B, after one inoperable sensor block module is restored, the provisions of Condition C still apply to the remaining inoperable automatic block removal feature, with the Completion Time measured from the point of the initial bypass channel failure. The 1-hour Completion Time minimizes the time that the plant is in two-out-of-two logic. The 48-hour Completion Time limits the time the plant is in one-out-of-two logic. Limits on the time in these logic conditions are similar to those found in Action B.

The Required Action is modified by a Note stating that LCO 3.0.4 is not applicable. The Note was added to allow

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the changing of MODEs, even though two automatic block removal features are inoperable, with one sensor block module bypassed and one disabled. MODE changes in this configuration are allowed, to permit maintenance and testing on one of the inoperable automatic block removal features. In this configuration, the protective system is in a one-out-of-two logic, and the probability of a common cause failure affecting both of the OPERABLE automatic block removal features during the 48 hours permitted is remote.

E.1 and E.2

If the Required Actions and associated Completion Times of Condition A, B, C, or D 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 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.

SURVEILLANCE REQUIREMENTS

The SRs for any particular ESFAS Function are found in the SRs column of Table 3.3.4-1 for that Function. Most Functions are subject to CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, CHANNEL CALIBRATION, and response time testing.

SR 3.3.4.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one sensor channel to a similar parameter on other sensor channels. It is based on the assumption that sensor channels monitoring the same parameter should read approximately the same value. Significant deviations between sensor channels could be an indication of excessive sensor channel 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 the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff based on a qualitative assessment of the sensor channel, which considers sensor channel 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. If the channels are within the criteria, it is an indication that the channels are OPERABLE. If the channels are normally off-scale during times when surveillance testing is required, the CHANNEL CHECK will only verify that they are off-scale in the same direction. Off-scale low current loop channels are verified to be reading at the bottom of the range and not failed down-scale.

The Frequency of about once every shift is based on operating experience that demonstrates sensor channel failure is rare. Since the probability of two random failures in redundant channels in any 12-hour period is extremely low, the CHANNEL CHECK minimizes the chance of loss of ESFAS Function due to failure of redundant channels. The CHANNEL CHECK supplements less formal, but more frequent, checks of the channel during normal operational use of displays.

SR 3.3.4.2

A CHANNEL FUNCTIONAL TEST is performed every 92 days to ensure the entire sensor channel will perform its intended function when needed.

The CHANNEL FUNCTIONAL TEST tests the individual sensor channels using an analog or level switch test input to each bistable.

A test signal is substituted for the input in one sensor channel at a time to verify that the bistable trips within the specified tolerance around the setpoint. Any setpoint adjustment shall be consistent with the assumptions of the Reference 5.

SR 3.3.4.3

Surveillance Requirement 3.3.4.3 is a CHANNEL FUNCTIONAL TEST similar to SR 3.3.4.2, except 3.3.4.3 is performed every 24 months and is only applicable to automatic block removal features of the sensor block modules. These include the Pressurizer Pressure-Low trip block and the SGIS Steam Generator Pressure-Low trip block.

The CHANNEL FUNCTIONAL TEST for proper operation of the automatic block removal features is critical during plant heatups because the blocks may be in place prior to entering MODE 3, but must be removed at the appropriate points during plant startup to enable the ESFAS Function. A 24-month SR Frequency is adequate to ensure proper automatic block removal module operation as described in Reference 3. Once the blocks are removed, the blocks must not fail in such a way that the associated ESFAS Function is inappropriately blocked. This feature is verified by the appropriate ESFAS Function CHANNEL FUNCTIONAL TEST.

The 24-month SR Frequency is adequate to ensure proper automatic block removal feature operation as described in Reference 3.

<u>SR 3.3.4.4</u>

CHANNEL CALIBRATION is a check of the sensor channel, including the automatic block removal feature of the sensor block module and the sensor. The SR verifies that the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for sensor channel drift between successive calibrations to ensure that the channel remains operational between successive surveillance tests. CHANNEL CALIBRATIONS must be performed consistent with Reference 5.

The as-found and as-left values must also be recorded and reviewed for consistency with the assumptions of the extension analysis. The requirements for this review are outlined in Reference 6.

The Frequency is based upon the assumption of a 24-month calibration interval for the determination of the magnitude of equipment drift in the setpoint analysis.

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SR 3.3.4.5

This SR ensures that the train actuation response times are the maximum values assumed in the safety analyses. Individual component response times are not modeled in the analyses. The analysis models 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 are rated discharge pressure, valves in full open or closed position). Response time testing acceptance criteria are included in Reference 1, Section 7.3. The test may be performed in one measurement or in overlapping segments, which verification that all components are measured.

Engineered Safety Feature Response Time tests are conducted on a STAGGERED TEST BASIS of once every 24 months. This results in the interval between successive tests of a given channel of n x 24 months, where n is the number of channels in the Function. Surveillance of the final actuation devices, which make up the bulk of the response time, is included in the testing of each channel. Therefore, staggered testing results in Response Time verification of these devices every 24 months. The 24-month STAGGERED TEST BASIS Frequency is based upon plant operating experience, which shows that random failures of instrumentation components causing serious response time degradation, but not channel failure, are infrequent occurrences.

REFERENCES

1. UFSAR
2. IEEE No. 279, "Proposed IEEE Criteria for Nuclear Power Plant Protection Systems," August 1968
3. Letter from Mr. R. E. Denton (BGE) to NRC Document Control Desk, dated June 5, 1995, "Response to NRC Request for Review & Comment on Review of Preliminary Accident Precursor Analysis of Trip; Loss of 13.8 kV Bus; Short-Term Saltwater Cooling System Unavailability, CCNPP Unit 2"
4. 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants"
5. CCNPP Setpoint File

6. Calvert Cliffs Procedure EN-4-104, "Surveillance Testing"

B 3.3 INSTRUMENTATION

B 3.3.5 Engineered Safety Features Actuation System (ESFAS) Logic and Manual Actuation

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BACKGROUND

The ESFAS initiates necessary safety systems, based upon the values of selected unit parameters to mitigate accidents in order to protect the public and plant personnel from the accidental release of radioactive fission products.

The ESFAS contains devices and circuitry that generate the following signals when the monitored variables reach levels that are indicative of conditions requiring protective action:

1. SIAS;
2. CSAS;
3. CIS;
4. SGIS;
5. RAS for the Containment Sump; and
6. AFAS.

Equipment actuated by each of the above signals is identified in Reference 1.

Each of the above ESFAS actuation systems is segmented into four sensor channels addressed by LCO 3.3.4 and two actuation subsystems addressed by this LCO. Each sensor subsystem includes measurement channels and bistables (sensor modules). The SIAS actuation logic channels include two sets of logic circuitry (actuation logic modules) and actuation relay equipment. The actuation logic channels actuate ESFAS equipment trains that are sequentially loaded on the DGs.

Each of the four sensor modules monitors redundant and independent process measurement channels. Each sensor is monitored by at least one bistable. The bistable associated with each ESFAS sensor channel will trip when the monitored variable exceeds the trip setpoint. When tripped, the sensor channels provide outputs to the two actuation logic channels.

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The two independent actuation logic channels each compare the four associated sensor channel outputs. If a trip occurs in two or more sensor channels, the two-out-of-four logic in each actuation logic channel will actuate the associated train of ESFAS. Each has sufficient equipment to provide protection to the public in the case of a DBA. The sensor logic channel is addressed in LCO 3.3.4. This LCO addresses the actuation logic channel.

Each of the four sensor channels is mounted in a separate cabinet, excluding the sensors and field wiring.

The role of the sensor channel (measurement channels and sensor module) and sensor block module is discussed in LCO 3.3.4. That of the actuation logic channel is discussed below.

ESFAS Logic

The two independent actuation logic channels compare the four sensor channel outputs. If a trip occurs in the same parameter in two or more sensor channels, the two-out-of-four logic in each actuation logic channel initiates one train of ESFAS. Either train actuates sufficient redundant and independent equipment.

Each actuation logic channel is housed in two cabinets. One cabinet contains the logic circuitry (actuation logic modules) for the actuation logic channel, while the other cabinet contains the actuation relay equipment. This actuation relay equipment includes the actuation relays that actuate the ESFAS equipment in response to a signal from the actuation logic channels.

It is possible to change the two-out-of-four ESFAS logic to a two-out-of-three logic for a given input parameter in one sensor channel at a time by blocking one channel input to the logic. Thus, the actuation logic modules will function normally, producing normal trip indication and annunciation, but ESFAS actuation will not occur since the blocked channel is effectively removed from the coincidence logic. Maintenance bypassing can be simultaneously performed on any number of parameters in any number of Functions, providing each parameter is bypassed in only one sensor channel per Function at a time.

Maintenance bypassing is normally employed during maintenance or testing.

In addition to the maintenance bypasses, there are operating bypasses (blocks) on the Pressurizer Pressure-Low trip input to the SIAS and on the Steam Generator Pressure-Low trip input to the SGIS when these inputs are no longer required for protection. These blocks are enabled manually when the enabling conditions are satisfied in three of the four sensor block modules. The block circuitry employs four sensor block modules, sensing pressurizer pressure (for the SIAS) and steam generator pressure (for the SGIS). These sensor block modules provide contact output to the three-out-of-four logic in the two block logic modules. When the logic is satisfied, manual blocking is permitted. There are two manual block controls for each Function, one per actuation logic channel.

The block logic modules provide one of the signals to a logic circuit that senses the actuation logic output and the output from the block logic module. When block logic is met, and the block is manually enabled, the logic circuit receiving signals from both logic sources prevents an actuation signal from being sent to the ESFAS equipment.

All blocks are automatically removed when enabling block conditions are no longer satisfied.

Manual ESFAS actuation capability is provided to permit the operator to manually actuate an ESF system when necessary. Two push buttons are provided in the Control Room for each ESFAS Function, except SGIS and AFAS. Manual AFAS start capability is provided in the Control Room. Steam generator isolation signal manual actuation requires operation of MSIV handswitches and feedwater header isolation handswitches. Each push button actuates one equipment train via the ESFAS logic.

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The actuation logic is tested by inserting a local test signal. A coincidence logic trip will occur if there is the simultaneous presence of a sensor channel trip, either legitimate or due to testing, and block logic allows. The automatic block removal feature of block logic modules is tested with the actuation logic modules. Most ESFAS Functions employ several separate parallel two-out-of-four actuation logic module subchannels, with each subchannel actuating a subset of the ESFAS equipment associated with that Function. Each of these subchannels can be tested individually so that simultaneous actuation of an entire train can be avoided during testing.

Except in the case of actuation subchannels SIAS Nos. 5 and 10, CIS No. 5, CSAS No. 3, and SGIS No. 1, all actuation logic channels can be tested at power. The above designated subchannels must be tested when shut down because they actuate the following equipment, which cannot be actuated at power:

- Reactor coolant pump (RCP) seal bleedoff isolation valves;
- Service water isolation valves;
- Volume control tank discharge valves;
- Letdown stop valves;
- Component cooling to RCPs;
- Component Cooling from RCPs;
- MSIVs;
- Feedwater isolation valves;
- Instrument air containment isolation valves (CIVs);
- Heater drain pumps;
- Main Feedwater Pump; and
- Condensate Booster Pumps.

APPLICABLE SAFETY ANALYSES

Most 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. Functions such as manual actuation, not specifically credited in the accident analysis, serve as backups to Functions and are part of the NRC staff-approved licensing basis for the plant.

Engineered Safety Features Actuation System protective Functions are as follows:

1. SIAS

   The SIAS ensures acceptable consequences during LOCA events, including steam generator tube rupture, and other DBAs. To provide the required protection, either a high containment pressure or a low pressurizer pressure signal will actuate SIAS. Safety injection actuation signal actuates the Emergency Core Cooling System (ECCS) and performs several other Functions, such as starting the DGs.

2. CSAS

   The CSAS actuates containment spray, preventing containment overpressurization during a LOCA or MSLB. Both a high containment pressure signal and a SIAS have to actuate to provide the required protection. This configuration reduces the likelihood of inadvertent containment spray.

3. CIS

   The CIS actuates the Containment Isolation System, ensuring acceptable consequences during LOCAs and other DBAs (inside Containment). A high containment pressure signal will actuate CIS.

4. SGIS

   The SGIS ensures acceptable consequences during an excessive loss of steam from the Main Steam System by isolating both steam generators if either generator indicates a low steam generator pressure. The SGIS, concurrent with or following a reactor trip, minimizes the rate of heat extraction and subsequent cooldown of the RCS during these events.

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5. RAS

At the end of the injection phase of a LOCA, the refueling water tank (RWT) 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. Switchover from RWT to containment sump must occur before the RWT empties to prevent damage to the ECCS 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 pump suction. Furthermore, early switchover must not occur so sufficient borated water is injected from the RWT to ensure the reactor remains shut down in the recirculation mode. An RWT Level-Low trip signal generated by a level switch actuates the RAS.

6. AFAS Signal

An AFAS signal actuates feedwater flow to both steam generators if a low level is indicated in either steam generator, unless the generator is ruptured.

The AFAS maintains a steam generator heat sink during the following events:

- MSLB;
- FWLB; and
- Loss of feedwater.

A low steam generator water level signal will actuate auxiliary feed to both steam generators.

Secondary steam generator (SG) differential pressure signal, (SG-1 > SG-2) or (SG-2 > SG-1), blocks auxiliary feed to a ruptured steam generator. This input to the AFAS logic prevents loss of the intact generator while preventing feeding a ruptured generator during MSLBs and FWLBs. This prevents containment overpressurization and/or excessive RCS cooldown during these events.

BASES

The ESFAS satisfies 10 CFR 50.36(c)(2)(ii), Criterion 3.

LCO

The LCO requires that all components necessary to provide an ESFAS actuation be OPERABLE.

Actions allow maintenance bypass of individual sensor channels. Plants are in a maintenance bypass condition before either restoring the Function to four channel operation (two-out-of-four logic) or placing the channel in trip (one-out-of-three logic).

The Bases for the LCO on ESFAS automatic actuation Functions are addressed in the Bases for LCO 3.3.4. Those associated with the manual actuation or actuation logic are addressed below.

1. <u>SIAS</u>

   a. <u>Manual Actuation</u>

      This LCO requires two channels of SIAS manual actuation to be OPERABLE in MODEs 1, 2, 3, and 4.

   b. <u>Actuation Logic</u>

      This LCO requires two channels of SIAS actuation logic to be OPERABLE in MODEs 1, 2, and 3.

      Failures in the actuation logic channels, including the manual bypass key switches, are actuation logic failures and are addressed in this LCO.

      Actuation logic consists of all circuitry housed within the actuation logic channels, including the actuating relay contacts responsible for actuating the ESF equipment.

2. <u>CSAS</u>

   The CSAS is actuated either manually or automatically. It is also necessary to have an automatic or manual SIAS for a complete actuation. The CSAS opens the containment spray valves, whereas the SIAS actuates other required components. The SIAS requirement should always be satisfied on a legitimate CSAS, since the Containment Pressure-High trip analytical setpoint used in the SIAS is the same analytical setpoint used in the CSAS. The transmitters used to actuate CSAS are independent of those used in the SIAS to prevent inadvertent containment spray due to failures in two sensor channels.

a. Manual Actuation

This LCO requires two channels of CSAS manual actuation to be OPERABLE in MODEs 1, 2, 3, and 4.

b. Actuation Logic

This LCO requires two channels of CSAS actuation logic to be OPERABLE in MODEs 1, 2, and 3.

Actuation logic consists of all circuitry housed within the actuation logic channels, including the actuating relay contacts responsible for actuating the ESF equipment.

3. CIS

a. Manual Actuation

This LCO requires two channels of CIS manual actuation to be OPERABLE in MODEs 1, 2, 3, and 4.

b. Actuation Logic

This LCO requires two channels of actuation logic for CIS to be OPERABLE in MODEs 1, 2, and 3.

Actuation logic consists of all circuitry housed within the actuation logic channels, including the actuating relay contacts responsible for actuating the ESF equipment.

4. SGIS

   a. Manual Actuation

      This LCO requires one channel per MSIV of the SGIS manual actuation to be OPERABLE in MODEs 1, 2, 3, and 4.

   b. Actuation Logic

      This LCO requires two channels of SGIS actuation logic to be OPERABLE in MODEs 1, 2, and 3.

      Failures in the actuation logic channels, including the manual bypass key switches, are considered actuation logic failures and are addressed in the logic LCO.

5. RAS

   a. Manual Actuation

      This LCO requires two channels of RAS manual actuation to be OPERABLE in MODEs 1, 2, 3, and 4.

   b. Actuation Logic

      This LCO requires two channels of RAS actuation logic to be OPERABLE in MODEs 1, 2, and 3.

6. AFAS Signal

   A low level in either generator, as sensed by a two-out-of-four coincidence of four wide range sensors for each generator, will generate an AFAS signal, which starts both trains of AFW pumps and feeds both steam generators. The AFAS also monitors the secondary differential pressure in both steam generators and actuates an AFAS block signal to a ruptured generator if the pressure in that generator is lower than the other generator by the differential pressure setpoint (Reference 2).

   a. Manual Actuation

      This LCO requires two channels of AFAS manual actuation start to be OPERABLE in MODEs 1, 2, and 3.

BASES

b. Actuation Logic

This LCO requires two channels of AFAS actuation logic to be OPERABLE in MODEs 1, 2, and 3.

Actuation logic consists of all circuitry housed within the actuation logic channels, including the actuating relay contacts responsible for actuating the ESF equipment.

APPLICABILITY

All ESFAS Functions are required to be OPERABLE in MODEs 1, 2, and 3. In MODEs 1, 2, and 3, there is sufficient energy in the primary and secondary systems to warrant automatic ESF system responses to:

- Close the MSIVs to limit a positive reactivity addition;
- Actuate AFW to preclude the loss of the steam generators as a heat sink (in the event the normal feedwater system is not available);
- Actuate ESF systems to prevent or limit the release of fission product radioactivity to the environment by isolating Containment and limiting the containment pressure from exceeding the containment design pressure during a design basis LOCA or other DBAs; and
- Actuate ESF systems to ensure sufficient borated inventory to permit adequate core cooling and reactivity control during a design basis LOCA or MSLB accident.

In MODEs 4, 5, and 6, automatic actuation of ESFAS Functions is not required, because adequate time is available for plant operators to evaluate plant conditions and respond by manually operating the ESF components if required. Engineered Safety Features Actuation System manual actuation capability is required for Functions other than AFAS in MODE 4, even though automatic actuation is not required. Because of the large number of components actuated on each ESFAS, actuation is simplified by the use of the manual actuation push buttons. Manual actuation of AFAS is not required in MODE 4 because AFW or shutdown cooling will already be in operation or available.

The ESFAS actuation logic must be OPERABLE in the same MODEs as the automatic and manual actuations. In MODE 4, only the portion of the ESFAS logic responsible for the required manual actuation must be OPERABLE.

In MODEs 5 and 6, ESFAS actuated systems are either reconfigured or disabled for shutdown cooling operation. Accidents in these MODEs are slow to develop and would be mitigated by manual operation of individual components.

ACTIONS

When the number of inoperable actuation logic or manual actuation channels in an ESFAS Function exceeds those specified in any related Condition associated with the same ESFAS Function, the plant is outside the safety analysis. Therefore, LCO 3.0.3 should be immediately entered.

A Note has been added to the ACTIONS to clarify the application of the Completion Time rules. The Conditions of this Specification may be entered independently for each Function in Table 3.3.5-1 in the LCO. Completion Times for the inoperable actuation logic channel of a Function will be tracked separately.

A.1

Condition A applies to one AFAS manual actuation or AFAS actuation logic channel inoperable. It is identical to Condition C for the other ESFAS Functions, except for the shutdown track imposed by Condition D.

The channel must be restored to OPERABLE status to restore redundancy of the AFAS Function. The 48-hour Completion Time is commensurate with the importance of avoiding the vulnerability of a single failure in the only remaining OPERABLE channel.

B.1 and B.2

If the Required Action and associated Completion Time of Condition A cannot be met, the reactor should 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.

C.1

Condition C applies to one manual actuation or actuation logic channel inoperable for those ESFAS Functions that must be OPERABLE in MODEs 1, 2, 3, and 4 (manual actuation) or MODEs 1, 2, and 3 (actuation logic channel). Actuation logic includes the block logic modules when the affected block is in effect. The shutdown track imposed by Condition D or E requires entry into MODE 4 or 5, respectively, where the LCO does not apply to the affected Functions.

The channel must be restored to OPERABLE status to restore redundancy of the affected Functions. The 48-hour Completion Time is commensurate with the importance of avoiding the vulnerability of a single failure in the only remaining OPERABLE channel.

D.1 and D.2

Condition D is entered when the Required Action and associated Completion Time of Condition C are not met for one manual actuation channel. If Required Action C.1 for one manual actuation channel cannot be met within the required Completion Time, the plant must be brought to a mode in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours 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.

E.1 and E.2

Condition E is entered when the Required Action and associated Completion Time of Condition C are not met for one actuation logic channel. If Required Action C.1 for one actuation logic channel cannot be met within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours 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.

SURVEILLANCE REQUIREMENTS

SR 3.3.5.1

A CHANNEL FUNCTIONAL TEST is performed every 92 days to ensure the entire actuation logic channel will perform its intended function when needed. Sensor channel tests are addressed in LCO 3.3.4. This SR addresses actuation logic tests.

Actuation Logic Tests

Actuation logic channel testing includes injecting one actuation signal into each two-out-of-four logic actuation modules in each ESFAS Function, and using a bistable trip input to satisfy the actuation logic. Testing includes block logic modules.

Note 1 requires that actuation logic tests include operation of actuation relays. Note 2 allows deferred at power testing of certain subchannel relays to allow for the fact that operating certain relays during power operation could cause plant transients or equipment damage. Those subchannel relays that cannot be tested at power must be tested in accordance with Note 2. These include SIAS No. 5, SIAS No. 10, CIS No. 5, SGIS No. 1, and CSAS No. 3.

These subchannel relays actuate the following components, which cannot be tested at power:

- RCP seal bleedoff isolation valves;
- Service water isolation valves;
- Volume control tank discharge valves;
- Letdown stop valves;
- Component Cooling to and from the RCPs;
- MSIVs and feedwater isolation valves;

BASES

- Instrument air CIVs;
- Heater drain pumps;
- Main feedwater pumps; and
- Condensate booster pumps.

The reasons each of the above cannot be fully tested at power are stated in Reference 1.

Actuation logic tests verify that the ESFAS is capable of performing its intended function, from bistable input through the actuated components.

The Frequency of 92 days is based on operating experience that has shown these components usually pass the surveillance test when performed at this Frequency.

SR 3.3.5.2

A CHANNEL FUNCTIONAL TEST is performed on the manual ESFAS actuation circuitry, de-energizing relays and providing manual actuation of the Function.

This surveillance test verifies that the actuation push buttons are capable of opening contacts in the actuation logic as designed, de-energizing the actuation relays and providing manual trip of the Function. The 24-month Frequency is based on the need to perform this surveillance test under the conditions that apply during a plant outage, and the potential for an unplanned transient if the test were to be performed with the reactor at power. Operating experience has shown these components usually pass the surveillance test when performed at a Frequency of once every 24 months.

BASES

REFERENCES	1.	UFSAR, Section 7.3, "Engineered Safety Features Actuation Systems"
	2.	Letter from Mr. R. E. Denton (BGE) to NRC Document Control Desk, dated June 5, 1995, "Response to NRC Request for Review & Comment on Review of Preliminary Accident Precursor Analysis of Trip; Loss of 13.8 kV Bus; Short-Term Saltwater Cooling System Unavailability, CCNPP Unit 2"

B 3.3 INSTRUMENTATION

B 3.3.6 Diesel Generator (DG)-Loss of Voltage Start (LOVS)

BASES

BACKGROUND

The DGs provide a source of emergency power, when offsite power is unavailable, to allow safe plant operation. Undervoltage protection will generate a Loss of Voltage Start (LOVS) signal in the event a loss of voltage or degraded voltage condition occurs. There are three LOVS Functions; loss of voltage, transient degraded voltage, and steady state degraded voltage, for each 4.16 kV emergency bus.

Each of the redundant 4 kV emergency buses is equipped with two sets of two undervoltage relays. Each of the four redundant and independent undervoltage relays is comprised of three sensing elements. The first element of the four relays is set to provide a two-out-of-four undervoltage signal upon a loss of bus voltage. The second element of the four relays is set to provide a two-out-of-four transient undervoltage signal on 4 kV emergency bus undervoltage. The third element of the four relays provides a two-out-of-four steady state undervoltage signal on a sustained 4 kV emergency bus undervoltage condition.

Settings and Tolerances

The settings and tolerances are based on the analytical limits presented in Reference 1. The selection of these settings is such that adequate protection is provided when all test equipment time delays are taken into account. The transient and steady state undervoltage setpoints ensure that the safety-related motors relied upon for accident mitigation are provided with a minimum of 75% and 90% of their rated voltage, respectively. The setting specified in SR 3.3.6.3 allows for calibration tolerances, potential transformer correction factors, test equipment uncertainties, and relay drift. A detailed description of the methodology used to calculate the settings is provided in Reference 2. The nominal setting accounts for factors described above, plus additional margin to the analytical limit. If the measured setting does not exceed the documented surveillance trip acceptance criteria, the undervoltage relay is considered OPERABLE.

BASES

Settings will ensure that the consequences of accidents will be acceptable, providing the plant is operated from within the LCOs at the onset of the accident and the equipment functions as designed. A setting is the desired characteristic, obtained as a result of having set a device, stated in terms of calibration markings or of actual performance benchmarks, such as pickup current and operating time at a given value of input. The term setpoint applies to instruments, and since protective relays discussed here are not instruments as discussed in other LCOs, the term setpoint does not apply here.

The undervoltage detection scheme has been designed to sense a degraded (transient or steady state) or total loss of voltage at the 4 kV safety buses. The LOVS is sensed by 2-second delay bistables in the ESFAS undervoltage sensor logic module. A complete loss of offsite power will result in approximately a 2-second delay in LOVS actuation. The DG starts and is available to accept loads within a 10 second time interval after the ESFAS receives a LOVS. Emergency power is established within the maximum time delay assumed for each event analyzed in the accident analysis (Reference 1).

Sensor channels, measurement channels, sensor modules and actuation logic are described in the Background for B 3.3.4.

Since there are four protective channels in a two-out-of-four logic for each division of the 4.16 kV power system, no single failure will prevent protective system actuation. This arrangement meets Reference 3 criteria.

APPLICABLE SAFETY ANALYSES

The DG-LOVS is required for ESF systems to function in any accident with a loss of offsite power. Its design basis is that of the ESFAS.

Accident analyses credit the loading of the DG based on a loss of offsite power during a LOCA. The actual DG start has historically been associated with the ESFAS actuation. The diesel 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 analysis assumes a nonmechanistic DG loading, which does not explicitly account for each individual component of the loss of power detection and subsequent actions. This delay time includes contributions from the DG start, DG loading, and Safety Injection System component actuation. The response of the DG to a loss of power must be demonstrated to fall within this analysis response time when including the contributions of all portions of the delay.

The required channels of LOVS, in conjunction with the ESF systems powered from the DGs, provide plant protection in the event of any of the analyzed accidents discussed in Reference 1, Chapter 8 in which a loss of offsite power is assumed. Loss of voltage start channels are required to meet the redundancy and testability requirements of Reference 1, Appendix 1C.

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 include the appropriate DG loading and sequencing delay.

The DG-LOVS channels satisfy 10 CFR 50.36(c)(2)(ii), Criterion 3.

LCO

The LCO for the LOVS requires that four channels per bus of each LOVS instrumentation Function be OPERABLE in MODEs 1, 2, 3, and 4. The LOVS supports safety systems associated with the ESFAS.

Actions allow maintenance bypass of individual sensor channels. The plant is restricted to 48 hours in a maintenance bypass condition before either restoring the Function to four channel operation (two-out-of-four logic) or placing the channel in trip (one-out-of-three logic).

Loss of LOVS Function could result in the delay of safety system actuation when required. This could lead to unacceptable consequences during accidents. During the loss of offsite power, which is a AOO, the DG powers the motor-driven AFW pump. Failure of this pump to start would leave two turbine-driven pumps as well as an increased potential

BASES

for a loss of decay heat removal through the secondary system.

Only Allowable Values are specified for each Function in the LCO. Nominal trip settings are specified in the plant-specific procedures. The nominal settings are selected to ensure that the setting measured by CHANNEL FUNCTIONAL TESTS does not exceed the Allowable Value if the bistable is performing as required. Operation with a trip setting less conservative than the nominal trip setting, but within the Allowable Value, is acceptable, provided that operation and testing are consistent with the assumptions of the plant-specific setting calculation. A channel is inoperable if its actual trip setting is not within its required Allowable Value.

The Allowable Values and trip settings are established in order to start the DGs at the appropriate time, in response to plant conditions, in order to provide emergency power to start and supply the essential electrical loads necessary to safely shut down the plant and maintain it in a safe shutdown condition.

APPLICABILITY

The DG-LOVS actuation Function is required in MODEs 1, 2, 3, and 4 because ESF Functions are designed to provide protection in these MODEs.

ACTIONS

A LOVS sensor channel is inoperable when it does not satisfy the OPERABILITY criteria for the channel's Function. The most common cause of sensor channel inoperability is outright failure or drift of the bistable (sensor module) or measurement channel sufficient to exceed the tolerance allowed by the plant-specific setting analysis. Typically, the drift is found to be small and results in a delay of actuation rather than a total loss of function. Determination of setting drift is generally made during the performance of a CHANNEL CALIBRATION when the process instrument is set up for adjustment to bring it to within specification. Sensor drift could also be identified during the CHANNEL CHECKS. CHANNEL FUNCTIONAL TESTS identify sensor module drift. If the actual trip setting is not within the Allowable Value, the channel is inoperable and the appropriate Conditions must be entered.

In the event a sensor channel's setting is found to be nonconservative with respect to the Allowable Value, or the channel is found to be inoperable, then all affected Functions provided by that channel must be declared inoperable and the LCO Condition entered. The required channels are specified on a per DG basis.

When the number of inoperable channels in a Function exceeds those specified in any related Condition associated with the same Function, the plant is outside the safety analysis. Therefore, LCO 3.0.3 should be entered immediately if applicable in the current MODE of operation.

A Note has been added to the ACTIONS to clarify the application of Completion Time rules. The Conditions of this LCO may be entered independently 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, A.2.1, and A.2.2

Condition A applies if one sensor channel is inoperable for one or more Functions per DG bus.

If the channel cannot be restored to OPERABLE status, the affected channel should either be bypassed or tripped within 1 hour (Required Action A.1).

Placing this channel in either Condition ensures that logic is in a known configuration. In trip, the LOVS logic is one-out-of-three. In bypass, the LOVS logic is two-out-of-three. The 1-hour Completion Time is sufficient to perform these Required Actions.

Once Required Action A.1 has been complied with, Required Action A.2.1 allows 48 hours to repair the inoperable sensor channel. If the channel cannot be restored to OPERABLE status, it must be tripped in accordance with Required Action A.2.2. The time allowed to repair or trip the channel is reasonable to repair the affected channel while ensuring that the risk involved in operating with the inoperable channel is acceptable. The 48-hour Completion Time is based upon operating experience, which has demonstrated that a random failure of a second channel is a rare event during any given 48-hour period.

B.1, B.2.1, and B.2.2

Condition B applies if two sensor channels are inoperable for one or more Functions per DG.

The Required Action is modified by a Note stating that LCO 3.0.4 is not applicable. The Note was added to allow the changing of MODEs, even though two sensor channels are inoperable, with one channel bypassed and one tripped. In this configuration, the protective system is in a one-out-of-two logic, which is adequate to ensure that no random failure will prevent protective system operation.

Restoring at least one channel to OPERABLE status is the preferred action. If the channel cannot be restored to OPERABLE status within 1 hour, the Conditions and Required Actions for the associated DG made inoperable by DG-LOVS instrumentation are required to be entered. Alternatively, one affected channel is required to be bypassed and the other is tripped, in accordance with Required Action B.2.1. This places the Function in one-out-of-two logic. The 1-hour Completion Time is sufficient to perform the Required Actions.

Once Required Action B.2.1 has been complied with, Required Action B.2.2 allows 48 hours to repair the bypassed or inoperable channel.

After one channel is restored to OPERABLE status, the provisions of Condition A still apply to the remaining inoperable channel. Therefore, the channel that is still inoperable after completion of Required Action B.2.2 shall be placed in trip if more than 48 hours have elapsed since the initial channel failure.

C.1

Condition C applies when more than two undervoltage or degraded (transient or steady state) voltage sensor channels on a single bus are inoperable.

Required Action C.1 requires all but two channels to be restored to OPERABLE status within 1 hour. With more than two channels inoperable, the logic is not capable of providing a DG-LOVS signal for valid loss of voltage or degraded voltage conditions. The 1 hour Completion Time is reasonable to evaluate and take action to correct the degraded condition in an orderly manner and takes into account the low probability of an event requiring LOVS occurring during this interval.

D.1

Condition D applies if the Required Actions and associated Completion Times are not met.

Required Action D.1 ensures that Required Actions for the affected DG inoperabilities are initiated. The actions specified in LCO 3.8.1 are required immediately.

SURVEILLANCE REQUIREMENTS

The following SRs apply to each DG-LOVS Function.

SR 3.3.6.1

A CHANNEL FUNCTIONAL TEST is performed every 92 days to ensure that the entire sensor channel will perform its intended function when needed.

The Frequency of 92 days is based on plant operating experience with regard to channel OPERABILITY and drift, which demonstrates that failure of more than one sensor channel of a given function in any 92 day Frequency is a rare event. Any setting adjustment shall be consistent with the assumptions of the current plant specific setting analysis.

SR 3.3.6.2

Surveillance Requirement 3.3.6.3 is the performance of a CHANNEL CALIBRATION every 24 months. The CHANNEL

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CALIBRATION verifies the accuracy of each component within the sensor channel, except stepdown transformers, which are not calibrated. This includes calibration of the undervoltage relays and demonstrates that the equipment falls within the specified operating characteristics defined by the manufacturer.

The SR verifies that the sensor channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift between successive calibrations to ensure that the channel remains operational between successive tests. CHANNEL CALIBRATIONS must be performed consistent with the plant-specific setting analysis.

The as-found and as-left values must also be recorded and reviewed for consistency with the assumptions of the SR interval extension analysis. The requirements for this review are outlined in Reference 4.

The settings, as well as the response to a Loss of Voltage and Degraded Voltage test, shall include a single point verification that the trip occurs within the required delay time as shown in Reference 1, Section 7.3. The Frequency is based upon the assumption of a 24-month calibration interval for the determination of the magnitude of equipment drift in the plant setting analyses.

REFERENCES

1. UFSAR
2. CCNPP Setpoint File
3. IEEE No. 279, "Proposed IEEE Criteria for Nuclear Power Plant Protection Systems," August 1968
4. Calvert Cliffs Procedure EN-4-104, "Surveillance Testing"

B 3.3 INSTRUMENTATION

B 3.3.7 Containment Radiation Signal (CRS)

BASES

BACKGROUND

This LCO encompasses CRS actuation, which is a plant-specific instrumentation system that performs an actuation Function required to mitigate offsite dose, but is not otherwise included in LCO 3.3.5 or LCO 3.3.6. This is a non-Nuclear Steam Supply System ESFAS Function that, because of differences in purpose, design, and operating requirements, is not included in LCOs 3.3.5 and 3.3.6.

The CRS provides protection from radioactive contamination in the Containment in the event an irradiated fuel assembly should be severely damaged during handling.

The CRS will detect abnormal amounts of radioactive material in the Containment and will initiate purge valve closure to limit the release of radioactivity to the environment. The containment purge supply and exhaust valves are closed on a CRS when a high radiation level in Containment is detected.

The CRS includes two independent, redundant actuation logic channels. A single actuation subsystem initiates valve closure. One actuation logic channel also isolates the containment air exhaust fan, whereas the other actuation logic channel actuates the containment air supply fan. A list of actuated valves and an additional description of the CRS are included in Reference 1, Section 7.3. Both trains of CRS are actuated on a two-out-of-four coincidence from the same four containment radiation sensor channels. Containment purge isolation also occurs on a SIAS. The SIAS is addressed by LCO 3.3.4.

<u>Trip Setpoints and Allowable Values</u>

Trip setpoints used in the sensor modules are based on the analytical limits stated in Reference 1, Chapter 14. The selection of these trip setpoints is such that adequate protection is provided when all sensor and processing time delays are taken into account in the respective analytical limits. To allow for calibration tolerances, instrumentation uncertainties, and sensor channel drift, sensor module trip setpoints are conservatively adjusted with respect to the analytical limits. A detailed description of the methodology used to calculate the trip setpoints, including their explicit uncertainties, is provided in Reference 2. The actual nominal trip setpoint entered into the sensor module is more conservative than that specified by the Allowable Value. One example of such a change in measurement error is drift during the SR interval. If the measured setpoint does not exceed the Allowable Value, the bistable is considered OPERABLE.

Sensor channels, measurement channels, sensor modules, and actuation logic are described in the Background for B 3.3.4.

Setpoints in accordance with the Allowable Value will help ensure that 10 CFR Part 100 exposure limits are not violated during a Fuel Handling Accident, providing the plant is operated from within the LCOs at the onset of the Fuel Handling Accident and the equipment functions as designed.

APPLICABLE SAFETY ANALYSES

The CRS satisfies the requirements of 10 CFR 50.36(c)(2)(ii), Criterion 3.

LCO

Only the Allowable Values are specified in the LCO. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable, provided that operation and testing are consistent with the assumptions of the plant-specific setpoint calculations.

Each nominal trip setpoint specified is more conservative than the analytical limit assumed in the Fuel Handling Accident analysis in order to account for instrument uncertainties appropriate to the actuation Function. These uncertainties are defined in Reference 2. A sensor channel is inoperable if its actual trip setpoint is not within its required Allowable Value.

The Bases for the LCO on the CRS are discussed below for each Function:

a. Manual Actuation

The LCO on manual actuation backs up the automatic actuations and ensures operators have the capability to rapidly initiate the CRS Function if any parameter is trending toward its setpoint. At least one channel must be OPERABLE to be consistent with the requirements of LCO 3.9.3.

b. Containment Radiation-High Trip

The LCO on the radiation sensor channels requires that all four be OPERABLE. The radiation sensor channels have a measurement range of 10^{-1} - 10^{4} mr/hr.

The Containment Radiation-High trip setpoint is based on sensing radiation resulting from a fuel handling accident in order to prevent a release of radioactivity through the containment purge system.

c. Actuation Logic

One channel of actuation logic must be OPERABLE to be consistent with the requirements of LCO 3.9.3. If one fails, it must be restored to OPERABLE status.

APPLICABILITY

In MODE 5 or 6, the CRS isolation of containment purge valves is not required to be OPERABLE. However, during CORE ALTERATIONS or during movement of irradiated fuel, there is the possibility of a Fuel Handling Accident requiring the CRS on high radiation in Containment. Accordingly, the CRS must be OPERABLE during CORE ALTERATIONS and when moving any irradiated fuel in Containment when the containment purge valves are open.

In MODEs 1, 2, 3, and 4, the containment purge valves are sealed closed.

ACTIONS

A CRS sensor channel is inoperable when it does not satisfy the OPERABILITY criteria for the channel's Function. The most common cause of channel inoperability is outright failure or drift of the sensor module or measurement channel sufficient to exceed the tolerance allowed by Reference 2. Typically, the drift is not large, which at worst would result in a delay of actuation rather than a total loss of Function. This determination is generally made during the performance of a CHANNEL CALIBRATION when the process instrument is set up for adjustment to bring it within specification. Sensor drift could also be identified during

CHANNEL CHECKS. CHANNEL FUNCTIONAL TESTS identify sensor module drift. If the actual trip setpoint is not within the Allowable Value in SR 3.3.7.2, the channel is inoperable and the appropriate Conditions must be entered.

In the event that either a sensor channel's trip setpoint is found nonconservative with respect to the Allowable Value, or the sensor, instrument loop, signal processing electronics, or sensor module is found inoperable, that channel should be declared inoperable and the LCO Condition entered.

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

Condition A applies to the failure of one Containment Radiation-High trip CRS channel. The Required Action is to place the affected channel in the trip condition within 4 hours, or suspend CORE ALTERATIONS and suspend movement of irradiated fuel assemblies within Containment immediately. The Completion Time accounts for the fact that three redundant channels monitoring containment radiation are still available to provide a single trip input to the CRS logic to provide the automatic mitigation of a radiation release.

B.1 and B.2

Condition B applies to the failure of the required manual actuation or actuation logic, to the failure of more than one radiation sensor channel, or if the Required Action and associated Completion Time of Condition A are not met. Required Action B.1 is to place the containment purge and exhaust isolation valves in the closed position. The Required Action immediately performs the isolation Function of the CRS. Required Action B.2 is to immediately enter the applicable Conditions and Required Actions for the affected isolation valves of LCO 3.9.3 that were made inoperable by the inoperable instrumentation of the CRS LCO. The Required Action directs the operator to take actions appropriate for the containment isolation Function of the CRS. The Completion Time accounts for the fact that the automatic capability to isolate Containment on valid containment high radiation signals is degraded during conditions in which a

Fuel Handling Accident is possible and CRS provides the only automatic mitigation of radiation release.

SURVEILLANCE REQUIREMENTS

SR 3.3.7.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one sensor channel to a similar parameter on other channels. It is based on the assumption that sensor channels monitoring the same parameter should read approximately the same value.

Significant deviations between the two sensor channels could be an indication of excessive sensor channel drift in one of the channels or of something more serious. 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 plant staff, based on a qualitative assessment of the sensor channel that considers sensor channel uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the transmitter or the signal processing equipment has drifted outside its limits.

The Frequency, about once every shift, is based on operating experience that demonstrates the rarity of sensor channel failure. Since the probability of two random failures in redundant channels in any 12-hour period is low, the CHANNEL CHECK minimizes the chance of loss of protective function due to failure of redundant channels. The CHANNEL CHECK supplements less formal, but more frequent, checks of the channel during normal operational use of the displays.

SR 3.3.7.2

Proper operation of the actuation relays is verified by verification of the relay driver output signal.

The Frequency of 92 days is based on plant operating experience with regard to actuation channel OPERABILITY, which demonstrates that failure of more than one channel of a given Function in any 92-day interval is a rare event.

SR 3.3.7.3

A CHANNEL FUNCTIONAL TEST is performed on each containment radiation sensor channel to ensure the entire channel, except for sensor and initiating relays, will perform its intended function.

The Frequency of 92 days is based on plant operating experience with regard to sensor channel OPERABILITY and drift, which demonstrates that failure of more than one channel of a given Function in any 92-day interval is a rare event.

SR 3.3.7.4

CHANNEL CALIBRATION is a check of the sensor channel including the sensor. The Surveillance verifies that the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for sensor channel drift between successive calibrations to ensure that the channel remains operational between successive tests. CHANNEL CALIBRATIONS must be performed consistent with Reference 2.

The Frequency is based upon the assumption of a 24-month calibration interval based on the refueling interval and the instruments not being inservice during power operations, but part of preparation for being placed in service is a CHANNEL CALIBRATION.

SR 3.3.7.5

Every 24 months, a CHANNEL FUNCTIONAL TEST is performed on the manual CRS actuation circuitry.

This surveillance test verifies that the actuation push buttons are capable of opening contacts in the actuation logic as designed, de-energizing the actuation relays and providing manual actuation of the Function. The 24-month Frequency is based on the need to perform this SR under the conditions that apply during a plant outage and the potential for an unplanned transient if the SR were performed with the reactor at power. Operating experience has shown these components usually pass the surveillance test when performed at a Frequency of once every 24 months.

SR 3.3.7.6

This surveillance test ensures that the train actuation response times are less than or equal to the maximum times assumed in the analyses. Response times are defined in the same manner as ESF RESPONSE TIME. Response time testing acceptance criteria are included in Reference 1, Section 7.3. The 24-month Frequency is based upon plant operating experience, which shows random failures of instrumentation components causing serious response time degradation, but not channel failure, are infrequent occurrences. Testing of the final actuating devices, which make up the bulk of the response time, is included. Testing of the final actuating device is one channel is included in the testing of each actuation logic channel.

REFERENCES

1. UFSAR
2. CCNPP Setpoint File

B 3.3 INSTRUMENTATION

B 3.3.8 Control Room Recirculation Signal (CRRS)

BASES

BACKGROUND

This LCO encompasses CRRS actuation, which is a plant-specific instrumentation channel that performs an actuation Function required for plant protection, but is not otherwise included in LCO 3.3.5 or LCO 3.3.6. This is a non-Nuclear Steam Supply System ESFAS Function that, because of differences in purpose, design, and operating requirements, is not included in LCO 3.3.5 and LCO 3.3.6.

The CRRS terminates the normal supply of outside air to the Control Room, shuts off the kitchen and toilet exhaust fan, and initiates actuation of the Control Room Emergency Ventilation System (CREVS) fans and places filters in service to minimize operator radiation exposure. This places CREVS in filtered recirculation mode. When the radiation level signal from the measurement channel exceeds the trip setpoint, the trip circuit sends a CRRS to actuate equipment placing CREVS in filtered recirculation mode. Control room isolation also occurs on a SIAS.

Trip Setpoints and Allowable Values

The Trip setpoint used in the trip circuit is conservatively adjusted with respect to the Allowable Value. One example of such a change in measurement error is drift during the SR interval. If the measured setpoint does not exceed the Allowable Value, the trip circuit is considered OPERABLE.

APPLICABLE SAFETY ANALYSES

The CRRS, in conjunction with the CREVS, maintains the control room atmosphere within conditions suitable for prolonged occupancy throughout the duration of any one of the accidents discussed in Reference 1, Chapter 14. The radiation exposure of control room personnel, through the duration of any one of the postulated accidents discussed in Reference 1, Chapter 14, meets the intent of Reference 1, Appendix 1C.

The CRRS satisfies the requirements of 10 CFR 50.36(c)(2)(ii), Criterion 3.

BASES

LCO

LCO 3.3.8 requires one channel of CRRS to be OPERABLE. The required channel consists of a trip circuit and the gaseous radiation monitor (measurement channel). The specific Allowable Value for the setpoint of the CRRS is listed in the SRs.

Only the Allowable Value is specified for the trip Function in the LCO. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable.

Operation and testing are consistent with the assumptions of Reference 2. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value.

The Bases for the LCO on the CRRS is that one channel of airborne radiation detection and trip circuitry is required to be OPERABLE to ensure the Control Room isolates on high gaseous concentration. The Allowable Value was established as part of original plant design. It provides reasonable assurance of safety for control room personnel.

APPLICABILITY

The CRRS Functions must be OPERABLE in MODEs 1, 2, 3, and 4, and during movement of irradiated fuel assemblies to ensure a habitable environment for the control room operators.

ACTIONS

A CRRS channel is inoperable when it does not satisfy the OPERABILITY criteria for the channel's function. The most common cause of channel inoperability is outright failure or drift of the trip circuit or measurement channel sufficient to exceed the nominal trip setpoint. Typically, the drift is not large, which at worst would result in a delay of actuation rather than a total loss of function. This determination is generally made during the performance of a CHANNEL CALIBRATION when the process instrument is set up for adjustment to bring it to within specification. CHANNEL FUNCTIONAL TESTS identify trip circuit drift. If the trip setpoint is not within the Allowable Value, the channel is inoperable and the appropriate Conditions must be entered.

A.1, B.1, B.2, C.1, C.2.1, and C.2.2

Conditions A, B, and C are applicable to the CRRS trip circuit and measurement channel. Condition A applies to the failure of the CRRS trip circuit or measurement channel in MODE 1, 2, 3, or 4. Entry into this Condition requires action to either restore the failed channel or manually perform the CREVS function (Required Action A.1). The Completion Time of 1 hour is sufficient to complete the Required Actions. If the channel cannot be restored to OPERABLE status, 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 Completion Times of 6 hours and 36 hours for reaching MODEs 3 and 5 from MODE 1 are reasonable, based on operating experience and normal cooldown rates, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant safety systems or operators.

Condition C applies to the failure of the CRRS trip circuit or measurement channel when moving irradiated assemblies. The Required Actions are immediately taken to place one OPERABLE CREVS train in the recirculation mode with post-LOCA fans in service or to suspend movement of irradiated fuel assemblies. The Completion Time recognizes the fact that the radiation signal is the only Function available to initiate control room isolation in the event of a Fuel Handling Accident.

SURVEILLANCE REQUIREMENTS

SR 3.3.8.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Acceptance criteria are determined by the plant 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 transmitter or the signal processing equipment has drifted outside its limit.

The Frequency, about once every shift, is based on operating experience that demonstrates the rarity of channel failure. The CHANNEL CHECK supplements less formal, but more frequent, checks of channel OPERABILITY during normal operational use of the displays associated with the LCO required channels. In addition, a down-scale alarm and up-scale alarm immediately alert operations to loss of the channel.

SR 3.3.8.2

A CHANNEL FUNCTIONAL TEST is performed on the control room radiation monitoring channel to ensure the entire channel will perform its intended function.

The Frequency of 92 days is based on plant operating experience with regard to channel OPERABILITY and drift.

SR 3.3.8.3

CHANNEL CALIBRATION is a check of the CRRS channel, including the sensor. The surveillance test verifies that the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for channel drift between successive calibrations to ensure that the channel remains operational between successive surveillance tests. CHANNEL CALIBRATIONS must be performed consistent with Reference 2.

The Frequency of 24 months has been shown by operating experience to be adequate to detect any failures.

REFERENCES

1. UFSAR
2. CCNPP Setpoint File

B 3.3 INSTRUMENTATION

B 3.3.9 Chemical and Volume Control System (CVCS) Isolation Signal

BASES

BACKGROUND

This LCO encompasses CVCS Isolation Signal actuation. This is a plant-specific instrumentation channel that performs an actuation Function required for plant protection and is not otherwise included in LCO 3.3.5 or LCO 3.3.6. This is a non-Nuclear Steam Supply System ESFAS Function that, because of differences in purpose, design, and operating requirements, is not included in LCOs 3.3.5 and 3.3.6.

The CVCS Isolation Signal isolates the RCS and provides protection from radioactive contamination, as well as personnel and equipment protection in the event of a letdown line rupture outside Containment (Reference 1).

Each of the two actuation logic channels will isolate a separate letdown isolation valve in response to a high pressure condition in either the West Penetration Room or Letdown Heat Exchanger Room. Two pressure detectors in each of these rooms feed the four sensor channels. On a two-out-of-four coincidence, both actuation logic channels will actuate.

Trip Setpoints and Allowable Values

Trip setpoints used in the sensor modules are based on Reference 2 to protect personnel and equipment and minimize radioactive contamination. The selection of these trip setpoints 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, and sensor channel drift, sensor module trip setpoints are conservatively adjusted with respect to the Allowable Value. A detailed description of the methodology used to calculate the trip setpoints, including their explicit uncertainties, is provided in Reference 2. The actual nominal trip setpoint entered into the sensor module is more conservative than that specified by the Allowable Value. One example of such a change in measurement error is drift during the SR interval. If the measured setpoint does not exceed the Allowable Value, the sensor module is considered OPERABLE.

BASES

Sensor modules, measurement channels, sensor channels, and actuation logic are described in the Background for B 3.3.4.

APPLICABLE SAFETY ANALYSES

The CVCS Isolation Signal is redundant to the SIAS for letdown line breaks outside Containment (Reference 2). In addition, an excess flow check valve is located in Containment just downstream of the regenerative heat exchanger, which is designed to isolate letdown when flow exceeds 255 gpm.

The CVCS satisfies the requirements of 10 CFR 50.36(c)(2)(ii), Criterion 3.

LCO

Only the Allowable Values are specified in the LCO. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable, provided that operation and testing are consistent with the assumptions of the plant-specific setpoint calculations.

Each nominal trip setpoint specified is more conservative than the Allowable Value, in order to account for instrument uncertainties appropriate to the trip Function. These uncertainties are defined in Reference 2.

Chemical and Volume Control System isolation consists of closing the appropriate valve. This is undesirable at power, since letdown isolation will result (Reference 3). The absence of letdown flow will significantly decrease the charging flow temperature due to the absence of the regenerative heat exchanger preheating, causing unnecessary thermal stress to the charging nozzle. Therefore, the preferred action is to restore the valve function to OPERABLE status.

Four channels of west penetration room and letdown heat exchanger room pressure sensors, and two actuation logic channels are required to be OPERABLE.

The Allowable Values and trip setpoints are established in order to isolate the CVCS from Containment, in the event of a letdown line rupture outside Containment, to minimize radioactive contamination and protect personnel and equipment.

APPLICABILITY

The CVCS Isolation Signal must be OPERABLE in MODEs 1, 2, 3, and 4, since the possibility of a loss of coolant accident is greatest in these MODEs. In MODE 5 or 6, the probability is greatly diminished, and there is time to manually isolate CVCS.

ACTIONS

A CVCS isolation channel is inoperable when it does not satisfy the OPERABILITY criteria for the channel's Function. The most common cause of channel inoperability is outright failure or drift of the sensor module or measurement channel sufficient to exceed the tolerance allowed by Reference 2. Typically, the drift is not large and would result in a delay of actuation rather than a total loss of function. This determination is generally made during the performance of a CHANNEL CALIBRATION when the process instrument is set up for adjustment to bring it to within specification. Sensor drift could also be identified during CHANNEL CHECKS. CHANNEL FUNCTIONAL TESTS identify sensor module drift. If the trip setpoint is not consistent with the Allowable Value in SR 3.3.9.2, the channel must be declared inoperable immediately and the appropriate Conditions must be entered.

In the event a sensor channel's trip setpoint is found nonconservative with respect to the Allowable Value, or the sensor, instrument loop, signal processing electronics, or bistable is found inoperable, that channel should be declared inoperable and the LCO Condition entered.

When the number of inoperable sensor channels in a trip Function exceeds those specified in any related Condition associated with the same trip Function, the plant is outside the safety analysis. Therefore, LCO 3.0.3 should be immediately entered if applicable in the current MODE of operation.

<u>A.1</u>

Condition A applies to the failure of one CVCS actuation logic channel associated with the CVCS Isolation Signal. Required Action A.1 requires restoration of the inoperable

BASES

channel to restore redundancy of the affected Function. The Completion Time of 48 hours is consistent with the Completion Time of other ESFAS Functions and should be adequate for most repairs, while minimizing the risk of operating with an inoperable channel.

B.1, B.2.1, and B.2.2

Condition B applies if one of the four CVCS sensor channels is inoperable. The Required Actions are identical to those of ESFAS Functions employing four redundant sensors specified in LCO 3.3.4. The channel must be placed in bypass or trip if it cannot be repaired within 1 hour (Required Action B.1). The provision of four sensor channels allows one channel to be bypassed (removed from service) during operations, placing the ESFAS in two-out-of-three coincidence logic. Placing the channel in bypass is preferred, since the CVCS isolation Function will be in two-out-of-three logic. This will avoid possible inadvertent CVCS isolation if an additional channel fails. The 1-hour Completion Time to bypass or trip the channel is sufficient time to perform the Required Actions.

Once the Required Action to trip or bypass the sensor channel has been complied with, Required Action B.2.1 and Required Action B.2.2 provide for restoring the channel to OPERABLE status or placing it in trip within 48 hours. Required Action B.2.1 restores the full capability of the Function. Required Action B.2.2 places the Function in a one-out-of-three configuration. In this configuration, common cause failure of dependent channels cannot prevent CVCS isolation actuation. The Completion Time provides the operator with time to take appropriate actions and still ensures that any risk involved in operating with a failed channel is acceptable. It is improbable that a failure of a second channel will occur during any given 48-hour period.

C.1 and C.2

Condition C applies if two of the four CVCS west penetration room/letdown heat exchanger room Pressure-High trip sensor channels are inoperable. The Required Actions are identical to those for other ESFAS Functions employing four redundant sensors in LCO 3.3.4.

Restoring at least one sensor channel to OPERABLE status is the preferred Required Action. If this cannot be accomplished, one channel should be placed in bypass and the other channel in trip. The allowed Completion Time of 1 hour is sufficient time to perform the Required Actions.

Once the Required Action to trip or bypass the channel has been complied with, Required Action C.2 provides for restoring one channel to OPERABLE status within 48 hours. The justification of the 48-hour Completion Time is the same as for Condition B.

After one channel is restored to OPERABLE status, the provisions of Condition C still apply to the remaining inoperable channel.

The Required Action is modified by a Note stating that LCO 3.0.4 is not applicable. The Note was added to allow the changing of MODEs even though two channels are inoperable, with one channel bypassed and one tripped. MODE changes in this configuration are allowed, to permit maintenance and testing on one of the inoperable channels. In this configuration, the protective system is in a one-out-of-two logic, and the probability of a common cause failure affecting both of the OPERABLE channels during the 48 hours permitted is remote.

D.1 and D.2

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

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.9.1

Performance of the CHANNEL CHECK on each CVCS isolation pressure indicating sensor channel once every 12 hours 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 sensor channels monitoring the same parameter should read approximately the same value.

Significant deviations between the two sensor channels could be an indication of excessive sensor channel drift in one of the channels or of something more serious. 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 plant staff, based on a qualitative assessment of the sensor channel that considers sensor channel uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the transmitter or the signal processing equipment has drifted outside its limit.

The Frequency, about once every shift, is based on operating experience that demonstrates the rarity of channel failure. Since the probability of two random failures in redundant channels in any 12-hour period is low, the CHANNEL CHECK minimizes the chance of loss of protective function due to failure of redundant channels. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays.

SR 3.3.9.2

A CHANNEL FUNCTIONAL TEST is performed on each sensor channel to ensure the entire channel, except for sensor and initiation logic, will perform its intended function.

The Frequency of 92 days is based on plant operating experience with regard to channel OPERABILITY and drift, which demonstrates that failure of more than one channel of a given Function in any 92-day interval is a rare event.

Note 1 indicates proper operation of the individual actuation relays is verified by verification of proper relay driver output signal. Note 2 indicates that relays that cannot be tested at power are excepted from the SR while at power. These relays must, however, be tested once per 24 months.

SR 3.3.9.3

CHANNEL CALIBRATION is a check of the sensor channel including the sensor. The surveillance test verifies that the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for sensor channel drift between successive calibrations to ensure that the channel remains operational between successive tests. CHANNEL CALIBRATIONS must be performed consistent with Reference 2.

The as-found and as-left values must also be recorded and reviewed for consistency with the assumptions of the SR interval extension analysis. The requirements for this review are outlined in Reference 4.

Radiation detectors may be removed and calibrated in a laboratory, calibrated in place using a transfer source, or replaced with an equivalent laboratory calibrated unit.

The Frequency is based upon the assumptions of a 24-month calibration interval for the determination of the magnitude of equipment drift in the setpoint analysis, and includes operating experience, as well as consistency with a 24-month fuel cycle.

SR 3.3.9.4

This surveillance test ensures that the train actuation response times are less than or equal to the maximum times assumed in the analyses. Response times are defined in the same manner as ESF RESPONSE TIME. The 24-month Frequency is based upon plant operating experience, which shows random failures of instrumentation components causing serious

BASES

response time degradation, but not channel failure, are infrequent occurrences. Testing of the final actuating devices, which make up the bulk of the response time, is included. Testing of the final actuating device in one channel is included in the testing of each actuation logic channel.

REFERENCES

1. Updated Final Safety Analysis Report, Section 7.3, "Engineered Safety Features Actuation Systems" and Section 10A.7.17, "Leak Detection Equipment"
2. CCNPP Setpoint File
3. Letter from Mr.R. E. Denton (BGE) to NRC Document Control Desk, dated June 6, 1995, "License Amendment Request; Extension of Instrument Surveillance Intervals"
4. Calvert Cliffs Procedure EN-4-104, "Surveillance Testing"

B 3.3 INSTRUMENTATION

B 3.3.10 Post-Accident Monitoring (PAM) Instrumentation

BASES

BACKGROUND

The primary purpose of the PAM instrumentation is to display plant 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, that are required for safety systems to accomplish their safety Functions for DBAs (Reference 1).

The OPERABILITY of the PAM instrumentation ensures that there is sufficient information available on selected plant parameters to monitor and assess plant status and behavior following an accident.

The availability of PAM 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 indicator channels are identified by plant-specific documents (Reference 2) addressing the recommendations of Reference 3, as required by Reference 4.

Type A variables are included in this LCO because they provide the primary information required to permit the control room operator to take specific manually-controlled actions, for which no automatic control is provided, that are required for safety systems to accomplish their safety functions for some DBAs.

Category I variables are the key variables deemed risk significant because they are needed to:

- Determine whether other systems important to safety are performing their intended functions;
- Provide information to the operators that will enable them to determine the potential for causing a gross breach of the barriers to radioactivity release; and
- Provide information regarding the release of radioactive materials to allow for early indication of the need to initiate action necessary to protect the public and for an estimate of the magnitude of any impending threat.

These key variables are identified by plant-specific analyses in Reference 2. These analyses identified the plant-specific Type A and Category I variables and provided justification for deviating from the NRC proposed list of Category I variables.

APPLICABLE SAFETY ANALYSES

The PAM instrumentation ensures the OPERABILITY of Reference 3 Type A variables, so that the control room operating staff can:

- Perform the diagnosis specified in the emergency operating procedures. These variables are restricted to preplanned actions for the primary success path of DBAs; and
- Take the specified, preplanned, manually-controlled actions, for which no automatic control is provided, that are required for safety systems to accomplish their safety functions.

The PAM instrumentation also ensures OPERABILITY of Category I, non-Type A variables. This ensures the control room operating staff can:

- Determine whether systems important to safety are performing their intended functions;
- Determine the potential for causing a gross breach of the barriers to radioactivity release;
- Determine if a gross breach of a barrier has occurred; and
- Initiate action necessary to protect the public, as well as to obtain an estimate of the magnitude of any impending threat.

Post-accident monitoring instrumentation that satisfies the definition of Type A in Reference 3 meets 10 CFR 50.36(c)(2)(ii), Criterion 3.

Category I, non-Type A PAM instruments are retained in the Specification because they are intended to assist operators in minimizing the consequences of accidents. Therefore, these Category I variables are important in reducing public risk.

LCO

Limiting Condition for Operation 3.3.10 requires two OPERABLE indication channels for all but one Function to ensure no single failure prevents the operators from being presented with the information necessary to determine the status of the plant and to bring the plant to, and maintain it in, a safe condition following that accident.

Furthermore, provision of two indication channels allows a CHANNEL CHECK during the post-accident phase to confirm the validity of displayed information.

An indication channel consists of field transmitters or process sensors and associated instrumentation, providing a measurable electronic signal based upon the physical characteristics of the parameter being measured, plus a display of the measured parameter.

The exceptions to the two-channel requirement are CIV position and the subcooled SMM. In the case of valve position, 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 the 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. Alternate means are available for obtaining information provided by the SMM.

Listed below are discussions of the specified instrument Functions listed in Table 3.3.10-1.

1. <u>Wide Range Logarithmic Neutron Flux Monitors</u>

   Wide range logarithmic neutron flux is a Category I variable indication is provided to verify reactor shutdown.

The wide range logarithmic neutron flux PAM channels consist of two wide range neutron monitoring channels.

2, 3. RCS Outlet and Inlet Temperature

Reactor Coolant System outlet and inlet temperatures are Category I variables provided for verification of core cooling and long-term surveillance.

Reactor outlet temperature inputs to the PAM are provided by four resistance elements and associated transmitters in each loop. The channels provide indication over a range of 50°F to 700°F.

4. SMM

The RCS SMM is provided to monitor for inadequate core cooling by calculating the margin to saturation based on the RCS pressure/temperature relationships and displaying the calculated margin (1°F to 100°F) on a control room indicator. The SMM also generates a low subcooled margin alarm should the temperature margin drop below predetermined limits. The SMM is a microprocessor based instrument provided with inputs from the RCS hot and cold legs temperature instrumentation and wide range RCS pressure channels.

The RCS SMM is one of three components of inadequate core cooling instrumentation with the RCS SMM inoperable, the core exit thermocouple and reactor vessel water level monitoring systems provide diverse indication of core cooling.

5. Reactor Vessel Water Level Monitor

Reactor vessel water level monitors are provided for verification and long-term surveillance testing of core cooling.

The reactor vessel water level monitoring system uses a heated junction thermocouple system. The heated junction thermocouple system measures reactor coolant inventory with discrete heated junction thermocouple sensors located in different levels within a separator tube. The system provides a direct measurement of the collapsed liquid level above the fuel alignment plate. The collapsed level represents the amount of liquid mass that is in the reactor vessel above the core. Measurement of the collapsed water level is selected because it is a direct indication of the water inventory. The collapsed level is obtained over the same temperature and pressure range as the saturation measurements, thereby encompassing all operating and accident conditions where it must function. Also, it functions during the recovery interval. Therefore, it is designed to survive the high steam temperature that may occur during the preceding core recovery interval.

The level range extends from the top of the vessel down to 10" above the top of the fuel alignment plate. The response time is short enough to track the level during small break LOCA events. The resolution is sufficient to show the initial level drop, the key locations near the hot leg elevation, and the lowest levels just above the alignment plate. This provides the operator with adequate indication to track the progression of the accident and to detect the consequences of its mitigating actions or the functionality of automatic equipment.

A channel has eight sensors in a probe. A channel is OPERABLE if four sensors, one in the upper three and three in the lower five, are OPERABLE.

6. <u>Containment Sump Water Level (wide range) Monitor</u>

Containment sump water level monitors are provided for verification and long-term surveillance of RCS integrity.

Containment sump water level instrumentation consists of two level transmitters that provide input to control room indicators. The transmitters are located above the containment flood level and utilize sealed reference legs to sense water level.

7. <u>Containment Pressure (wide range) Monitor</u>

The containment pressure monitor is provided for verification of RCS and containment OPERABILITY.

Containment pressure instrumentation consists of three containment pressure transmitters with overlapping ranges that provide input to control room indicators. The transmitters are located outside the Containment and are not subject to a harsh environment.

8. <u>CIV Position Indicator</u>

Containment isolation valve position indicators are provided for verification of containment OPERABILITY and integrity.

In the case of CIV position, the important information is the isolation status of the containment penetration. The LCO requires one channel of valve position indication in the Control Room to be OPERABLE for each active CIV in a containment penetration flow path, i.e., two total channels of CIV position indication for a penetration flow path with two active valves. For containment penetrations with only one active CIV having control room indication, Note (b) requires a single channel of valve position indication to be OPERABLE. This is sufficient to redundantly verify the isolation status of each isolable penetration via indicated status of the active valve, as applicable, and prior knowledge of passive valve or system boundary status. If a penetration flow path is isolated, position indication for the CIV(s) in the associated penetration flow path is not needed to determine status. Therefore, the position indication for valves in an isolated penetration flow path is not required to be OPERABLE.

The CIV position PAM instrumentation consists of ZL-505, 506, 515, 516, 2080, 2180, 2181, 3832, 3833, 4260, 5291, 5292, 6900, and 6901 (Reference 5).

9. Containment Area Radiation (high range) Detector

Containment area radiation detectors are provided to monitor for the potential of significant radiation releases and to provide release assessment for use by operations in determining the need to invoke site emergency plans.

Containment area radiation instrumentation consists of two radiation detectors with displays and alarm in the Control Room. The radiation detectors have a measurement range of 1 to 10^8 R/hr.

10. Containment Hydrogen Monitors

Containment hydrogen monitors are provided to detect high hydrogen concentration conditions that represent a potential for Containment breach. This variable is also important in verifying the adequacy of mitigating actions.

Containment hydrogen instrumentation monitors samples from six locations inside the Containment. Two groups of three sampling lines from each Containment provide samples to the two cabinets in the sampling room. The cabinets contain the hydrogen analyzers and related equipment. The sampling system is fully operational from this remote station due to post-accident personnel safety considerations. Once the system is placed in the sampling mode, any malfunction or high hydrogen condition will generate a signal in the annunciator system in the main Control Room.

11. Pressurizer Pressure (wide range)

Pressurizer wide range pressure is a Category I variable provided for verification of core cooling and RCS integrity long-term surveillance.

Wide range pressurizer pressure is measured by two pressure transmitters with a span of 0 psia to 4000 psia. The pressure transmitters are located inside the Containment. Redundant monitoring capability is provided by two indication channels. Control Room indications are provided.

Pressurizer pressure is a Type I variable because the operator uses this indication to monitor the cooldown of the RCS following a LOCA and other DBAs. Operator actions to maintain a controlled cooldown, such as adjusting steam generator pressure or level, would use this indication. Furthermore, pressurizer pressure is one factor that may be used in decisions to terminate RCP operation.

12. <u>Steam Generator Pressure Transmitter</u>

Steam generator pressure transmitters are Category 1 instruments and are provided to monitor operation of decay heat removal via the steam generators.

There are four redundant pressure transmitters per steam generator, but only two per steam generator are required to satisfy the Technical Specification Requirements. The transmitter provides wide range indication over the range from 0 to 1200 psia. Each transmitter provides input to control room indication. Since the primary indication used by the operator during an accident is the control room indicator, the PAM instrumentation Specification deals specifically with this portion of the instrument channel.

13. <u>Pressurizer Level Transmitters</u>

Pressurizer level transmitters are used to determine whether to terminate safety injection, if still in progress, or to reinitiate safety injection if it has been stopped. Knowledge of pressurizer water level is also used to verify the plant conditions necessary to establish natural circulation in the RCS and to verify that the plant is maintained in a safe shutdown condition.

Pressurizer Level instrumentation consists of two pressurizer level transmitters that provide input to control room indicators.

14. Steam Generator Water Level Transmitters

Steam Generator Water Level transmitters are provided to monitor operation of decay heat removal via the steam generators. The Category I indication of steam generator level is the extended startup range level instrumentation. The extended startup range level covers a span of -40 inches to -63 inches (relative to normal operating level), above the lower tubesheet. The measured differential pressure is displayed in inches of water at process conditions of the fluid. Redundant monitoring capability is provided by four transmitters. The uncompensated level signal is input to the plant computer and a control room indicator. Steam generator water level instrumentation consists of two level transmitters.

Operator action is based on the control room indication of steam generator water level. The RCS response during a design basis small break LOCA is dependent on the break size. For a certain range of break sizes, the boiler condenser mode of heat transfer is necessary to remove decay heat. Extended startup range level is a Type A variable because the operator must manually raise and control the steam generator level to establish boiler condenser heat transfer. Feedwater flow is increased until indication is in range.

15. Condensate Storage Tank Level Monitor

Condensate storage tank (CST) level monitors are provided to ensure water supply for AFW. The CST provides the ensured safety grade water supply for the AFW System. Inventory is monitored by a 0- to 144-inch level indication for each tank. Condensate storage tank level is displayed on a control room indicator and plant computer. In addition, a control room annunciator alarms on low level.

Condensate storage tank level is considered a Type A variable because the control room meter and annunciator are considered the primary indication used by the Operator. The DBAs that require AFW are the steam line break and loss of main feedwater. The CST is the initial source of water for the AFW System. However, as the CST is depleted, manual operator action is necessary to replenish the CST or align suction to the AFW pumps from the hotwell.

16, 17, 18, 19. Core Exit Temperature Monitor

Core Exit Temperature monitors are provided for verification and long--term surveillance of core cooling.

An evaluation was made of the minimum number of valid core exit thermocouples necessary for inadequate core cooling detection. The evaluation determined the reduced complement of core exit thermocouples necessary to detect initial core uncovery and trend the ensuing core heatup. The evaluations account for core nonuniformities, including incore effects of the radial decay power distribution and excore effects of condensate runback in the hot legs and nonuniform inlet temperatures. Based on these evaluations, adequate or inadequate core cooling detection is ensured with two valid core exit thermocouples per quadrant.

The design of the Incore Instrumentation System includes a Type K (chromel alumel) thermocouple within each of the 45 incore instrument detector assemblies.

The junction of each thermocouple is located more than a foot above the fuel assembly, inside a structure that supports and shields the incore instrument detector assembly string from flow forces in the outlet plenum region. These core exit thermocouples monitor the temperature of the reactor coolant as it exits the fuel assemblies.

The core exit thermocouples have a usable temperature range from 40°F to 2300°F, although accuracy is reduced at temperatures above 1800°F.

20. Pressurizer Pressure (low range)

    Pressurizer low range pressure is a Category I variable provided for verification of core cooling and RCS integrity long-term surveillance.

    Low-range pressurizer pressure is measured by two pressure transmitters with a span of 0 psia to 1600 psia. The pressure transmitters are located inside the Containment. Redundant monitoring capability is provided by two indication channels. Control Room indications are provided.

    Pressurizer pressure is a Type I variable because the operator uses this indication to monitor the cooldown of the RCS following a LOCA and other DBAs. Operator actions to maintain a controlled cooldown, such as adjusting steam generator pressure or level, would use this indication. Furthermore, pressurizer pressure is one factor that may be used in decisions to terminate RCP operations.

Two indication channels are required to be OPERABLE for all but two Functions. Two OPERABLE channels ensure that no single failure, within either the PAM instrumentation or its auxiliary supporting features or power sources (concurrent with the failures that are a condition of or result from a specific accident), prevents the operators from being presented the information necessary for them to determine the safety status of the plant, and to bring the plant to and maintain it in a safe condition following that accident.

In Table 3.3.10-1 the exceptions to the two channel requirement are CIV position and the SMMs.

Two OPERABLE core exit thermocouples are required for each channel in each quadrant to provide indication of radial distribution of the coolant temperature rise across representative regions of the core. Power distribution symmetry was considered in determining the specific number and locations provided for diagnosis of local core problems. Therefore, two randomly selected thermocouples may not be sufficient to meet the two thermocouples per channel requirement in any quadrant. The two thermocouples in each channel must meet the additional requirement that one be located near the center of the core and the other near the core perimeter, such that the pair of core exit thermocouples indicate the radial temperature gradient across their core quadrant. The two channels in each core quadrant must be electronically independent. A core exit thermocouple's operability is based on a comparison of the core exit thermocouple temperature indication with the hot leg resistance temperature detector temperature indication. Different criteria have been specified for interior core exit thermocouples and peripheral core exit thermocouples to account for the core radial power distribution. Plant specific evaluations in response to Item II.F.2 of NUREG-0737 should have identified the thermocouple pairings that satisfy these requirements. Two sets of two thermocouples in each quadrant ensure a single failure will not disable the ability to determine the radial temperature gradient.

For loop- and steam generator-related variables, the required information is individual loop temperature and individual steam generator level. In these cases, two channels are required to be OPERABLE for each loop of steam generator to redundantly provide the necessary information.

In the case of CIV position, 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 the 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.

The SMM monitors, core exit thermocouples, and the reactor vessel water level monitoring system comprise the inadequate core cooling instrumentation. The function of the inadequate core cooling instrumentation is to enhance the ability of the plant operator to diagnose the approach to, and recovery from, inadequate core cooling.

APPLICABILITY

The PAM instrumentation LCO is applicable in MODEs 1, 2, and 3. These variables are related to the diagnosis and preplanned actions required to mitigate DBAs. The applicable DBAs are assumed to occur in MODEs 1, 2, and 3. In MODEs 4, 5, and 6, plant conditions are such that the likelihood of an event occurring requiring PAM instrumentation is low; therefore, PAM instrumentation is not required to be OPERABLE in these MODEs.

ACTIONS

Note 1 has been added in the ACTIONS to exclude the MODE change restriction of LCO 3.0.4. This exception allows entry into the applicable MODE while relying on the ACTIONS, even though the ACTIONS may eventually require plant shutdown. This exception is acceptable due to the passive function of the indication channels, the operator's ability to monitor an accident using alternate instruments and methods, and the low probability of an event requiring these indication channels.

Note 2 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 Table 3.3.10-1. 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

When one or more Functions have one required indication channel that is inoperable, the required inoperable channel must be restored 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-Reference 3 indication 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.

B.1

This Required Action specifies initiation of actions in accordance with Specification 5.6.7, which requires a written report to be submitted to the NRC. This report discusses the results of the root cause evaluation of the inoperability and identifies proposed restorative Required Actions. This Required Action is appropriate in lieu of a shutdown requirement, given the likelihood of plant conditions that would require information provided by this instrumentation. Also, alternative Required Actions such as grab sampling or diverse indications are identified before a loss of functional capability condition occurs.

C.1

When one or more Functions have two required indication channels inoperable (i.e., two channels inoperable in the same Function), one channel in the Function should be restored to OPERABLE status within 7 days. The Completion Time of 7 days is based on the relatively low probability of an event requiring PAM instrumentation 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

When two required hydrogen monitor channels are inoperable, Required Action D.1 requires one channel to be restored to OPERABLE status. This Required Action restores the monitoring capability of the hydrogen monitor. The 72-hour Completion Time is based on the relatively low probability of an event requiring hydrogen monitoring and the availability of alternative means to obtain the required information. Continuous operation with two required channels inoperable is not acceptable because alternate indications are not available.

E.1

This Required Action directs entry into the appropriate Condition referenced in Table 3.3.10-1. The applicable Condition referenced in the Table is Function-dependent. Each time Required Action C.1 or D.1 is not met and the associated Completion Time has expired, Condition E is entered for that channel and provides for transfer to the appropriate subsequent Condition.

F.1 and F.2

If the Required Action and associated Completion Time of Condition C are not met, and Table 3.3.10-1 directs entry into Condition F, the plant must be brought to a MODE in which the requirements of this LCO do 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.

G.1

Alternate means of monitoring containment area radiation have been developed and tested. These alternate means may be temporarily installed if the normal PAM channel cannot be restored to OPERABLE status within the allotted time. The reactor vessel water level monitoring system is one of three components of the inadequate core cooling instrumentation. The SMMs and core exit thermocouples could be used to monitor inadequate core cooling. If these alternate means are used, the Required Action is not to shut down the plant, but rather to follow the directions of Specification 5.6.7. 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.

SURVEILLANCE REQUIREMENTS

A Note at the beginning of the SRs specifies that the following SRs apply to each PAM instrumentation Function in Table 3.3.10-1.

SR 3.3.10.1

Performance of the CHANNEL CHECK once every 31 days ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one indication channel to a similar parameter on other channels. It is based on the assumption that indication channels monitoring the same parameter should read approximately the same value. Significant deviations between the two indication channels could be an indication of excessive instrument drift in one of the channels or of something 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 plant staff, based on a qualitative assessment of the indication channel that considers indication channel 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. If the channels are within the criteria, it is an indication that the channels are OPERABLE. If the channels are normally off-scale during times when surveillance testing is required, the CHANNEL CHECK will only verify that they are off-scale in the same direction. Off-scale low current loop channels are verified to be reading at the bottom of the range and not failed down-scale.

For the Hydrogen Monitors, a CHANNEL CHECK is performed by drawing a sample from the Waste Gas System through the monitor.

The Frequency of 31 days is based upon plant operating experience with regard to channel OPERABILITY and drift, which demonstrates that failure of more than one indication channel of a given Function in any 31 day interval is a rare event. The CHANNEL CHECK supplements less formal, but more frequent, checks of channel during normal operational use of the displays associated with this LCO's required channels.

SR 3.3.10.2

A CHANNEL CALIBRATION is performed every 46 days on a staggered test basis for the Containment Hydrogen Analyzers. The CHANNEL CALIBRATION is performed using sample gases in accordance with manufacturer's recommendations.

SR 3.3.10.3

A CHANNEL CALIBRATION is performed every 24 months or approximately every refueling. CHANNEL CALIBRATION is a check of the indication channel including the sensor. The SR verifies the channel responds to the measured parameter within the necessary range and accuracy. A Note allows exclusion of neutron detectors, Core Exit Thermocouples, and Reactor Vessel Level Monitor System from the CHANNEL CALIBRATION.

The Frequency is based upon operating experience and consistency with the typical industry refueling cycle and is justified by an 24 month calibration interval for the determination of the magnitude of equipment drift.

REFERENCES

1. Letter from Mr. R. E. Denton (BGE) to NRC Document Control Desk, dated June 6, 1995, "License Amendment Request; Extension of Instrument Surveillance Intervals"
2. Letter from Mr. J. A. Tiernan (BGE) to NRC Document Control Desk, dated August 9, 1988, "Regulatory Guide 1.97 Review Update"
3. Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants To Assess Plant and Environs Conditions During and Following an Accident (Errata Published July 1981), December 1975
4. NUREG-0737, Supplement 1, Requirements for Emergency Response Capabilities (Generic Letter 82-33), December 17, 1982
5. UFSAR, Chapter 7, "Instrumentation and Control"

B 3.3 INSTRUMENTATION

B 3.3.11 Remote Shutdown Instrumentation

BASES

BACKGROUND

The Remote Shutdown Instrumentation provides the control room operator with sufficient instrumentation to help place and maintain 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 AFW System and the steam generator safety valves or the steam generator atmospheric dump valves 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 RCS from outside the Control Room allow extended operation in MODE 3.

In the event that the Control Room becomes inaccessible, the operators can establish control outside the control room and monitor remote shutdown instrumentation at the remote shutdown panel and place and maintain the unit in MODE 3. 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 Instrumentation Functions ensures that there is sufficient information available on selected plant parameters to place and maintain the plant in MODE 3, should the Control Room become inaccessible.

APPLICABLE SAFETY ANALYSES

The Remote Shutdown Instrumentation is required to provide equipment at appropriate locations outside the Control Room to help operators promptly shut down and maintain the plant in a safe condition in MODE 3.

Remote Shutdown Instrumentation assists in meeting the requirements of Reference 1.

The Remote Shutdown Instrumentation does not meet any of the criteria in 10 CFR 50.36(c)(2)(ii) but has been retained at the request of the NRC.

BASES

LCO

The Remote Shutdown Instrumentation LCO provides the requirements for the OPERABILITY of the instrumentation necessary to help place and maintain the unit in MODE 3 from a location other than the Control Room.

The instrumentation typically required are listed in Table 3.3.11-1 in the accompanying LCO.

The instrumentation are those required for:

- Core Reactivity Monitoring (initial and long term);
- RCS Pressure Monitoring;
- Monitoring Decay Heat Removal via Steam Generators; and
- RCS Inventory Monitoring.

A Function of a Remote Shutdown Instrumentation is OPERABLE if all indication channels needed to support the remote shutdown Functions are OPERABLE. In some cases, Table 3.3.11-1 may indicate that the required information capability is available from several alternate sources. In these cases, the Function is OPERABLE as long as one channel of any of the alternate information sources for each Function is OPERABLE.

An indication channel consists of field transmitters or process sensors and associated instrumentation, providing a measurable electronic signal based upon the physical characteristics of the parameter being measured, plus a display of the measured parameter.

The Remote Shutdown Instrumentation circuits covered by this LCO do not need to be energized to be considered OPERABLE. This LCO is intended to ensure that the instrument circuits will be OPERABLE if plant conditions require that the Remote Shutdown Instrumentation be placed in operation.

APPLICABILITY

The Remote Shutdown 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 unit is already subcritical and in the condition of reduced RCS energy. Under these conditions, considerable time is available to restore necessary instrument Functions if control room instruments become unavailable.

ACTIONS

A Note has been included that excludes the MODE change restrictions of LCO 3.0.4. This exception allows entry into an applicable MODE while relying on the ACTIONS, even though the ACTIONS may eventually require a plant shutdown. This is acceptable due to the low probability of an event requiring this system. The Remote Shutdown Instrumentation equipment can generally be repaired during operation without significant risk of spurious trip.

A Remote Shutdown Instrumentation Function is inoperable when the Function is not accomplished by at least one designated Remote Shutdown Instrumentation channel that satisfies the OPERABILITY criteria for each Function requirement, except Manual Reactor Shutdown Control, which requires two channels. These criteria are outlined in the LCO section of the Bases.

Note 2 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 Table 3.3.11-1. 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.

<u>A.1</u>

Condition A addresses the situation where one or more Functions of the Remote Shutdown System are inoperable. This includes any Function listed in Table 3.3.11-1.

The Required Action is to restore the Functions 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 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 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required MODE from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.3.11.1

Performance of the CHANNEL CHECK once every 31 days ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one indication channel to a similar parameter on other channels. It is based on the assumption that indication channels monitoring the same parameter should read approximately the same value. Significant deviations between the indication channels could be an indication of excessive instrument drift in one of the channels or of something 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 plant staff, based on a qualitative assessment of the indication channel that considers indication 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. As specified in the SR, a CHANNEL CHECK is only required for those channels that are normally energized. If the channels are within the criteria, it is an indication that the channels are OPERABLE. If the channels are normally off-scale during times when surveillance testing is required, the CHANNEL CHECK will only verify that they are off-scale in the same direction. Off-scale low current loop channels are verified to be reading at the bottom of the range and not failed down-scale.

BASES

The Frequency is based on plant operating experience that demonstrates indication channel failure is rare.

SR 3.3.11.2

CHANNEL CALIBRATION is a check of the indication channel including the sensor. The surveillance test verifies that the channel responds to the measured parameter within the necessary range and accuracy.

The 24-month Frequency is based upon the need to perform this SR under the conditions that apply during a plant outage, and the potential for an unplanned transient if the surveillance test were to be performed with the reactor at power.

The SR is modified by a Note, which excludes neutron detectors and reactor trip breaker indication from the CHANNEL CALIBRATION.

REFERENCES

1. Updated Final Safety Analysis Report, Appendix 1C, "AEC Proposed General Design Criteria for Nuclear Power Plants"

B 3.3 INSTRUMENTATION

B 3.3.12 Wide Range Logarithmic Neutron Flux Monitor Channels

BASES

BACKGROUND

The wide range logarithmic neutron flux monitor channels provide neutron flux power indication from < 1E-7% RATED THERMAL POWER to > 100% RATED THERMAL POWER. They also provide reactor protection when the RTCBs are shut, in the form of a Rate of Change of Power-High trip.

This LCO addresses MODEs 3, 4, and 5 with the RTCBs open. When the RTCBs are shut, the wide range logarithmic neutron flux monitor channels are addressed by LCO 3.3.2.

When the RTCBs are open, two of the four wide range logarithmic neutron flux monitor channels must be available to monitor neutron flux power. In this application, the RPS channels need not be OPERABLE since the reactor trip Function is not required. By monitoring neutron flux power when the RTCBs are open, loss of SHUTDOWN MARGIN (SDM) caused by boron dilution can be detected as an increase in flux. Alarms are also provided when power increases above the fixed bistable setpoints. Two channels must be OPERABLE to provide single failure protection and to facilitate detection of channel failure by providing CHANNEL CHECK capability.

APPLICABLE SAFETY ANALYSES

The wide range logarithmic neutron flux monitor channels are necessary to monitor core reactivity changes. They are the primary means for detecting and triggering operator actions to respond to reactivity transients initiated from conditions in which the RPS is not required to be OPERABLE. They also trigger operator actions to anticipate RPS actuation in the event of reactivity transients starting from shutdown or low power conditions.

The OPERABILITY of wide range logarithmic neutron flux monitor channels is not necessary to meet the assumptions of the safety analyses and provide for the mitigation of accident and transient conditions.

The wide range logarithmic neutron flux monitor channels satisfy 10 CFR 50.36(c)(2)(ii), Criterion 3.

BASES

LCO

The LCO on the wide range logarithmic neutron flux monitor channels ensures that adequate information is available to verify core reactivity conditions while shut down. A minimum of two wide range logarithmic neutron flux monitor channels are required to be OPERABLE.

APPLICABILITY

In MODEs 3, 4, and 5, with RTCBs open or the CEDM System not capable of CEA withdrawal, wide range logarithmic neutron flux monitor channels must be OPERABLE to monitor core power for reactivity changes. In MODEs 1 and 2, and in MODEs 3, 4, and 5 with the RTCBs shut and the CEAs capable of withdrawal, the wide range logarithmic neutron flux monitor channels are addressed as part of the RPS in LCO 3.3.1.

The requirements for source range neutron flux monitoring in MODE 6 are addressed in LCO 3.9.2. The source range nuclear instrumentation channels provide neutron flux coverage the logarithmic channels use during refueling, when neutron flux may be extremely low. They are built into the wide range logarithmic neutron flux channels and PAM channels.

ACTIONS

A.1 and A.2

With one required channel inoperable, it may not be possible to perform a CHANNEL CHECK to verify that the other required channel is OPERABLE. Therefore, with one or more required channels inoperable, the wide range logarithmic neutron flux monitoring Function cannot be reliably performed. Consequently, the Required Actions are the same for one required channel inoperable or more than one required channel inoperable. The absence of reliable neutron flux indication makes it difficult to ensure SDM is maintained. Required Action A.1, therefore, requires that all positive reactivity additions that are under operator control, such as boron dilution or RCS temperature changes, be halted immediately, preserving SDM.

SHUTDOWN MARGIN must be verified periodically to ensure that it is being maintained. Both required channels must be restored as soon as possible. The initial Completion Time of 4 hours, and once every 12 hours thereafter, to perform SDM verification takes into consideration that Required Action A.1 eliminates many of the means by which SDM can be

BASES

reduced. These Completion Times are also based on operating experience in performing the Required Actions and the fact that plant conditions will change slowly.

SURVEILLANCE REQUIREMENTS

SR 3.3.12.1

Surveillance Requirement 3.3.12.1 is the performance of a CHANNEL CHECK on each required channel every 12 hours. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based upon the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between instrument channels could be an indication of excessive instrument drift in one of the channels or of something 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.

Agreement criteria are determined by the plant staff and should be based on a qualitative assessment of the indication channel that considers indication channel uncertainties, including control isolation, indication, and readability. If a channel is outside the criteria, it may be an indication that the transmitter or the signal processing equipment has drifted outside its limits. If the channels are within the criteria, it is an indication that the channels are OPERABLE.

The Frequency, about once every shift, is based on operating experience that demonstrates the rarity of channel failure. Since the probability of two random failures in redundant channels in any 12-hour period is extremely low, CHANNEL CHECK minimizes the chance of loss of indication due to failure of redundant channels. CHANNEL CHECK supplements less formal, but more frequent, checks of channel OPERABILITY during normal operational use of displays associated with the LCO required channels.

BASES

SR 3.3.12.2

A CHANNEL FUNCTIONAL TEST is performed once within 7 days prior to each reactor startup. This SR ensures that the entire channel is capable of properly indicating neutron flux. Internal test circuitry is used to feed pre-adjusted test signals into the preamplifier to verify channel alignment. It is not necessary to test the detector, because generating a meaningful test signal is difficult; the detectors are of simple construction, and any failures in the detectors will be apparent as change in channel output. This Frequency is the same as that employed for the same channels in the other applicable MODEs.

SR 3.3.12.3

Surveillance Requirement 3.3.12.3 is the performance of a CHANNEL CALIBRATION. A CHANNEL CALIBRATION is performed every 24 months. The surveillance test is a complete check and readjustment of the wide range logarithmic neutron flux monitor channel from the preamplifier input through to the remote indicators. The surveillance test verifies that the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift between successive calibrations to ensure that the channel remains operational between successive surveillance tests. CHANNEL CALIBRATIONS must be performed consistent with the plant-specific setpoint analysis.

This SR is modified by a Note to indicate that it is not necessary to test the detector because generating a meaningful test signal is difficult; the detectors are of simple construction, and any failures in the detectors will be apparent as change in channel output. This Frequency is the same as that employed for the same channels in the other applicable MODEs.

REFERENCES None

B 3.4 REACTOR COOLANT SYSTEM (RCS)

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

BASES

BACKGROUND

These Bases address requirements for maintaining RCS pressure, temperature, and flow rate within limits assumed in the safety analyses. The safety analyses (Reference 1) of normal operating conditions and anticipated operational occurrences, assume initial conditions within the normal steady-state envelope. The limits placed on departure from nucleate boiling (DNB) related parameters ensure that these parameters will not be less conservative than were assumed in the analyses, and thereby provide assurance that the minimum departure from nucleate boiling ratio (DNBR) will meet the required criteria for each of the transients analyzed.

The Limiting Condition for Operation (LCO) limit for minimum RCS pressure as measured at the pressurizer is consistent with operation within the nominal operating envelope and is bounded by the initial pressure in the analyses.

The LCO limit for maximum RCS cold leg temperature is consistent with operation at the indicated power level and is bounded by the initial temperature in the analyses.

The LCO limit for minimum RCS flow rate is bounded by the initial flow rate in the analyses. The RCS flow rate is not expected to vary during plant operation with all pumps running.

APPLICABLE SAFETY ANALYSES

The requirements of LCO 3.4.1 represent the initial conditions for DNB limited transients analyzed in the safety analyses (Reference 1). The safety analyses have shown that transients initiated from the limits of this LCO will meet the DNBR criterion. Changes to the facility that could impact these parameters must be assessed for their impact on the DNBR criterion. The transients analyzed include loss of coolant flow events and dropped or stuck control element assembly 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, LCO 3.2.4, and LCO 3.2.5. The safety analyses are performed over the following range of initial

BASES

values: RCS pressure 2154-2300 psia, core inlet temperature ≤ 548°F, and reactor vessel inlet coolant flow rate ≥ 340,000 gpm**.

The RCS DNB limits satisfy 10 CFR 50.36(c)(2)(ii), Criterion 2.

LCO

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

The LCO numerical values for pressure and temperature (P/T) are given for the measurement location and have been adjusted for instrument error. Reactor Coolant System flow rate is given as an analytical value.

APPLICABILITY

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

A Note has been added to indicate the limit on pressurizer pressure may be exceeded during short-term operational transients such as a THERMAL POWER ramp increase of > 5% RATED THERMAL POWER (RTP) per minute or a THERMAL POWER step increase of > 10% RTP. These conditions represent short-term perturbations where actions to control pressure variations might be counterproductive. Also, since they represent transients initiated from power levels < 100% RTP, an increased DNBR margin exists to offset the temporary pressure variations.

** The Reactor Coolant System Total Flow Rate limit shall be ≥ 370,000 gpm through Unit 2, Cycle 12.

BASES

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

ACTIONS

A.1

Pressurizer pressure and RCS cold leg temperature are controllable and measurable parameters. Reactor Coolant System flow rate is not a controllable parameter and is not expected to vary during steady-state operation. With any parameter not within its LCO limit, action must be taken to restore the parameter.

The two hour Completion Time for restoration of the parameters provides sufficient time to adjust plant parameters, to determine the cause of the off normal condition, and to restore the readings within limits. The Completion Time is based on plant operating experience that shows the parameter can be restored in this time period.

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 six hours. In MODE 2, the reduced power condition eliminates the potential for violation of the accident analysis bounds.

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

SURVEILLANCE REQUIREMENTS

SR 3.4.1.1

Since Required Action A.1 allows a Completion Time of two hours to restore parameters that are not within limits, the 12 hour SR Frequency for pressurizer pressure is sufficient to ensure that the pressure can be restored to a normal operation, steady-state condition following load changes, and other expected transient operations. The

BASES

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

SR 3.4.1.2

Since Required Action A.1 allows a Completion Time of two hours to restore parameters that are not within limits, the 12 hour SR Frequency for cold leg temperature is sufficient to ensure that the RCS coolant temperature can be restored to a normal operation, steady-state condition following load changes, and other expected transient operations. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess for potential degradation and to verify operation is within safety analysis assumptions.

SR 3.4.1.3

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

SR 3.4.1.4

Measurement of RCS total flow rate is performed once every 24 months. This verifies that the actual RCS flow rate is within the bounds of the analyses.

The Frequency of 24 months reflects the importance of verifying flow after a refueling outage where the core has been altered, which may have caused an alteration of flow resistance.

REFERENCES

1. Updated Final Safety Analysis Report (UFSAR), Section 14.1.2, "Plant Characteristics Considered in Safety Analysis"

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.2 RCS Minimum Temperature for Criticality

BASES

BACKGROUND

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

a. Operation within the existing instrumentation ranges and accuracies;

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

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

The reactor coolant moderator temperature coefficient used in core operating and accident analysis is defined for the normal operating temperature range, as specified in the operating procedures. The Reactor Protective System (RPS) receives inputs from the narrow range hot and cold leg temperature detectors, which have a range of 515°F to 665°F and 465°F to 615°F, respectively. The RCS temperature is controlled using inputs of the same range. Nominal T_{avg} for making the reactor critical is 532°F. Safety and operating analyses for lower temperature have not been made.

APPLICABLE SAFETY ANALYSES

There are no accident analyses that dictate the minimum temperature for criticality, but all low power safety analyses can accommodate initial temperatures near the 515°F limit (References 1 and 2).

The RCS minimum temperature for criticality satisfies 10 CFR 50.36(c)(2)(ii), Criterion 2.

LCO

The purpose of the LCO is to prevent criticality outside the normal operating regime and to prevent operation in an unanalyzed condition.

APPLICABILITY

The reactor has been designed and analyzed to be critical in MODEs 1 and 2 only and in accordance with this specification. Criticality is not permitted in any other MODE. Therefore, this LCO is applicable in MODE 1 and MODE 2 when $K_{eff} \geq 1.0$.

BASES

ACTIONS

A.1

If T_{avg} is below 515°F, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 2 with $K_{eff} < 1.0$ within 30 minutes. Rapid reactor shutdown can be readily and practically achieved within a 30 minute period. The allowed time reflects the ability to perform this action and to maintain the plant within the analyzed range.

SURVEILLANCE REQUIREMENTS

SR 3.4.2.1

T_{avg} is initially required to be verified ≥ 515°F within 30 minutes prior to reaching reactor criticality, then T_{avg} is required to be verified ≥ 515°F every 30 minutes. The 30 minute time period is frequent enough to prevent inadvertent violation of the LCO. The second frequency is modified by a Note which states that the ~~S~~surveillance test is only required to be performed when RCS T_{avg} is less than 525°F. This provides a reasonable distance to the limit of 515°F. Adequate time will be available to trend RCS T_{avg} as it approaches 515°F, and take corrective action(s) prior to exceeding the limit.

REFERENCES

1. UFSAR, Chapter ~~Section~~ 14, "Safety Analysis"
2. Combustion Engineering Report CENPSD-1026, Evaluations of NRC Information Notice 94-75

B 3.4 REACTOR COOLANT SYSTEM (RCS)

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

BASES

BACKGROUND

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

Figures 3.4.3-1 and 3.4.3-2 contains P/T limit curves for heatup, cooldown, and inservice leak and hydrostatic (ISLH) testing, and data for the maximum rate of change of reactor coolant temperature (these P/T limits do not apply to the pressurizer).

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 P/T indications are monitored and compared to the applicable curve to determine that operation is within the allowable region.

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

Reference 1, Appendix G requires the establishment of P/T limits for material fracture toughness requirements of the RCPB materials. Reference 1, Appendix G requires an adequate margin to brittle failure during normal operation, anticipated operational occurrences, and system hydrostatic tests. It mandates the use of the Reference 2, Section III, Appendix G.

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

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with References 1 (Appendix H) and 3. The operating P/T limit curves will be adjusted, as necessary, based on the evaluation findings and the recommendations of Reference 2, Section III, Appendix G.

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.

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 can alter the location of the tensile stress between the outer and inner walls.

The criticality limit includes the Reference 1, Appendix G requirement that the limit be no less than 40°F above the heatup curve or the cooldown curve and not less than the minimum permissible temperature for the ISLH testing. However, the criticality limit is not operationally limiting; a more restrictive limit exists in LCO 3.4.2.

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 (LOCA). In the event these limits are exceeded, an evaluation must be performed to determine the effect on the structural integrity of the RCPB components. Reference 2, Section XI, Appendix E, provides a recommended methodology for evaluating an operating event that causes an excursion outside the limits.

APPLICABLE SAFETY ANALYSES

The P/T limits are not derived from Design Basis Accident (DBA) Analyses. They are prescribed during normal operation to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and cause nonductile failure of the RCPB, an unanalyzed condition. Since the P/T limits are not derived from any DBA, there are no acceptance limits related to the P/T limits. Rather, the P/T limits are acceptance limits

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themselves, since they preclude operation in an unanalyzed condition.

The RCS P/T limits satisfy 10 CFR 50.36(c)(2)(ii), Criterion 2.

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.

The P/T limits are corrected for instrument uncertainty, and for static and dynamic head between the limiting material location and the pressurizer. The limits assume not more than the following number of RCPs are running:

Heatup

RCS Temperature (Unit 1)	RCS Temperature (Unit 2)	Number of RCPs
70°F to 330°F	70°F to 308°F	2
> 330°F	> 308°F	4

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Cooldown

RCS Temperature (Unit 1)	RCS Temperature (Unit 2)	Number of RCPs
> 350°F	> 350°F	4
350°F to 150°F	350°F to 150°F	2
< 150°F	< 150°F	0

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

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

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

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

APPLICABILITY

The RCS P/T limits specification provides a definition of acceptable operation for prevention of nonductile failure in accordance with Reference 1, Appendix G. 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, LCO 3.4.2, and SL 2.1, also provide operational restrictions for P/T and maximum pressure. Furthermore, MODEs 1 and 2 are above the temperature range of concern for nonductile failure, and stress analyses have been performed for normal maneuvering profiles, such as power ascension or descent.

The actions of this LCO consider the premise that a violation of the limits occurred during normal plant maneuvering. Severe violations caused by abnormal

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transients, at times accompanied by equipment failures, may also require additional actions from Emergency Operating Procedures.

ACTIONS

A.1 and A.2

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

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

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

Reference 5, Section XI, Appendix E 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

B.1 and B.2

If a Required Action and associated Completion Time of Condition A are not met, the plant must be placed in a lower MODE because:

a. The RCS remained in an unacceptable P/T region for an extended period of increased stress; or

b. A sufficiently severe event caused entry into an unacceptable region.

Either possibility indicates a need for more careful examination of the event, best accomplished with the RCS at reduced P/T. With reduced P/T conditions, the possibility of propagation of undetected flaws is decreased.

Pressure and temperature are reduced by placing the plant in MODE 3 within 6 hours and in MODE 5 with RCS pressure < 300 psia within 36 hours.

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

C.1 and C.2

The actions of this LCO, anytime other than in MODEs 1, 2, 3, or 4, consider the premise that a violation of the limits occurred during normal plant maneuvering. Severe violations caused by abnormal transients, at times accompanied by equipment failures, may also require additional actions from Emergency Operating Procedures. Operation outside the P/T limits must be corrected so that the RCPB is returned to a condition that has been verified by stress analyses.

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

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

Reference 5, Section XI, Appendix E, may be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline.

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

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

SURVEILLANCE REQUIREMENTS

SR 3.4.3.1

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

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

BASES

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

REFERENCES

1. 10 CFR Part 50, Domestic Licensing of Production and Utilization Facilities
2. American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel Code
3. American Society for Testing Materials E 185-82, July 1982

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.4 RCS LOOPS - MODEs 1 and 2

BASES

BACKGROUND

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

The secondary functions of the RCS include:

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

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

c. Carrying the soluble neutron poison, boric acid;

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

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

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

APPLICABLE SAFETY ANALYSES

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

Both transient and steady-state analyses have been performed to establish the effect of flow on DNB. The transient or accident analysis for the plant has been performed assuming four RCPs are in operation. The majority of the plant

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safety analyses are based on initial conditions at high core power or zero power. The accident analyses that are of most importance to RCP operation are loss of coolant flow and seized rotor (Reference 1).

RCS Loops - MODEs 1 and 2 satisfy 10 CFR 50.36(c)(2)(ii), Criteria 2 and 3.

LCO

The purpose of this LCO is to require adequate forced flow for core heat removal. Flow is represented by having both RCS loops with both RCPs in each loop in operation for removal of heat by the two SGs. To meet safety analysis acceptance criteria for DNB, four pumps are required at rated power.

Each OPERABLE loop consists of two RCPs providing forced flow for heat transport to an SG that is OPERABLE in accordance with the Steam Generator Tube Surveillance Program. Steam generator, and hence RCS loop, OPERABILITY with regard to SG water level is ensured by the RPS in MODEs 1 and 2. A reactor trip places the plant in MODE 3 if any SG level is ≥ 10 inches below the top of the feed ring as sensed by the RPS. The minimum water level to declare the SG OPERABLE is < 10 inches below the top of the feed ring.

APPLICABILITY

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

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

Operation in other MODEs is covered by: LCO 3.4.5, LCO 3.4.6, LCO 3.4.7, LCO 3.4.8, LCO 3.9.4, and LCO 3.9.5.

BASES

ACTIONS

A.1

If the requirements of the LCO are not met, the Required Action is to reduce power and bring the plant to MODE 3. This lowers power level and thus reduces the core heat removal needs, and minimizes the possibility of violating DNB limits. It should be noted that the reactor will trip and place the plant in MODE 3 as soon as the RPS senses less than 340,000 gpm** RCS flow.

The Completion Time of six hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging safety systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.4.1

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

REFERENCES

1. UFSAR, Chapter 14, "Safety Analysis"

** The Reactor Coolant System Flow Rate limit shall be ≥ 370,000 gpm through Unit 2, Cycle 12.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.5 RCS Loops - MODE 3

BASES

BACKGROUND

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

In MODE 3, RCPs are used to provide forced circulation heat removal during heatup and cooldown. The MODE 3 decay heat removal requirements are low enough that a single RCS loop with one RCP is sufficient to remove core decay heat. However, two RCS loops (i.e., RCS loop Nos. 11 and 12 for Unit 1 and RCS loop Nos. 21 and 22 for Unit 2) are required to be OPERABLE to provide redundant paths for decay heat removal. Only one RCP needs to be OPERABLE to declare the associated RCS loop OPERABLE.

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

APPLICABLE SAFETY ANALYSES

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

Reactor Coolant System Loops - MODE 3 satisfy 10 CFR 50.36(c)(2)(ii), Criterion 3.

LCO

The purpose of this LCO is to require two RCS loops to be available for heat removal, thus providing redundancy. The LCO requires the two loops to be OPERABLE with the intent of requiring both SGs to be capable (> -50 inches water level) of transferring heat from the reactor coolant at a controlled rate. Forced reactor coolant flow is the required way to transport heat, although natural circulation

BASES

flow provides adequate removal. A minimum of one running RCP meets the LCO requirement for one loop in operation.

Note 1 permits a limited period of operation without RCPs. All RCPs may not be in operation for ≤ 1 hour per eight hour period and ≤ 2 hours per eight hour period for low flow testing. This means that natural circulation has been established. When in natural circulation, a reduction in boron concentration is prohibited because an even concentration distribution throughout the RCS cannot be ensured. Core outlet temperature is to be maintained at least 10°F below the saturation temperature so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

In MODE 3, it is sometimes necessary to stop all RCPs (e.g., to perform surveillance or startup testing). The time period is acceptable because natural circulation is adequate for heat removal and the reactor coolant temperature can be maintained subcooled.

Note 2 requires that all of the following three conditions be satisfied before an RCP can be started when any RCS cold leg temperature is ≤ 365°F (Unit 1), ≤ 301°F (Unit 2):

a. the pressurizer water level is ≤ 170 inches;

b. the pressurizer pressure is ≤ 300 psia (Unit 1), ≤ 320 psia (Unit 2); and

c. the secondary water temperature of each SG is ≤ 30°F above the RCS T_{avg}.

Ensuring the above conditions are satisfied will preclude a power-operated relief valve (PORV) from opening as a result of the pressure surge in the RCS, when an RCP is started.

An OPERABLE loop consists of at least one OPERABLE RCP and an SG that is OPERABLE in accordance with the Steam Generator Tube Surveillance Program. An RCP is OPERABLE if it is capable of being powered and is able to provide forced flow, if required.

BASES

APPLICABILITY

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

Operation in other MODEs is covered by: LCO 3.4.4, LCO 3.4.6, LCO 3.4.7, LCO 3.4.8, LCO 3.9.4, and LCO 3.9.5.

ACTIONS

A.1

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

B.1

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

C.1 and C.2

If no RCS loop is in operation, except as provided in Note 1 in the LCO section, all operations involving a reduction of RCS boron concentration must be immediately suspended. This is necessary because boron dilution requires forced circulation for proper homogenization. Action to restore one RCS loop to OPERABLE status and operation shall be initiated immediately and continued until one RCS loop is restored to OPERABLE status and operation. The immediate Completion Times reflect the importance of maintaining operation for decay heat removal.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.5.1

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

SR 3.4.5.2

This SR requires verification every 12 hours that the secondary side water level in each SG is > -50 inches. An adequate SG water level is required in order to have a heat sink for removal of the core decay heat from the reactor coolant. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess degradation and verify operation within the safety analyses assumptions.

SR 3.4.5.3

Verification that the required number of RCPs are OPERABLE ensures that the single failure criterion is met and that an additional RCS loop can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power availability to the required RCPs. The Frequency of seven days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

REFERENCES None

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.6 RCS Loops - MODE 4

BASES

BACKGROUND

In MODE 4, the primary function of the reactor coolant is the removal of decay heat and transfer of this heat to the SGs or SDC heat exchangers. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid.

In MODE 4, either RCPs or SDC loops can be used for coolant circulation. The intent of this LCO is to provide forced flow from at least one RCP or one SDC loop for decay heat removal and transport. The flow provided by one RCP or SDC loop is adequate for heat removal. The other intent of this LCO is to require that two paths be available to provide redundancy for heat removal. For Unit 1, the two paths can be any combination of RCS loop No. 11, RCS loop No. 12, SDC loop No. 11, or SDC loop No. 12. For Unit 2, the two paths can be any combination of RCS loop No. 21, RCS loop No. 22, SDC loop No. 21, or SDC loop No. 22.

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 SDC loops provide this circulation.

Reactor Coolant System loops - MODE 4 have been identified in 10 CFR 50.36(c)(2)(ii) as important contributors to risk reduction.

LCO

The purpose of this LCO is to require that at least two loops, RCS or SDC, be OPERABLE in MODE 4, and one of these loops be in operation. The LCO allows the two loops that are required to be OPERABLE to consist of any combination of RCS and SDC System 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 and SDC pumps to not be in operation $\leq$ 1 hour per eight hour period. The Note prohibits boron dilution when forced flow is stopped because an even concentration distribution cannot be ensured. Core outlet temperature is to be maintained at least 10°F below saturation temperature so that no vapor bubble may form and possibly cause a natural circulation flow obstruction. The response of the RCS without the RCPs or SDC pumps depends on the core decay heat load and the length of time that the pumps are stopped. As decay heat diminishes, the effects on RCS temperature and pressure diminish. Without cooling by forced flow, higher heat loads will cause the reactor coolant temperature and pressure to increase at a rate proportional to the decay heat load. Because pressure can increase, the applicable system pressure limits [P/T limits or low temperature overpressure protection (LTOP) limits] must be observed and forced SDC flow or heat removal via the SGs must be re-established prior to reaching the pressure limit. The circumstances for stopping both RCPs or SDC pumps are to be limited to situations where:

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

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

Note 2 requires that the following conditions be satisfied before an RCP may be started with any RCS cold leg temperature ≤ 365°F (Unit 1), ≤ 301°F (Unit 2):

a. Pressurizer water level is ≤ 170 inches;

b. Pressurizer pressure is ≤ 300 psia (Unit 1), ≤ 320 psia (Unit 2); and

c. Secondary side water temperature in each SG is ≤ 30°F above each of the RCS cold leg temperatures.

Satisfying the above conditions will preclude a PORV from opening due to a pressure surge in the RCS when the RCP is started.

An OPERABLE RCS loop consists of at least one OPERABLE RCP and an SG that is OPERABLE in accordance with the Steam Generator Tube Surveillance Program, and has the minimum water level specified in SR 3.4.6.2.

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Similarly, for the SDC System, an OPERABLE SDC loop is composed of the OPERABLE SDC pump(s) capable of providing forced flow to the SDC heat exchanger(s). Reactor coolant pumps and SDC pumps are OPERABLE if they are capable of being powered and are able to provide flow if required.

APPLICABILITY

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

Operation in other MODEs is covered by: LCO 3.4.4, LCO 3.4.5, LCO 3.4.7, LCO 3.4.8, LCO 3.9.4, and LCO 3.9.5.

ACTIONS

A.1

If only one required RCS loop is OPERABLE and in operation, and no SDC loops are OPERABLE, redundancy for heat removal is lost. Action must be initiated immediately to restore a second loop to OPERABLE status. The immediate Completion Time reflects the importance of maintaining the availability of two paths for decay heat removal.

B.1

If one required SDC loop is OPERABLE and in operation and no RCS loops are OPERABLE, redundancy for heat removal is lost. The plant must be placed in MODE 5 within the next 24 hours. Placing the plant in MODE 5 is a conservative action with regard to decay heat removal. With only one SDC loop OPERABLE, redundancy for decay heat removal is lost and, in the event of a loss of the remaining SDC loop, it would be safer to initiate that loss from MODE 5 (≤ 200°F) rather than MODE 4 (> 200°F to < 300°F). The Completion Time of 24 hours is reasonable, based on operating experience, to reach MODE 5 from MODE 4, with only one SDC loop operating, in an orderly manner and without challenging plant systems.

C.1 and C.2

If no RCS or SDC loops are OPERABLE or in operation, except during conditions permitted by Note 1 in the LCO section, all operations involving reduction of RCS boron concentration must be suspended and action to restore one RCS or SDC loop to OPERABLE status and operation must be initiated. Boron dilution requires forced circulation for proper mixing, and the margin to criticality must not be reduced in this type of operation. The immediate Completion Times reflect the importance of decay heat removal. The action to restore must continue until one loop is restored to operation.

SURVEILLANCE REQUIREMENTS

SR 3.4.6.1

This SR requires verification every 12 hours that one required loop is in operation. This ensures forced flow is providing heat removal. Verification includes flow rate, temperature, or pump status monitoring. The 12 hour Frequency has been shown by operating practice to be sufficient to regularly assess RCS loop status. In addition, Control Room indication and alarms will normally indicate loop status.

SR 3.4.6.2

This SR requires verification every 12 hours of secondary side water level in the required SG(s) > -50 inches. An adequate SG water level is required in order to have a heat sink for removal of the core decay heat from the reactor coolant. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess degradation and verify operation within safety analyses assumptions.

SR 3.4.6.3

Verification that the required pump is OPERABLE ensures that an additional RCS or SDC loop can be placed in operation, if needed to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the required loop components that are not in operation. For an RCS loop, the required component is a pump. For an SDC loop, the required components are the pump and valves. The Frequency of seven days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

REFERENCES

None

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.7 RCS Loops - MODE 5, Loops Filled

BASES

BACKGROUND

In MODE 5 with the RCS loops filled, the primary function of the reactor coolant is the removal of decay heat, and the transfer of this heat either to the SG secondary side coolant, or the component cooling water via the SDC heat exchangers. While the principal means for decay heat removal is via the SDC System, the SGs are specified as a backup means for redundancy. Even though the SGs cannot produce steam in this MODE, they are capable of being a heat sink due to their large contained volume of secondary side water. As long as the SG secondary side water is at a lower temperature than the reactor coolant, heat transfer will occur. The rate of heat transfer is directly proportional to the temperature difference. Due to the non-condensable gasses that come out of solution and restrict flow through the SG tubes, the SGs can only be credited when the RCS is capable of being pressurized. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid.

In MODE 5 with RCS loops filled, the SDC loops are the principal means for decay heat removal. The number of loops in operation can vary to suit the operational needs. The intent of this LCO is to provide forced flow from at least one SDC loop for decay heat removal and transport. The flow provided by one SDC loop is adequate for decay heat removal. The other intent of this LCO is to require that a second path be available to provide redundancy for decay heat removal.

The LCO provides for redundant paths of decay heat removal capability. The first path can be an SDC loop (i.e., SDC loop No. 11 or No. 12 for Unit 1, and SDC loop No. 21 or No. 22 for Unit 2) that must be OPERABLE and in operation. The second path can be another OPERABLE SDC loop (i.e., SDC loop No. 11 or No. 12 for Unit 1, and SDC loop No. 21 or No. 22 for Unit 2), or through the SGs, each having an adequate water level.

BASES

APPLICABLE SAFETY ANALYSES

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

Reactor Coolant System loops - MODE 5 (Loops Filled) have been identified in 10 CFR 50.36(c)(2)(ii) as important contributors to risk reduction.

LCO

The purpose of this LCO is to require at least one of the SDC loops be OPERABLE and in operation with an additional SDC loop OPERABLE, or secondary side water level of each SG shall be ≥ -50 inches. One SDC loop provides sufficient forced circulation to perform the safety functions of the reactor coolant under these conditions. The second SDC loop is normally maintained OPERABLE as a backup to the operating SDC loop, to provide redundant paths for decay heat removal. However, if the standby SDC loop is not OPERABLE, a sufficient alternate method to provide redundant paths for decay heat removal is two SGs with their secondary side water levels ≥ -50 inches. Should the operating SDC loop fail, the SGs could be used to remove the decay heat.

Note 1 permits all SDC pumps to not be in operation ≤ 1 hour per eight hour period. The circumstances for stopping both SDC loops are to be limited to situations where P/T increases can be maintained well within the allowable pressure (P/T and LTOP) and 10°F subcooling limits, or an alternate heat removal path through the SG(s) is in operation.

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

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

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

Note 3 requires that the following conditions be satisfied before an RCP may be started with any RCS cold leg temperature ≤ 365°F (Unit 1), ≤ 301°F (Unit 2):

a. Pressurizer water level must be ≤ 170 inches;

b. Pressurizer pressure ≤ 300 psia (Unit 1), ≤ 320 psia (Unit 2); and

c. Secondary side water temperature in each SG must be ≤ 30°F above each of the RCS cold leg temperatures.

Satisfying the above conditions will preclude opening a PORV during a pressure transient when the RCP is started.

Note 4 provides for an orderly transition from MODE 5 to MODE 4 during a planned heatup by permitting SDC loops to not be in operation when at least one RCP is in operation. This Note provides for the transition to MODE 4 where an RCP is permitted to be in operation and replaces the RCS circulation function provided by the SDC loops.

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

SDC pumps are OPERABLE if they are capable of being powered and are able to provide flow if required. An OPERABLE SG can perform as a heat sink when it has an adequate water

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

APPLICABILITY

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

Operation in other MODEs is covered by: LCO 3.4.4, LCO 3.4.5, LCO 3.4.6, LCO 3.4.8, LCO 3.9.4, and LCO 3.9.5.

ACTIONS

A.1 and A.2

If the required SDC loop is inoperable and any SGs have secondary side water levels < -50 inches, redundancy for heat removal is lost. Action must be initiated immediately to restore a second SDC loop to OPERABLE status or to restore the water level in the required SGs. Either Required Action A.1 or Required Action A.2 will restore redundant decay heat removal paths. The immediate Completion Times reflect the importance of maintaining the availability of two paths for decay heat removal.

B.1 and B.2

If no SDC loop is in operation, except as permitted in Note 1, all operations involving the reduction of RCS boron concentration must be suspended. Action to restore one SDC loop to OPERABLE status and place it in operation must be initiated. Boron dilution requires forced circulation for proper mixing and the margin to criticality must not be reduced in this type of operation. The immediate Completion Times reflect the importance of maintaining operation for decay heat removal.

SURVEILLANCE REQUIREMENTS

SR 3.4.7.1

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

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operation is within safety analyses assumptions. In addition, Control Room indication and alarms will normally indicate loop status.

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

SR 3.4.7.2

Verifying the SGs are OPERABLE by ensuring their secondary side water levels are ≥ -50 inches ensures that redundant heat removal paths are available if the second SDC loop is inoperable. This surveillance test is required to be performed when the LCO requirement is being met by use of the SGs. If both SDC loops are OPERABLE, this SR is not needed. The 12 hour Frequency has been shown by operating practice to be sufficient to regularly assess degradation and verify operation within safety analyses assumptions.

SR 3.4.7.3

Verification that the second SDC loop is OPERABLE ensures that redundant paths for decay heat removal are available. The requirement also ensures that the additional loop can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the required pumps and valves that are not in operation. This surveillance test is required to be performed when the LCO requirement is being met by one of two SDC loops, e.g., both SGs have < -50 inches water level. The Frequency of seven days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

REFERENCES None

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.8 RCS Loops - MODE 5, Loops Not Filled

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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 and transfer of this heat to the SDC heat exchangers. The SGs are not available as a heat sink when the loops are not filled. The secondary function of the reactor coolant is to act as a carrier for the soluble neutron poison, boric acid.

In MODE 5 with loops not filled, only the SDC System can be used for coolant circulation. The number of loops in operation can vary to suit the operational needs. The intent of this LCO is to provide forced flow from at least one SDC loop for decay heat removal and transport and to require that two paths (i.e., SDC loop No. 11 or No. 12 for Unit 1, and SDC loop No. 21 or No. 22 for Unit 2) be available to provide redundancy for heat removal.

APPLICABLE SAFETY ANALYSES

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

Reactor Coolant System loops - MODE 5 (loops not filled) satisfy 10 CFR 50.36(c)(2)(ii), Criterion 4.

LCO

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

Note 1 permits the SDC pumps to not be in operation for ≤ 15 minutes when switching from one loop to another. The circumstances for stopping both SDC pumps are to be limited to situations when the outage time is short and the core outlet temperature is maintained at least 10°F below

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outlet temperature is maintained at least 10°F below saturation temperature. The Note prohibits boron dilution or draining operations when SDC forced flow is stopped.

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

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

APPLICABILITY

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

Operation in other MODEs~~S~~ is covered by: LCO 3.4.4, LCO 3.4.5, LCO 3.4.6, LCO 3.4.7, LCO 3.9.4, and LCO 3.9.5.

ACTIONS

A.1

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

B.1 and B.2

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

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.8.1

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

SR 3.4.8.2

Verification that the required number of loops are OPERABLE ensures that redundant paths for heat removal are available and that additional loops can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and indicated power available to the required pumps and valves that are not in operation. The Frequency of seven days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

REFERENCES

None

B 3.4 REACTOR COOLANT SYSTEMS (RCS)

B 3.4.9 Pressurizer

BASES

BACKGROUND

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

The pressure control components addressed by this LCO include the pressurizer water level, the required heaters and their backup heater controls, and emergency power supplies. Pressurizer safety valves and pressurizer PORVs are addressed by LCO 3.4.10 and LCO 3.4.11, respectively.

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

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

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

The maximum water level limit permits pressure control equipment to function as designed. The limit preserves the steam space during normal operation, thus, both sprays and heaters can operate to maintain the design operating pressure. The level limit also prevents filling the pressurizer (water solid) for anticipated design basis transients, thus ensuring that pressure relief devices (PORVs or pressurizer safety valves) can control pressure by steam relief rather than water relief. If the level limits were exceeded prior to a transient that creates a large

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pressurizer insurge volume leading to water relief, the maximum RCS pressure might exceed the SL of 2750 psia.

The requirement to have two banks of pressurizer heaters, which are permanently powered by Class 1E power supplies, ensures that RCS pressure can be maintained. The pressurizer heaters maintain RCS pressure to keep the reactor coolant subcooled. Inability to control RCS pressure during natural circulation flow could result in loss of single phase flow and decreased capability to remove core decay heat.

APPLICABLE SAFETY ANALYSES

In MODEs 1, 2, and 3, the LCO requirement for a steam bubble is reflected implicitly in the accident analyses. 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 do not take credit for pressurizer heater operation; however, an implicit initial condition assumption of the safety analyses is that the RCS is operating at normal pressure.

Although the heaters are not specifically used in accident analysis, the need to maintain subcooling in the long-term during loss of offsite power, as indicated in Reference 1, is the reason for their inclusion. The requirement for emergency power supplies is based on Reference 1. The intent is to keep the reactor coolant in a subcooled condition using natural circulation at hot, high pressure conditions for an undefined, but extended, time period after a loss of offsite power. While loss of offsite power is a coincident occurrence assumed in the accident analyses, maintaining hot, high pressure conditions over an extended time period is not evaluated in the accident analyses.

The pressurizer satisfies 10 CFR 50.36(c)(2)(ii), Criteria 2 and 3.

LCO

The LCO requirement for the pressurizer to be OPERABLE with water level ≥ 133 inches and ≤ 225 inches ensures that a steam bubble exists. Limiting the maximum operating water level preserves the steam space for pressure control. The LCO has been established to minimize the consequences of potential overpressure transients. Requiring the presence of a steam bubble is also consistent with analytical assumptions.

The LCO requires two banks of OPERABLE pressurizer heaters, each with a capacity ≥ 150 kW and capable of being powered from an emergency power supply. The minimum heater capacity required is sufficient to maintain the RCS near normal operating pressure. By maintaining the pressure near the operating conditions, a wide subcooling margin to saturation can be obtained in the loops. The generic value of 150 kW is derived from the use of 12 heaters rated at 12.5 kW each. The amount needed to maintain pressure is dependent on the ambient heat losses.

APPLICABILITY

The need for pressure control is most pertinent when core heat can cause the greatest effect on RCS temperature resulting in the greatest effect on pressurizer level and RCS pressure control. Thus, Applicability has been designated for MODEs 1 and 2. The Applicability is also provided for MODE 3. The purpose is to prevent solid water RCS operation during heatup and cooldown to avoid rapid pressure rises caused by normal operational perturbation, such as RCP startup. The LCO does not apply to MODE 5 (Loops Filled) because LCO 3.4.12 applies. The LCO does not apply to MODEs 5 and 6 with partial loop operation.

In MODEs 1, 2, and 3, there is the need to maintain the availability of pressurizer heaters capable of being powered from an emergency power supply. In the event of a loss of offsite power, the initial conditions of these MODEs gives the greatest demand for maintaining the RCS in a hot pressurized condition with loop subcooling for an extended period. For MODEs 4, 5, or 6, it is not necessary to control pressure (by heaters) to ensure loop subcooling for heat transfer. When the SDC System is in service, this LCO is not applicable.

BASES

ACTIONS

A.1 and A.2

With pressurizer water level not within the limit, action must be taken to restore the plant to operation within the bounds of the safety analyses. To achieve this status, the unit must be brought to MODE 3, with the reactor trip breakers open, within 6 hours and to MODE 4 within 12 hours. This takes the plant out of the applicable MODEs and restores the plant to operation within the bounds of the safety analyses. Six hours is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging plant systems. Further P/T reduction to MODE 4 brings the plant to a MODE where the LCO is not applicable. The 12 hour time to reach the nonapplicable MODE is reasonable based on operating experience for that evolution.

B.1

If one required bank of pressurizer heaters is inoperable, restoration is required within 72 hours. The Completion Time of 72 hours is reasonable considering that a demand caused by loss of offsite power would be unlikely in this period. Pressure control may be maintained during this time using normal station powered heaters.

C.1 and C.2

If one required bank of pressurizer heaters is inoperable and cannot be restored within the allowed Completion Time of Required Action B.1, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and to MODE 4 within 12 hours. The Completion Time of six hours is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging safety systems. Similarly, the Completion Time of 12 hours is reasonable, based on operating experience to reach MODE 4 from full power to an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.9.1

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

SR 3.4.9.2

The SR is satisfied when the power supplies are demonstrated to be capable of producing the minimum power and the associated pressurizer heaters are verified to be at their design rating. (This may be done by testing the power supply output and by performing an electrical check on heater element continuity and resistance.) The Frequency of 24 months is considered adequate to detect heater degradation and has been shown by operating experience to be acceptable.

REFERENCES

1. NUREG-0737, II.E.3.1, "Clarification of TMI Action Plan Requirements," November 1980

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.10 Pressurizer Safety Valves

BASES

BACKGROUND

The purpose of the two spring loaded pressurizer safety valves is to provide RCS overpressure protection. Operating in conjunction with the RPS, two valves are used to ensure that the SL of 2750 psia is not exceeded for analyzed transients during operation in MODEs 1 and 2. Two safety valves are used for portions of MODE 3. For the remainder of MODEs 3, 4, 5, and 6 with the head on, overpressure protection is provided by operating procedures and LCO 3.4.12.

The self actuated pressurizer safety valves are designed in accordance with the requirements set forth in Reference 1, Section III. The required lift pressures are 2500 psia ± 1% and 2565 psia ± 1%. The safety valves discharge steam from the pressurizer to a quench tank located in the Containment Structure. The discharge flow is indicated by an increase in temperature downstream of the safety valves and by an increase in the quench tank temperature and level.

The upper and lower pressure limits are based on the ± 1%-tolerance requirement (Reference 1) for lifting pressures above 1000 psig. The lift setting is for the ambient conditions associated with normal operating 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 ASME pressure limit (Reference 1) could include damage to RCS components, increased leakage, or a requirement to perform additional stress analyses prior to resumption of reactor operation.

APPLICABLE SAFETY ANALYSES

All accident analyses in the UFSAR that require safety valve actuation, assume operation of both pressurizer safety valves to limit increasing reactor coolant pressure. The overpressure protection analysis is also based on operation

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of both safety valves and assumes that the valves open at the high range of the as found setting. These valves must accommodate pressurizer insurges that could occur during a loss of load, loss of main feedwater, or main feedwater line break accident. The startup accident establishes the minimum safety valve capacity. Single failure of a safety valve is neither assumed in the accident analysis nor required to be addressed by the ASME Code. Compliance with this specification is required to ensure that the accident analysis and design basis calculations remain valid.

The pressurizer safety valves satisfy 10 CFR 50.36(c)(2)(ii), Criterion 3.

LCO

One pressurizer safety valve is set to open at 2500 psia and one is set to open at 2565 psia. These setpoints are within the ASME specified tolerance to avoid exceeding the maximum RCS design pressure SL, to maintain accident analysis assumptions, and to comply with ASME Code requirements. The upper and lower pressure tolerance limits are based on the ± 1% tolerance requirements (Reference 1) for lifting pressures above 1000 psig. The limit protected by this specification is the RCPB SL of 110% of design pressure. Inoperability of one or both valves could result in exceeding the SL, if a transient were to occur. The consequences of exceeding the ASME pressure limit could include damage to one or more RCS components, increased leakage, or additional stress analysis being required prior to resumption of reactor operation.

APPLICABILITY

In MODEs 1 and 2, and portions of MODE 3 above the LTOP temperature, OPERABILITY of two valves is required because the combined capacity is required to keep reactor coolant pressure below 110% of its design value during certain accidents. MODE 3 is conservatively included, although the listed accidents may not require both safety valves for protection.

The LCO is not applicable in MODE 3 when all RCS cold leg temperatures are ≤ 365°F (Unit 1), ≤ 301°F (Unit 2), and MODEs 4 and 5, and MODE 6 with the reactor vessel head on, because LTOP is provided. Overpressure protection is not required in MODE 6 with the reactor vessel head off.

The Note allows entry into MODE 3 > 365°F (Unit 1), > 301°F (Unit 2) with the lift settings outside the LCO limits. This permits testing and examination of the safety valves at high P/T near their normal operating range, but only after the valves have had a preliminary cold setting. The cold setting gives assurance that the valves are OPERABLE near their design condition. Only one valve at a time will be removed from service for testing. The 36 hour exception is based on 18 hour outage time for each of the two valves. The 18 hour period is derived from operating experience that hot testing can be performed within this time frame.

ACTIONS

A.1

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

B.1 and B.2

If the Required Action cannot be met within the required Completion Time or if two 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 at or below 365°F (Unit 1), 301°F (Unit 2) with all RCS cold leg temperatures ≤ 365°F (Unit 1), ≤ 301°F (Unit 2) within 12 hours. The six hours allowed is reasonable, based on operating experience, to reach MODE 3 from full power without challenging plant systems. Similarly, the 12 hours allowed is reasonable, based on operating experience, to reduce temperature to below 365°F (Unit 1), 301°F (Unit 2) without challenging plant systems. At or below 365°F (Unit 1), 301°F (Unit 2), overpressure protection is provided by LTOP. The change from MODEs 1 or 2, or MODE 3 > 365°F (Unit 1), > 301°F (Unit 2) to MODE 3 ≤ 365°F (Unit 1), ≤ 301°F (Unit 2) reduces the RCS energy (core power and pressure), lowers the potential for large pressurizer insurges, and thereby removes the need for overpressure protection by two pressurizer safety valves.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.10.1

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

The pressurizer safety valves' setpoints are 2500 psia (+ 2%, - 1%) and 2565 psia (± 2%) for OPERABILITY; however, the valves are reset to ± 1% during the surveillance test to allow for drift.

REFERENCES

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

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.11 Pressurizer Power-Operated Relief Valves (PORVs)

BASES

BACKGROUND

The pressurizer is equipped with two types of devices for pressure relief: pressurizer safety valves and PORVs. The PORV is an electric, solenoid-operated valve that is automatically opened at a specific set pressure when the pressurizer pressure increases and is automatically closed on decreasing pressure. The PORV may also be manually-operated using controls installed in the Control Room.

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

The PORV and its block valve controls are powered from normal power supplies. Their controls are also capable of being powered from emergency supplies. Power supplies for the PORV are separate from those for the block valve. Power supply requirements are defined in Reference 1.

The PORV setpoint is equal to the high pressure reactor trip setpoint and below the opening setpoint for the pressurizer safety valves as required by Reference 2. The purpose of the relationship of these setpoints is to reduce the frequency of challenges to the safety valves, which, unlike the PORV, cannot be isolated if they were to fail open.

The primary purpose of this LCO is to ensure that the PORV and the block valve are operating correctly so the potential for a small break LOCA through the PORV pathway is minimized; or if a small break LOCA were to occur through a failed open PORV, the block valve could be manually operated to isolate the path.

The PORV may be manually-operated to depressurize the RCS as deemed necessary by the operator in response to normal or abnormal transients. The PORV may be used for depressurization when the pressurizer spray is not available, a condition that may be encountered during loss of offsite power. Operators can manually open the PORVs to reduce RCS pressure in the event of a steam generator tube rupture (SGTR) with offsite power unavailable.

The PORV may also be used for once through core cooling in the case of multiple equipment failure events that are not within the design basis, such as a total loss of feedwater.

The PORV functions as an automatic overpressure device and limits challenges to the safety valves. Although the PORV acts as an overpressure device for operational purposes, safety analyses do not take credit for PORV actuation, but do take credit for the safety valves.

The PORV also provides LTOP during heatup and cooldown. Limiting Condition for Operation 3.4.12, addresses this function.

APPLICABLE SAFETY ANALYSES

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

The possibility of a small break LOCA through the PORV is reduced when the PORV flow path is OPERABLE. The possibility is minimized if the flow path is isolated.

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

Pressurizer PORVs satisfy 10 CFR 50.36(c)(2)(ii), Criterion 3.

LCO

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

Valve OPERABILITY also means the PORV setpoint is correct. Ensuring the PORV opening setpoint is correct reduces the frequency of challenges to the safety valves, which, unlike the PORVs, cannot be isolated if they were to fail open.

APPLICABILITY

In MODEs 1 and 2, and MODE 3 with all RCS cold leg temperatures > 365°F (Unit 1), > 301°F (Unit 2), the PORV and its block valve are required to be OPERABLE to limit the potential for a small break LOCA through the flow path. A likely cause for PORV small break LOCA is a result of pressure increase transients that cause the PORV to open. Imbalances in the energy output of the core and heat removal by the secondary system can cause the RCS pressure to increase to the PORV opening setpoint. Pressure increase transients can occur any time the SGs are used for heat removal. The most rapid increases will occur at higher operating power and pressure conditions of MODEs 1 and 2.

Pressure increases are less prominent in MODE 3 because the core input energy is reduced, but the RCS pressure is high. Therefore, this LCO is applicable in MODEs 1 and 2, and MODE 3 with all RCS cold leg temperatures > 365°F (Unit 1), > 301°F (Unit 2). The LCO is not applicable in MODE 3 with all RCS cold leg temperatures ≤ 365°F (Unit 1), ≤ 301°F (Unit 2), when both pressure and core energy are decreased and the pressure surges become much less significant. The PORV setpoint is reduced for LTOP in MODE 3 with T_{avg} ≤ 365°F (Unit 1), ≤ 301°F (Unit 2) and in MODEs 4, 5, and 6 with the reactor vessel head in place. Limiting Condition for Operation 3.4.12 addresses the PORV requirements in these MODEs.

ACTIONS

The ACTIONS are modified by two Notes. Note 1 clarifies that the pressurizer PORVs are treated as separate entities, each with separate Completion Times (i.e., the Completion Time is on a component basis). Note 2 is an exception to LCO 3.0.4. The exception to LCO 3.0.4 permits entry into MODEs 1, 2, and 3 to perform cycling of the PORV or block valve to verify their OPERABLE status. Testing is typically not performed in lower MODEs.

BASES

A.1

With one or two PORVs inoperable and capable of being manually cycled, either the inoperable PORV(s) must be restored or the flow path isolated within one hour. The block valve should be closed but power must be maintained to the associated block valve, since removal of power would render the block valve inoperable. Although the PORV may be designated inoperable, it may be able to be manually opened and closed, and in this manner can be used to perform its function. Power-operated relief valve inoperability may be due to seat leakage, instrumentation problems, automatic control problems, or other causes that do not prevent manual use, and do not create a possibility for a small break LOCA. For these reasons, the block-valve may be closed but the Action requires power be maintained to the valve. This Condition is only intended to permit operation of the plant for a limited period of time not to exceed the next refueling outage (MODE 6) so that maintenance can be performed on the PORVs to eliminate the problem condition. The PORVs should normally be available for automatic mitigation of overpressure events and should be returned to OPERABLE status prior to entering startup (MODE 2).

Quick access to the PORV for pressure control can be made when power remains on the closed block valve. The Completion Time of one hour is based on plant operating experience that minor problems can be corrected or closure can be 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 either be isolated, by closing the associated block valve and removing the power from the block valve, or restored to OPERABLE status. The Completion Time of one hour is reasonable, based on challenges to the PORVs during this time period, and provides the operator adequate time to correct the situation. If the inoperable valve cannot be restored to OPERABLE status, it must be isolated within the specified time. Because there is at least one PORV that remains OPERABLE, five days are provided to restore the inoperable PORV to OPERABLE status.

C.1 and C.2

If one block valve is inoperable, then it must be restored to OPERABLE status, or the associated PORV placed in override closed. The prime importance for the capability to close the block valve is to isolate a stuck open PORV. Therefore, if the block valve cannot be restored to OPERABLE status within one hour, the Required Action is to place the PORV in override closed to preclude its automatic opening for an overpressure event, and to avoid the potential for a stuck open PORV at a time that the block valve is inoperable. The Completion Times of one hour are reasonable based on the small potential for challenges to the system during this time period and provide the operator time to correct the situation. Because at least one PORV remains OPERABLE, the operator is permitted a Completion Time of five days to restore the inoperable block valve to OPERABLE status. The time allowed to restore the block valve is based upon the Completion Time for restoring an inoperable PORV in Condition B since the PORVs are not capable of automatically mitigating an overpressure event when placed in override closed. If the block valve is restored within the Completion Time of five days, the power will be restored and the PORV restored to OPERABLE status.

D.1, D.2, and D.3

If both 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 one hour or isolate the flow path by closing and removing the power to the associated block valves. The Completion Time of one 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 Required Actions D.1 and D.2 have been completed, Required Action D.3 allows 72 hours to restore a PORV to OPERABLE status. This time is reasonable to perform required repairs. This time also accounts for the overpressure protection provided by the pressurizer safety valves in LCO 3.4.10.

E.1 and E.2

If both block valves are inoperable, it is necessary to either restore the block valves within the Completion Time

BASES

of one hour or place the associated PORVs in override closed and restore at least one block valve to OPERABLE status within 72 hours, and the remaining block valve in five days, per Required Action C.2. The Completion Time of one hour to either restore the block valves or place the associated PORVs in override closed is reasonable based on the small potential for challenges to the system during this time and provides the operator time to correct the situation.

F.1 and F.2

If the Required Actions and associated Completion Times are not met, then the plant must be brought to a MODE in which the LCO does not apply. The plant must be brought to at least MODE 3 within 6 hours and reduce any RCS cold leg temperature ≤ 365°F (Unit 1), ≤ 301°F (Unit 2) within 12 hours. The Completion Time of six hours is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging safety systems. Similarly, the Completion Time of 12 hours to reduce any RCS cold leg temperature ≤ 365°F (Unit 1), ≤ 301°F (Unit 2) is reasonable considering that a plant can cool down within that time frame. In MODE 3 with any RCS cold leg temperature ≤ 365°F (Unit 1), ≤ 301°F (Unit 2) and in MODEs 4, 5, and 6, maintaining PORV OPERABILITY is required per LCO 3.4.12.

SURVEILLANCE REQUIREMENTS

SR 3.4.11.1

A CHANNEL FUNCTIONAL TEST is performed on each PORV instrument channel every 92 days to ensure the entire channel will perform its intended function when needed.

SR 3.4.11.2

Block valve cycling verifies that it can be closed if necessary. The basis for the Frequency of 92 days is found in Reference 3. If the block valve is closed to isolate a PORV that is capable of being manually cycled, the OPERABILITY of the block valve is of importance because opening the block valve is necessary to permit the PORV to be used for manual control of RCS pressure. If the block valve is closed to isolate an otherwise inoperable PORV, the maximum Completion Time to restore the PORV and open the

BASES

block valve is 120 hours, which is well within the allowable limits (25%) to extend the block valve surveillance interval of 92 days. Furthermore, these test requirements would be completed by the reopening of a recently closed block valve upon restoration of the PORV to OPERABLE status (i.e., completion of the Required Action fulfills the SR).

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

SR 3.4.11.3

Surveillance Requirement 3.4.11.3 requires complete cycling of each PORV. Power-operated relief valve cycling demonstrates its function. The Frequency of 24 months is based on a typical refueling cycle and industry accepted practice.

SR 3.4.11.4

Performance of a CHANNEL CALIBRATION on each required PORV actuation channel is required every 24 months to adjust the whole channel so that it responds, and the valve opens within the required range and with accuracy to known input.

The 24 month Frequency considers operating experience with equipment reliability and matches the refueling outage Frequency.

REFERENCES

1. NUREG-0737, Paragraph II, G.I, "Clarification of TMI Action Plan Requirements," November 1980
2. Inspection and Enforcement Bulletin 79-05B, "Nuclear Incident at Three Mile Island - Supplement," April 21, 1979
3. ASME, Boiler and Pressure Vessel Code, Section XI

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.12 Low Temperature Overpressure Protection (LTOP) System

BASES

BACKGROUND

The LTOP System controls RCS pressure at low temperatures so the integrity of the RCPB is not compromised by violating the P/T limits of Reference 1, Appendix G. The reactor vessel is the limiting RCPB component for demonstrating such protection. Limiting Condition for Operation 3.4.3, provides the allowable combinations for operational P/T during cooldown, shutdown, and heatup to keep from violating the Reference 1, Appendix G requirements during the LTOP MODEs.

The reactor vessel material is less tough at low temperatures than at normal operating temperatures. As the vessel neutron exposure accumulates, the material toughness decreases and becomes less resistant to pressure stress at low temperatures (Reference 2). Reactor Coolant System 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. Limiting Condition for Operation 3.4.3 requires administrative control of RCS P/T during heatup and cooldown to prevent exceeding the P/T limits.

This LCO provides RCS overpressure protection by having a minimum coolant input capability and having adequate pressure relief capacity. Limiting coolant input capability requires all but one high pressure safety injection (HPSI) pump incapable of injection into the RCS and this HPSI pump will only be capable of manually injecting into the RCS. When suction of this HPSI pump is aligned to the Refueling Water Tank (RWT), the HPSI pump will be throttled unless an adequate vent path exists. The HPSI motor-operator valves must be in pull-to-override so that valves do not automatically actuate. In addition, administrative controls are placed on charging pump operation. The pressure relief

BASES

capacity requires either two OPERABLE redundant PORVs, one PORV and an RCS vent of 1.3 square inches, or the RCS depressurized and an RCS vent of 2.6 square inches. One PORV or the 1.3 square inch RCS vent, is the overpressure protection device that acts to terminate an increasing pressure event. The extra PORV or extra 1.3 square inch vent is for single failure criteria.

With minimum coolant input capability, the ability to provide core coolant addition is restricted. The safety injection actuation circuits are blocked to HPSI. If conditions require the use of more than one HPSI 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, one PORV with reduced lift setting and an RCS vent of 1.3 square inches, or an RCS vent of 2.6 square inches. Two relief valves are required for redundancy. One PORV has adequate relieving capability to prevent overpressurization for the required coolant input capability.

<u>PORV Requirements</u>

As designed for the LTOP System, each PORV is signaled to open if the RCS pressure approaches a limit determined by the LTOP actuation logic. The actuation logic monitors RCS temperature and pressure, and determines when the LTOP overpressure setting is approached. If the indicated pressure meets or exceeds the calculated value, a PORV is signaled to open.

The LCO presents the PORV setpoints for LTOP. Having the setpoints of both valves within the limits of the LCO ensures the P/T limits will not be exceeded in any analyzed event.

When a PORV is opened in an increasing pressure transient, the release of coolant causes the pressure increase to slow and reverse. As the PORV releases coolant, the system pressure decreases until a reset pressure is reached. At this point the event is terminated and the operator manually closes the PORV.

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.

If the vent path is ≥ 8 square inches (e.g., removing the pressurizer manway) the RCS can not be pressurized above the P/T limits, and the LTOP System is not required.

APPLICABLE SAFETY ANALYSES

Safety analyses (Reference 3) demonstrate that the reactor vessel is adequately protected against exceeding the Reference 1, Appendix G, P/T limits during shutdown. In MODEs 1 and 2, and MODE 3 with all RCS cold leg temperatures > 365°F (Unit 1), > 301°F (Unit 2), the RCPB is sufficiently above the nil-ductility temperature that the pressurizer safety valves prevent brittle fracture. At 365°F (Unit 1), 301°F (Unit 2) and below, overpressure prevention falls to the OPERABLE PORVs and administrative controls or to a depressurized RCS and a sufficient sized RCS vent. Each of these means has a limited overpressure relief capability.

Each time the P/T limit curves are revised, the LTOP System will be re-evaluated to ensure its functional requirements can still be satisfied using the PORV method or the depressurized and vented RCS condition.

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

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

<u>Mass Input Type Transients</u>

a. Inadvertent HPSI pump start;

b. Inadvertent HPSI and charging pump start; or

c. Charging/letdown flow mismatch.

<u>Heat Input Type Transients</u>

a. Inadvertent actuation of pressurizer heaters;

b. Loss of SDC; or

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

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

a. Rendering all but one HPSI pump incapable of injection and blocking automatic initiation from the remaining HPSI pump;

b. When HPSI suction is aligned to the RWT, the HPSI pump shall be in manual control and either:

   1) HPSI flow is limited to ≤ 210 gpm, or

   2) an RCS vent > 2.6 square inches is established;

c. Rendering HPSI motor operated valves (MOVs) only capable of manually aligning HPSI pump flow to the RCS;

d. Running only one charging pump when injecting via HPSI (charging pump requirements are controlled administratively); and

e. Maintaining a pressure bubble with level ≤ 170 inches.

The Reference 3 analyses demonstrate that either one PORV or the RCS vent and pressurizer steam volume can maintain RCS pressure below limits when only one HPSI pump is actuated and the HPSI pump's flow is throttled. If HPSI pump flow is not throttled during addition of mass to the RCS through on

HPSI loop MOV, then two PORVs or an RCS vent ≥ 2.6 square inches are capable of maintaining RCS pressure below limits. Thus, the LCO allows only one HPSI pump OPERABLE with flow throttled, or with an RCS vent ≥ 2.6 square inches during the LTOP MODEs.

Also to limit pressure overshoot over the PORV setpoint, the remaining HPSI and two charging pumps are rendered incapable of injection, and the RCPs are disabled during water solid operation.

Heatup and cooldown analyses established the temperature of LTOP Applicability at 365°F (Unit 1), and 301°F (Unit 2) and below, based on Standard Review Plan criteria. Above this temperature, the RCPB is sufficiently above the nil-ductility temperature and the pressurizer safety valves provide the reactor vessel pressure protection against brittle fracture. The vessel materials were assumed to have a fluence level equal to 2.61×10^{19} n/cm^2 (Unit 1), 4.0×10^{19} n/cm^2 (Unit 2).

The consequences of a LOCA in LTOP conform to Reference 1, Appendix K and 10 CFR 50.46, requirements, by having SITs operable in MODE 3 and one HPSI pump available for manual actuation.

PORV Performance

The fracture mechanics analyses show that the vessel is protected when the PORVs are set to open at or below the curves in Figure 3.4.12-1 and are applicable when the SDC System is not in operation. The setpoint is derived by modeling the performance of the LTOP System, assuming the limiting case of loss of SDC and one charging pump injecting into the RCS during water solid operation. These analyses consider pressure overshoot beyond the PORV opening setpoints, resulting from signal processing and valve stroke times. The PORV setpoints below the derived limit ensure the Reference 1, Appendix G limits will be met. When the SDC System is in operation, the PORV lift setting must be ≤ 429 psia (Unit 1), ≤ 443 psia (Unit 2). This ensures that the PORV lift setting is low enough to mitigate overpressure transients when SDC is in operation, since RCS temperature measurement is not accurate in this condition.

The PORV setpoints will be re-evaluated for compliance when the revised P/T limits conflict with the LTOP analysis limits. The P/T limits are periodically modified as the reactor vessel material toughness decreases due to embrittlement caused by neutron irradiation. Revised P/T limits are determined using neutron fluence projections and the results of examinations of the reactor vessel material irradiation surveillance specimens. However, the Calvert Cliffs' P/T limits are not projected to change through the end of Calvert Cliffs current operating license. The Bases for LCO 3.4.3 discuss these examinations.

The PORVs are considered active components. Thus, the failure of one PORV represents the worst case, single active failure.

RCS Vent Performance

With the RCS depressurized, analyses shows a vent size of 1.3 square inches is capable of mitigating the limiting allowed LTOP overpressure transient provided a pressurizer steam volume exists, two of the three HPSI pumps are disabled and the remaining HPSI pump's flow is throttled. In that event, this size vent maintains RCS pressure less than the maximum RCS pressure on the P/T limit curve. A 2.6 square inch vent is required to allow for single failures of other equipment, such as HPSI throttle valves. An 8 square inch vent is sufficient to preclude RCS overpressure events. Therefore, when an 8 square inch vent is established, LTOP System requirements are not necessary to maintain RCS pressure within limits.

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

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

LTOP System satisfies 10 CFR 50.36(c)(2)(ii), Criterion 2.

LCO

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

To limit the coolant input capability, the LCO requires a maximum of one HPSI pump only capable of manually injecting into the RCS. This is accomplished by disabling two HPSI pumps by either removing (racking out) their motor circuit breakers from the electrical power supply circuit or by locking shut their discharge valves. During required testing, other means of preventing two HPSI pumps from injecting into the RCS may be used. In addition, when not in use the remaining HPSI pump shall have its handswitch in pull-to-lock. When HPSI suction is aligned to the RWT for injection into the RCS, the HPSI pump must be in manual control, and either HPSI flow shall be limited to ≤ 210 gpm, or an RCS vent of ≥ 2.6 square inches is established. To provide single failure protection against a HPSI pump mass addition transient, the HPSI loop MOV handswitches must be placed in pull-to-override so the valves do not automatically actuate upon receipt of a safety injection signal. During required testing this requirement may be suspended.

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

a. Two OPERABLE PORVs and associated block valves open;

b. One OPERABLE PORV and associated block valve open and an RCS vent open with an area ≥ 1.3 square inches; or

c. The depressurized RCS and an RCS vent open with an area ≥ 2.6 square inches.

A PORV is OPERABLE for LTOP when its block valve is open, its lift setpoint is set in accordance with the LCO and testing has proven its ability to open at that setpoint, and motive power is available to the two valves and their control circuits.

The combination of these methods of overpressure prevention (as specified in LCO 3.4.12) are capable of mitigating the limiting LTOP transient.

APPLICABILITY

This LCO is applicable in MODE 3 when the temperature of any RCS cold leg is ≤ 365°F (Unit 1), ≤ 301°F (Unit 2), in MODEs 4, 5, and 6.

Limiting Condition for Operation 3.4.3 provides the operational P/T limits for all MODEs. Limiting Condition for Operation 3.4.10, requires the OPERABILITY of the pressurizer safety valves that provide overpressure protection during MODEs 1 and 2, and MODE 3 above 365°F (Unit 1), 301°F (Unit 2).

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

The Applicability is modified by a Note stating that this Specification is not applicable when the RCS is vented ≥ 8 square inches. An RCS vent of this size precludes RCS overpressure events.

ACTIONS

A Note to the ACTIONS restricts entry into MODEs or other specified conditions in the Applicability of this LCO while complying with the ACTIONS (i.e., while the LCO is not met). Limiting Condition for Operation 3.0.4 typically allows entry into MODEs or other specified conditions in the Applicability as part of any unit shutdown, however, the restriction of this Note is necessary to assure LTOP is available prior to operating within the Applicability of this LCO.

<u>A.1</u>

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

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

B.1

With HPSI flow > 210 gpm and suction aligned to the RWT and an RCS vent < 2.6 square inches established, sufficient overpressure protection may not exist and overpressurization may be possible.

The immediate Completion Time to initiate actions to reduce HPSI flow to ≤ 210 gpm reflects the importance of maintaining overpressure protection of the RCS.

C.1

With one or more HPSI loop MOVs capable of automatically aligning HPSI pump flow to the RCS, single failure protection against a HPSI pump mass addition transient is lost. Therefore, action is required to be immediately initiated to restore single failure protection by placing the affected HPSI loop MOV handswitch to pull-to-override, or shutting and disabling the affected HPSI loop MOV, or isolating the affected HPSI header flow path.

The immediate Completion Time to initiate action to restore single failure protection for the HPSI pump mass addition transient reflects the importance of restoring single failure protection for low temperature overpressurization mitigation.

D.1

In MODE 3 when any RCS cold leg temperature is ≤ 365°F (Unit 1), ≤ 301°F (Unit 2) or in MODE 4, with one of the two required PORVs inoperable and an RCS vent < 1.3 square inches established, the inoperable PORV must be restored to OPERABLE status within a Completion Time of five days. The inoperable PORV is required to meet the LCO requirement and to provide low temperature overpressure mitigation while withstanding a single failure of an active component.

The Completion Time is based on the fact that only one PORV is required to mitigate an overpressure transient.

E.1

The consequences of operational events that will overpressure the RCS are more severe at lower temperature (Reference 4). Thus, with one of the two required PORVs inoperable and an RCS vent < 1.3 square inches established in MODE 5 or in MODE 6, the Completion Time to restore two valves to OPERABLE status is 24 hours.

The 24 hour Completion Time to restore the inoperable PORV to OPERABLE status in MODE 5 or in MODE 6 is a reasonable amount of time to investigate and repair several types of PORV failures without exposure to a lengthy period with only one PORV OPERABLE to protect against overpressure events.

F.1

If the required Actions and associated Completion Times of Conditions D or E cannot be met, the RCS is required to be depressurized and vented through a vent ≥ 1.3 square inches. This action must be completed within 48 hours. This action along with the OPERABLE PORV restores single failure protection and ensures the flow capacity is greater than that required for the worst case mass input transient reasonable during the applicable MODEs. This action protects the RCPB from an overpressure event and a possible brittle failure of the reactor vessel.

The Completion Time of 48 hours to depressurize and vent the RCS is based on the time required to place the plant in this condition and in a controlled manner. The probability of an overpressure event occurring along with a single failure of the remaining OPERABLE PORV is unlikely.

G.1

If all required PORVs (i.e., when one PORV is required and it is inoperable or when two PORVs are required and both are inoperable) are inoperable, the RCS must be depressurized and a vent established within 48 hours. The vent must be sized at least 2.6 square inches to ensure the flow capacity is greater than that required for the worst case mass input transient reasonable during the applicable MODEs. This action protects the RCPB from a low temperature overpressure event and a possible brittle failure of the reactor vessel.

The Completion Time of 48 hours to depressurize and vent the RCS is based on the time required to place the plant in this condition and the relatively low probability of an overpressure event during this time period due to increased operator awareness of administrative control requirements.

SURVEILLANCE REQUIREMENTS

SR 3.4.12.1 and SR 3.4.12.2

To minimize the potential for a low temperature overpressure event by limiting the mass input capability, verification that a maximum of one HPSI pump is only capable of manually injecting into the RCS, and automatic alignment of the HPSI loop MOVs, is prevented (by disabling the automatic opening features of the HPSI loop MOVs) is required. The HPSI pumps are rendered incapable of injecting into the RCS through removing the power from the pumps by racking the breakers out under administrative control or by verifying their discharge valves are locked shut.

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

SR 3.4.12.3

Surveillance Requirement 3.4.12.3 requires verifying that the required RCS vent is open, once every 12 hours for a valve that is unlocked open, and once every 31 days for a valve that is locked open.

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

SR 3.4.12.4

The PORV block valve must be verified open every 72 hours to provide the flow path for each required PORV to perform its function when actuated. The valve can be remotely verified open in the main Control Room.

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

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

SR 3.4.12.5

Performance of a CHANNEL FUNCTIONAL TEST is required every 31 days to verify and, as necessary, adjust the PORV open setpoints. The CHANNEL FUNCTIONAL TEST will verify on a monthly basis that the PORV lift setpoints are within the LCO limit. Power-operated relief valve actuation could depressurize the RCS and is not required. The 31 day Frequency considers experience with equipment reliability.

A Note has been added indicating this SR is required to be performed 12 hours after decreasing RCS cold leg temperature to ≤ 365°F (Unit 1), ≤ 301°F (Unit 2). The test cannot be performed until the RCS is in the LTOP MODEs when the PORV lift setpoint can be reduced to the LTOP setting. The test must be performed within 12 hours after entering the LTOP MODEs.

SR 3.4.12.6

Performance of a CHANNEL CALIBRATION on each required PORV actuation channel is required every 24 months to adjust the whole channel so that it responds and the valve opens within the required LTOP range and with accuracy to known input.

BASES

The 24 month Frequency considers operating experience with equipment reliability and matches the typical refueling outage schedule.

REFERENCES

1. 10 CFR Part 50, Domestic Licensing of Production and Utilization Facilities
2. Generic Letter 88-11, NRC Position on Radiation Embrittlement of Reactor Vessel Materials and its Impact on Plant Operations, July 12, 1988
3. UFSAR, Section 4.2.2, Low Temperature Overpressure Protection
4. Generic Letter 90-06, Resolution of Generic Issues 70, "PORV and Block Valve Reliability," and 94, "Additional LTOP Protection for PWRs," June 25, 1990

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.13 RCS Operational LEAKAGE

BASES

BACKGROUND

Components that contain or transport the coolant to or from the reactor core makeup the RCS. Component joints are made by welding, bolting, rolling, or pressure loading. 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.

Reference 1, Appendix 1C, Criterion 16 requires means for detecting reactor coolant LEAKAGE. Reference 2 describes acceptable methods for selecting leakage detection systems.

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

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

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

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 a LOCA; the amount of leakage can affect the probability of such an event. The safety analysis for an event resulting in steam discharge to the atmosphere assumes a 1 gpm primary to secondary LEAKAGE as the initial condition.

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

Reference 1, Section 14.15 analysis for SGTR assumes the contaminated secondary fluid is released via the atmospheric dump valves and main steam safety valves. Most of the released radiation is due to the ruptured tube. The 1 gpm primary to secondary LEAKAGE is relatively inconsequential.

The steam line break is more limiting for site radiation releases. The safety analysis for the steam line break accident assumes 1 gpm primary to secondary LEAKAGE in one generator as an initial condition. The dose consequences resulting from the steam line break accident are described in Reference 1, Section 14.14.

Reactor Coolant System operational LEAKAGE satisfies 10 CFR 50.36(c)(2)(ii), Criterion 2.

LCO

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

c. Identified LEAKAGE

Up to 10 gpm of identified LEAKAGE is considered allowable because LEAKAGE is from known sources that do not interfere with the detection of unidentified LEAKAGE and is well within the capability of the RCS makeup system. Identified LEAKAGE includes LEAKAGE to the Containment Structure from specifically known and located sources, but does not include pressure boundary LEAKAGE or controlled 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 Steam Generator

The 100 gallon per day limit on primary to secondary LEAKAGE through any one SG is consistent with SG tube sleeving commitments.

APPLICABILITY

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

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

ACTIONS

A.1

Unidentified LEAKAGE, identified LEAKAGE, or primary to secondary LEAKAGE in excess of the LCO limits must be reduced to within limits within four hours. This Completion Time allows time to verify leakage rates and either identify unidentified LEAKAGE or reduce LEAKAGE to within limits

BASES

before the reactor must be shut down. This action is necessary to prevent further deterioration of the RCPB.

B.1 and B.2

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

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

SURVEILLANCE REQUIREMENTS

SR 3.4.13.1

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

The RCS water inventory balance must be performed with the reactor at steady-state operating conditions and near operating pressure.

Steady-state operation is required to perform a proper water inventory balance; 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 leakoff flows.

An early warning of pressure boundary LEAKAGE or unidentified LEAKAGE is provided by the automatic systems that monitor the containment atmosphere radioactivity and the containment sump level. These leakage detection systems are specified in LCO 3.4.14.

The 72 hour Frequency is a reasonable interval to trend LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents.

SR 3.4.13.2

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

In the event one or more SGs are determined to not meet the requirements of the Steam Generator Tube Surveillance Program at anytime in MODEs 1 through 4, action to comply with LCO 3.0.3 must be taken.

REFERENCES

1. UFSAR
2. Regulatory Guide 1.45, Reactor Coolant Pressure Boundary Leakage Detection Systems, May 1973

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.14 RCS Leakage Detection Instrumentation

BASES

BACKGROUND

Reference 1, Appendix 1C, Criterion 16 requires means for detecting RCS LEAKAGE. Reference 2 describes acceptable methods for selecting leakage detection systems.

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

Industry practice has shown that water flow changes of 0.5 gpm to 1.0 gpm can readily be detected in contained volumes by monitoring changes in water level, in flow rate, or in the operating frequency of a pump. The containment sump used to collect unidentified LEAKAGE is instrumented to alarm when level increases above the alarm trip setpoint. The sump is then drained and time logged. If the alarm sounds again, the time is logged and a leakage rate is calculated. This is acceptable for detecting increases in unidentified LEAKAGE.

The reactor coolant contains radioactivity that, when released to the Containment Structure, can be detected by radiation monitoring instrumentation. Reactor coolant radioactivity levels will be low during initial reactor startup and for a few weeks thereafter, until activated corrosion products have been formed, and fission products appear from fuel element cladding contamination or cladding defects. Instrument sensitivities of 5E-12 μCi/cc DOSE EQUIVALENT I-131 for particulate monitoring and of 3E-6 μCi/cc Xe-133 for gaseous monitoring are practical for these leakage detection systems. Radioactivity detection systems are included for monitoring both particulate and gaseous activities, because of their sensitivities and responses to RCS LEAKAGE. These radioactivity monitors have a range of 10^{1}-10^{6} counts per minute.

An increase in humidity of the containment atmosphere would indicate release of water vapor to the Containment

BASES

Structure, which would be 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 sump. Humidity level monitoring is considered most useful as an indication to alert the operator to a potential problem. Humidity monitors are not required by this LCO.

Air temperature and pressure monitoring methods may also be used to infer unidentified LEAKAGE to the Containment Structure. 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 Structure. The relevance of temperature and pressure measurements are affected by containment free volume and, for temperature, detector location. Alarm signals from these instruments can be valuable in recognizing rapid and sizable leakage to the Containment Structure. Temperature and pressure monitors are not required by this LCO.

APPLICABLE SAFETY ANALYSES

The need to evaluate the severity of an alarm or an indication is important to the operators, and the ability to compare and verify with indications from other systems is necessary. The RCS leakage detection instrumentation is described in Reference 1, Section 4.3. Multiple instrument locations are utilized, if needed, to help identify the location of the LEAKAGE and its source.

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

Reactor Coolant System leakage detection instrumentation satisfies 10 CFR 50.36(c)(2)(ii), Criterion 1.

BASES

LCO

One method of protecting against large RCS LEAKAGE derives from the ability of instruments to rapidly detect extremely small leaks. This LCO requires instruments of diverse monitoring principles to be OPERABLE to provide a high degree of confidence that extremely small leaks are detected in time to allow actions to place the plant in a safe condition when RCS LEAKAGE indicates possible RCPB degradation.

The LCO is satisfied when monitors of diverse measurement means are available. Thus, the containment sump monitor, in combination with a particulate or gaseous radioactivity monitor, provides an acceptable minimum.

APPLICABILITY

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

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

ACTIONS

The actions are modified by a Note that indicates the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when the containment sump and required radiation monitor channels are inoperable. This allowance is provided because other means are available to monitor for RCS LEAKAGE.

A.1 and A.2

If the containment sump level alarm is inoperable, no other form of sampling can provide the equivalent information.

However, the containment atmosphere radioactivity monitor will provide indications of changes in leakage. Together with the atmosphere monitor, the periodic surveillance for RCS water inventory balance, SR 3.4.13.1, must be performed at an increased frequency of 24 hours to provide information that is adequate to detect leakage.

BASES

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

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

With both gaseous and particulate containment atmosphere radioactivity monitoring instrumentation channels inoperable, alternative action is required. Either grab samples of the containment atmosphere must be taken and analyzed, or water inventory balances, in accordance with SR 3.4.13.1, must be performed to provide alternate periodic information. With a sample obtained and analyzed, or an inventory balance performed every 24 hours, the reactor may be operated for up to 30 days to allow restoration of at least one of the radioactivity monitors.

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

C.1 and C.2

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

D.1

If all required alarms and monitors are inoperable, no automatic means of monitoring leakage are available, an immediate plant shutdown in accordance with LCO 3.0.3 is required.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1

Surveillance Requirement 3.4.14.1 requires the performance of a CHANNEL CHECK of the required containment atmosphere radioactivity monitors. The check gives reasonable confidence the channel is operating properly. The Frequency of 12 hours is based on instrument reliability and is reasonable for detecting off normal conditions.

SR 3.4.14.2

Surveillance Requirement 3.4.14.2 requires the performance of a CHANNEL FUNCTIONAL TEST of the required containment atmosphere radioactivity monitors. The test ensures that the monitor can perform its function in the desired manner. The test verifies the alarm setpoint and relative accuracy of the instrument string. The Frequency of 31 days considers instrument reliability, and operating experience has shown it proper for detecting degradation.

SR 3.4.14.3 and SR 3.4.14.4

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 Structure. The Frequency of 24 months is a typical refueling cycle and considers channel reliability. Operating experience has shown this Frequency is acceptable.

REFERENCES

1. UFSAR
2. Regulatory Guide 1.45, Reactor Coolant Pressure Boundary Leakage Detection Systems, May 1973

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.15 RCS Specific Activity

BASES

BACKGROUND

Title 10 CFR Part 100 specifies the maximum dose, to the whole body and the thyroid, an individual at the site boundary can receive for two hours during an accident. The limits on specific activity ensure that the doses are held to within the acceptance criteria given in Reference 1, Chapter 14, during analyzed transients and accidents.

The RCS specific activity LCO limits the allowable concentration level of radionuclides in the reactor coolant. The LCO limits are established to minimize the offsite radioactivity dose consequences in the event of a SGTR accident.

The LCO contains specific activity limits for both DOSE EQUIVALENT I-131 and gross activity. The allowable levels are intended to limit the dose at the site boundary to within the acceptance criteria given in Reference 1, Chapter 14.

APPLICABLE SAFETY ANALYSIS

The LCO limits on the specific activity of the reactor coolant ensure that the resulting doses at the site boundary will not exceed the acceptance criteria given in Reference 1, Chapter 14. The SGTR safety analysis (Reference 1, Section 4.15) assumes the specific activity of the reactor coolant at the LCO limits and an existing reactor coolant SG tube leakage rate of 1 gpm.

The rise in pressure in the ruptured SG causes radioactively contaminated steam to discharge to the atmosphere through the atmospheric dump valves and the main steam safety valves.

The safety analysis shows the radiological consequences of an SGTR accident, are within the Reference 1, Chapter 14 acceptance criteria. Operation with iodine specific activity levels greater than the LCO limit is permissible, if the activity levels do not exceed the limits shown in Figure 3.4.15-1 for more than 100 hours.

BASES

The remainder of the above limit permissible iodine levels shown in Figure 3.4.15-1, are acceptable because of the low probability of an SGTR accident occurring during the established 100 hour time limit. The occurrence of an SGTR accident at these permissible levels could increase the site boundary dose levels beyond the acceptance criteria given in Reference 1, Chapter 14.

Reactor Coolant System specific activity satisfies 10 CFR 50.36(c)(2)(ii), Criterion 2.

LCO

The specific activity is limited to 1.0 μCi/gm DOSE EQUIVALENT I-131, and the gross activity in the primary coolant is limited to the number of μCi/gm equal to 100 divided by $\bar{E}$ (average disintegration energy of the sum of the average beta and gamma energies of the coolant nuclides). The limit on DOSE EQUIVALENT I-131 ensures the thyroid dose to an individual at the site boundary during the DBA will be within the acceptance criteria given in Reference 1, Chapter 14. The limit on gross activity ensures the whole body dose to an individual at the site boundary during the DBA will be within the acceptance criteria given in Reference 1, Chapter 14.

The SGTR accident analysis (Reference 1, Section 4.15) shows that the site boundary dose levels are within acceptable limits. Violation of the LCO may result in reactor coolant radioactivity levels that could, in the event of an SGTR, lead to site boundary doses that exceed the Reference 1, Chapter 14 acceptance criteria.

APPLICABILITY

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

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

ACTIONS

A.1 and A.2

With the DOSE EQUIVALENT I-131 greater than the LCO limit, samples at intervals of four hours must be taken to demonstrate the limits of Figure 3.4.15-1 are not exceeded. The Completion Time of four hours is required to obtain and analyze a sample.

Sampling must continue for trending. The DOSE EQUIVALENT I-131 must be restored to within limits, within 100 hours. The Completion Time of 100 hours is required if the limit violation resulted from normal iodine spiking.

A Note to the Required Action of Condition A, excludes the MODE change restriction of LCO 3.0.4. This exception allows entry into the applicable MODE(s) while relying on the ACTIONS even though the ACTIONS may eventually require plant shutdown. This exception is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is limiting due to exceeding this limit, and the ability to restore transient DOSE EQUIVALENT I-131 specific activity excursions while the plant remains at, or proceeds to power operation.

B.1

If a Required Action and associated Completion Time of Condition A is not met or if the DOSE EQUIVALENT I-131 is in the unacceptable region of Figure 3.4.15-1, the reactor must be brought to MODE 3 with RCS average temperature < 500°F within six hours. The allowed Completion Time of six hours is required to reach MODE 3 below 500°F without challenging plant systems.

C.1

With the gross activity in excess of the allowed limit, the unit must be placed in a MODE in which the requirement does not apply.

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

SURVEILLANCE REQUIREMENTS

SR 3.4.15.1

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

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

SR 3.4.15.2

This SR is performed to ensure iodine remains within limits during normal operation and following fast power changes when fuel failure is more apt to occur. The 14 day Frequency is adequate to trend changes in the iodine activity level considering gross activity is monitored every 7 days. The Frequency, between two hours and six hours after a power change of ≥ 15% RTP within a one hour period, is established because the iodine levels peak during this time following fuel failure; samples at other times would provide inaccurate results.

The SR is modified by a Note which requires the surveillance test to only be performed in MODE 1. This is required because the level of fission products generated in other MODEs is much less. Also, fuel failures associated with fast power changes is more apt to occur in MODE 1 than in MODEs 2 and 3.

BASES

SR 3.4.15.3

A radiochemical analysis for $\bar{E}$ determination is required every 184 days (six months) with the plant operating in MODE 1 equilibrium conditions. The $\bar{E}$ determination directly relates to the LCO and is required to verify plant operation within the specified gross activity LCO limit. The analysis for $\bar{E}$ is a measurement of the average energies per disintegration for isotopes with half lives longer than 15 minutes, excluding iodines. The Frequency of 184 days recognizes $\bar{E}$ does not change rapidly.

This SR has been modified by a Note that indicates sampling is not required to be performed until 31 days after 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for $\geq$ 48 hours. This ensures the radioactive materials are at equilibrium so that analysis for $\bar{E}$ is representative and not skewed by a crud burst or other similar abnormal event.

REFERENCES

1. UFSAR

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.16 Special Test Exception (STE) RCS Loops - MODE 2

BASES

BACKGROUND	This STE to LCO 3.4.4 and LCO 3.3.1, permits reactor criticality under no flow conditions during PHYSICS TESTS (natural circulation demonstration, station blackout, and loss of offsite power) while at low THERMAL POWER levels. Reference 1, requires that a test program be established to ensure that structures, systems, and components will perform satisfactorily in service. All functions necessary to ensure that the specified design conditions are not exceeded during normal operation and anticipated operational occurrences must be tested. This testing is an integral part of the design, construction, and operation of the power plant. The key objectives of a test program are to: provide assurance that the facility has been adequately designed to validate the analytical models used in the design and analysis, to verify the assumptions used to predict plant response, to provide assurance that installation of equipment at the facility has been accomplished in accordance with the design, and to verify that the operating and emergency procedures are adequate. Testing is performed prior to initial criticality, during startup, and following low power operations. The tests will include: verifying the ability to establish and maintain natural circulation following a plant trip between 10% and 15% RTP, performing natural circulation cooldown on emergency power, and (during the cooldown), showing that adequate boron mixing occurs and that pressure can be controlled using auxiliary spray and pressurizer heaters powered from the emergency power sources.
APPLICABLE SAFETY ANALYSES	As described in LCO 3.0.7, compliance with Special Test Exception LCOs is optional, and therefore no criteria of 10 CFR 50.36(c)(2)(ii) apply. Special 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

LCO

This LCO is provided to allow for the performance of PHYSICS TESTS in MODE 2 (after a refueling), where the core cooling requirements are significantly different than after the core has been operating. Without this LCO, plant operations would be held bound to the normal operating LCOs for reactor coolant loops and circulation (MODEs 1 and 2), and the appropriate tests could not be performed.

In MODE 2, where core power level is considerably lower and the associated PHYSICS TESTS must be performed, operation is allowed under no flow conditions provided THERMAL POWER is < 5% RTP and the reactor trip setpoints of the OPERABLE power level channels are set ≤ 15% RTP. These limits ensure no SLs or fuel design limits will be violated.

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

APPLICABILITY

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

ACTIONS

A.1

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

SURVEILLANCE REQUIREMENTS

SR 3.4.16.1

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

BASES

degradation and verify operation is within the LCO limits. Plant operations are conducted slowly during the performance of PHYSICS TESTS, and monitoring the power level once per hour is sufficient to ensure that the power level does not exceed the limit.

SR 3.4.16.2

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

REFERENCES

1. 10 CFR Part 50, Appendix B, Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants, Section XI

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.17 Special Test Exception (STE) RCS Loops - MODES 4 and 5

BASES

BACKGROUND

This STE to LCO 3.4.6, LCO 3.4.7, and LCO 3.4.8, allows no RCS or SDC loops to be in operation during the time intervals required: 1) for local leak rate testing of Containment Penetration Number 41 (SDC); and 2) for maintenance on the common SDC suction line or on the SDC flow control valve (CV-306).

APPLICABLE SAFETY ANALYSIS

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

LCO

This LCO is provided to allow for the performance of testing and maintenance in MODEs 4 and 5 (normally after a refueling), where the core cooling requirements are significantly different than after the core has been operating. Without this LCO, plant operations would be held bound to the normal operation LCOs for reactor coolant loops and circulation (MODEs 4 and 5), and the appropriate tests or maintenance could not be performed in these MODEs.

In MODEs 4 and 5, operation is allowed under no flow conditions provided: the xenon reactivity is $\leq 0.1\%\ \Delta k/k$ and approaching stability, no operations are permitted which could cause reduction of boron concentration, the charging pumps are de-energized, the charging flow paths are isolated, and the SHUTDOWN MARGIN requirement of LCO 3.1.1 is verified at least once per eight hours. These limits along with the SRs ensure no SLs or fuel design limits will be violated.

The exception is allowed even though there are no bounding safety analyses. These tests or maintenance are allowed since they are performed under close supervision during the test program and must stay within the requirements of the LCO.

BASES

APPLICABILITY	The LCO ensures that while within this LCO the plant will not be operated in any other MODE besides MODEs 4 and 5 without forced circulation. This is because the MODEs of Applicability for this Specification are MODEs 4 and 5. This Specification allows testing and maintenance to be performed on the SDC System while SDC is required to be OPERABLE.
ACTIONS	<u>A.1</u> If one or more requirements of the LCO are not met, all activities being performed under this STE must be immediately suspended. These activities are local leak rate testing of the SDC penetration and maintenance on valves in the SDC System. The Completion Time to suspend these activities immediately ensures the plant is not placed in an unanalyzed condition and prevents exceeding the specified acceptable fuel design limits.
SURVEILLANCE REQUIREMENTS	<u>SR 3.4.17.1</u> Xenon reactivity must be verified to be within limits once within one hour prior to suspending the reactor coolant circulation requirements of LCO 3.4.6, LCO 3.4.7, and LCO 3.4.8. The frequency of once within one hour prior to suspending the applicable RCS Loops LCO will ensure that the xenon reactivity is within limits and trending toward stability prior to suspending forced flow cooling. This will ensure no SLs or fuel design limits will be violated while testing or maintenance are being conducted. <u>SR 3.4.17.2 and SR 3.4.17.3</u> Verifying the charging pumps are de-energized and the charging flow paths are isolated, ensures that the major source of a boron reduction is not available. These two SRs support the requirement that no source be available that could cause an RCS boron concentration reduction. These SRs are required to be verified at a frequency of one hour. The one hour frequency is sufficient to ensure that these sources will not be available to cause a reduction of the RCS boron concentration.

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Subsequent performance of these SRs after the initial verification that the charging pumps are de-energized and the charging flow paths are isolated, may be performed administratively.

SR 3.4.17.4

This SR requires that a SHUTDOWN MARGIN verification be performed in accordance with SR 3.1.1.1 once per eight hours. The normal Frequency for these SRs is once per 24 hours. The eight hour Frequency reflects that no forced flow cooling is available and that the SHUTDOWN MARGIN should be verified more frequently. The eight hour Frequency is sufficient to ensure that the SHUTDOWN MARGIN remains within limits while under this STE.

REFERENCES None

B 3.5 EMERGENCY CORE COOLING SYSTEM (ECCS)

B 3.5.1 Safety Injection Tanks (SITs)

BASES

BACKGROUND

The function of the four SITs is to inject large quantities of borated water to the reactor vessel following the blowdown phase of a large break loss of coolant accident (LOCA) and to provide inventory to help accomplish the refill phase that follows thereafter.

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

The refill phase of a LOCA follows immediately where reactor coolant inventory has vacated the core through steam flashing and ejection out through the break. The core is essentially in adiabatic heatup. The rest of the SITs' inventory is then available to help fill voids in the lower plenum and reactor vessel downcomer to establish a recovery water level at the bottom of the core and continue reflood of the core with the addition of safety injection water.

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

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

BASES

Additionally, the isolation valves are interlocked with the pressurizer pressure instrumentation channels, to ensure that the valves will automatically open as RCS pressure increases above SIT pressure, and to prevent inadvertent closure prior to an accident. The valves also receive a safety injection actuation signal (SIAS) to open. These features ensure that the valves meet the requirements of Reference 1 for "operating bypasses" and that the SITs will be available for injection without reliance on operator action.

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

APPLICABLE SAFETY ANALYSES

The large break LOCA analyses at full power (Reference 2, Section 6.3) credits the SITs. This is the Design Basis Accident (DBA) that establishes the acceptance limits for the SITs. Reference to the analysis for this DBA is used to assess changes to the SITs as they relate to the acceptance limits.

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

The limiting large break LOCA is a double ended guillotine cold leg break at the discharge of the reactor coolant pump. During this event, the SITs discharge to the RCS as soon as RCS pressure decreases to below SIT pressure. As a conservative estimate, no credit is taken for safety injection pump flow until the SITs are empty. This results in a minimum effective delay of over 60 seconds, during which the SITs must provide the core cooling function. The actual delay time is less. No operator action is assumed during the blowdown stage of a large break LOCA.

This Limiting Condition for Operation (LCO) helps to ensure that the following acceptance criteria, established by Reference 3 for the Emergency Core Cooling System (ECCS), will be met following a LOCA:

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

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

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

d. The core is maintained in a coolable geometry.

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

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

These precautions ensure that the SITs are available during an accident (Reference 2, Section 14.17). With power supplied to the valves, a single active failure could result in a valve closure, which would render one SIT unavailable for injection. If a second SIT is lost through the break, only two SITs would reach the core. Since the only active failure that could affect the SITs would be the closure of a motor-operated outlet valve, the requirement to remove power from these eliminates this failure mode.

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

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

A minimum level, corresponding to 1090 cubic feet of borated water is used in the safety analysis as the volume in the SITs. To provide margin, the low level alarms are set at 187 inches (corresponding to 1113 cubic feet) and 199 inches (corresponding to 1179 cubic feet). The analyses are based upon the cubic feet requirements; the level (inches) figures are provided for operator use because the level indicator provided in the Control Room is marked in inches, not in cubic feet.

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

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

A minimum pressure of 195 psia is used in the analyses. To allow for instrument accuracy, a 200 psig minimum and 250 psig maximum are specified. The maximum allowable boron concentration of 2700 ppm is based upon boron precipitation limits in the core following a LOCA. Establishing a maximum limit for boron is necessary since the time at which boron precipitation would occur in the core following a LOCA is a function of break location, break size, the amount of boron injected into the core, and the point of ECCS injection.

Post-LOCA emergency procedures directing the operator to establish simultaneous hot and cold leg injection are based on the worst case minimum boron precipitation time. Maintaining the maximum SIT boron concentration within the upper limit ensures that the SITs do not invalidate this calculation. An excessive boron concentration in any of the borated water sources used for injection during a LOCA could result in boron precipitation earlier than predicted.

The minimum boron requirements of 2300 ppm are based on beginning-of-life reactivity values and are selected to ensure that the reactor will remain subcritical during the reflood stage of a large break LOCA. During a large break LOCA, all control element assemblies are assumed not to insert into the core, and the initial reactor shutdown is accomplished by void formation during blowdown. Sufficient boron concentration must be maintained in the SITs to prevent a return to criticality during reflood.

The SITs satisfy 10 CFR 50.36(c)(2)(ii), Criterion 3.

LCO

The LCO establishes the minimum conditions required to ensure that the SITs are available to accomplish their core cooling safety function following a LOCA. Four SITs are required to be OPERABLE to ensure that 100% of the contents, for three of the SITs, will reach the core during a LOCA.

This is consistent with the assumption that the contents of one tank spill through the break. If the contents of fewer than three tanks are injected during the blowdown phase of a LOCA, the ECCS acceptance criteria of Reference 3 could be violated.

For an SIT to be considered OPERABLE, the isolation valve must be fully open, power removed above 2000 psig, and the limits established in the Surveillance Requirement (SR) for contained volume, boron concentration, and nitrogen cover pressure must be met.

APPLICABILITY

In MODEs 1, 2, and 3 the SIT OPERABILITY requirements are based on an assumption of full power operation. Although cooling requirements decrease as power decreases, the SITs

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are still required to provide core cooling as long as elevated RCS pressures and temperatures exist.

In MODEs 4, 5, and 6, the SIT motor-operated isolation valves are closed to isolate the SITs from the RCS. This allows RCS cooldown and depressurization without discharging the SITs into the RCS or requiring depressurization of the SITs.

ACTIONS

A.1

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

B.1

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

C.1 and C.2

If the SIT cannot be restored to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and 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.

D.1

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

SURVEILLANCE REQUIREMENTS

SR 3.5.1.1

Verification every 12 hours that each SIT isolation valve is fully open, as indicated in the Control Room, ensures that SITs are available for injection and ensures timely discovery if a valve should be partially closed. If an isolation valve is not fully open, the rate of injection to the RCS would be reduced. Although a motor-operated valve should not change position with power removed, a closed valve could result in not meeting accident analysis assumptions. A 12 hour Frequency is considered reasonable in view of other administrative controls that ensure the unlikelihood of a mispositioned isolation valve.

SR 3.5.1.2 and SR 3.5.1.3

Safety injection tank borated water volume and nitrogen cover pressure should be verified to be within specified limits every 12 hours in order to ensure adequate injection during a LOCA. Due to the static design of the SITs, a 12 hour Frequency usually allows the operator sufficient time to identify changes before the limits are reached. Operating experience has shown this Frequency to be appropriate for early detection and correction of off normal trends.

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SR 3.5.1.4

Thirty-one days is reasonable for verification to determine that each SIT's boron concentration is within the required limits, because the static design of the SITs limits the ways in which the concentration can be changed. The 31 day Frequency is adequate to identify changes that could occur from mechanisms, such as stratification or inleakage.

Sampling the affected SIT (by taking the sample at the discharge of the operating HPSI pump) within one hour prior to a 1% volume increase of normal tank volume, will ensure the boron concentration of the fluid to be added to the SIT is within the required limit prior to adding inventory to the SIT(s).

SR 3.5.1.5

Verification every 31 days that power is removed from each SIT isolation valve operator, by maintaining the feeder breaker open under administrative control, when the pressurizer pressure is ≥ 2000 psig ensures that an active failure could not result in the undetected closure of an SIT motor-operated isolation valve. If this were to occur, only two SITs would be available for injection, given a single failure coincident with a LOCA. Since installation and removal of power to the SIT isolation valve operators is conducted under administrative control, the 31 day Frequency was chosen to provide additional assurance that power is removed.

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

BASES

REFERENCES

1. Institute of Electrical and Electronic Engineers Standard 279-1971, "IEEE Standard: Criteria for Protection Systems for Nuclear Power Generating Stations"
2. Updated Final Safety Analysis Report (UFSAR)
3. 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plants"

B 3.5 EMERGENCY CORE COOLING SYSTEM (ECCS)

B 3.5.2 ECCS - Operating

BASES

BACKGROUND

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

a. Loss of coolant accident;

b. Control element assembly ejection accident;

c. Secondary event, including uncontrolled steam release or excess feedwater heat removal event; and

d. Steam generator tube rupture.

The addition of negative reactivity by the ECCS during a secondary event where primary cooldown could add enough positive reactivity to achieve criticality and return to significant power was considered in design requirements for the ECCS.

There are two phases of ECCS operation: injection and recirculation. In the injection phase, all injection is initially added to the RCS via the cold legs. After the refueling water tank (RWT) has been depleted, the ECCS recirculation phase is entered as the ECCS suction is automatically transferred to the containment sump.

Two redundant, 100% capacity trains are provided. In MODEs 1 and 2, and MODE 3 with pressurizer pressure ≥ 1750 psia, each train consists of HPSI and LPSI charging subsystems. In MODEs 1 and 2, and MODE 3 with pressurizer pressure ≥ 1750 psia, both trains must be OPERABLE with the exception that, when THERMAL POWER is ≤ 80% RATED THERMAL POWER (RTP), the charging pumps are not required to be OPERABLE to support the ECCS function. This ensures that 100% of the core cooling requirements can be provided in the event of a single active failure.

A suction header supplies water from the RWT or the containment sump to the ECCS pumps. Separate piping supplies each train. The discharge headers from each HPSI pump divide into four supply lines. Both HPSI trains feed into each of the four injection lines. The discharge header which is fed from both LPSI pumps divides into four supply lines, each feeding the injection line to each RCS cold leg.

For LOCAs that are too small to initially depressurize the RCS below the shutoff head of the HPSI pumps, the charging pumps supply water to maintain inventory until the RCS pressure decreases below the HPSI pump shutoff head (the charging subsystem is only credited with supporting the ECCS function when THERMAL POWER is > 80% RTP). During this period, the steam generators must provide the core cooling function. The charging pumps take suction from the boric acid tank, via the boric acid pumps or gravity feed, on a SIAS, and discharge directly to the RCS through the charging line. If the charging line is unavailable the charging pump can be manually aligned to discharge through the HPSI piping. The normal supply source (volume control tank) for the charging pumps is isolated on a SIAS to prevent noncondensible gas (e.g., air, nitrogen, or hydrogen) from being entrained in the charging pumps.

During low temperature conditions in the RCS, limitations are placed on the maximum number of HPSI pumps that may be OPERABLE. Refer to LCO 3.4.12 Bases, for the basis of these requirements.

During a large break LOCA, RCS pressure will decrease to < 200 psia in < 20 seconds. The safety injection systems are actuated upon receipt of a SIAS. If offsite power is available, the safeguard loads start immediately. If offsite power is not available, the engineered safety feature (ESF) buses shed normal operating loads and are connected to the diesel generators. Safeguard loads are then actuated in the programmed time sequence. The time delay associated with diesel starting, sequenced loading, and pump starting determines the time required before pumped flow is available to the core following a LOCA.

The active ECCS components, along with the passive SITs and RWT, covered in LCO 3.5.1 and LCO 3.5.4, provide the cooling water necessary to meet Reference 1, Appendix 1C, Criterion 44.

APPLICABLE SAFETY ANALYSES

The LCO helps to ensure that the following acceptance criteria, established by Reference 2 for ECCSs, will be met following a LOCA:

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

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

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

d. Core is maintained in a coolable geometry; and

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

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

Both HPSI and LPSI subsystems are assumed to be OPERABLE in the large break LOCA analysis at full power (Reference 1, Section 14.17). This analysis establishes a minimum required runout flow for the HPSI and LPSI pumps, as well as the maximum required response time for their actuation. The HPSI pumps and charging pumps are credited in the small break LOCA analysis. This analysis establishes the flow and discharge head requirements at the design point for the HPSI pump. The steam generator tube rupture and steam line break analyses also credit the HPSI pumps, but are not limiting in their design.

The large break LOCA event with a loss of offsite power and a single failure (disabling one ECCS train) establishes the OPERABILITY requirements for the ECCS. During the blowdown stage of a LOCA, the RCS depressurizes as primary coolant is ejected through the break into Containment. The nuclear reaction is terminated either by moderator voiding during large breaks or control element assembly insertion during small breaks. Following depressurization, emergency cooling water is injected into the cold legs, flows into the downcomer, fills the lower plenum, and refloods the core.

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

The LCO ensures that an ECCS train will deliver sufficient water to match decay heat boiloff rates soon enough to minimize core uncovery for a large LOCA. It also ensures that the accident assumptions are met for the small break LOCA and steam line break. For smaller LOCAs the charging pumps deliver sufficient fluid to maintain RCS inventory until the RCS can be depressurized below the HPSI pumps' shutoff head. During this period of a small break LOCA, the steam generators continue to serve as the heat sink providing core cooling.

Emergency Core Cooling System - Operating satisfies 10 CFR 50.36(c)(2)(ii), Criterion 3.

LCO

In MODEs 1, 2, and 3, with pressurizer pressure ≥ 1750 psia, two independent (and redundant) ECCS trains are required to ensure that sufficient ECCS flow is available, assuming there is a single failure affecting either train. Additionally, individual components within the ECCS trains may be called upon to mitigate the consequences of other transients and accidents.

In MODEs 1 and 2, and in MODE 3 with pressurizer pressure ≥ 1750 psia, an ECCS train consists of a HPSI subsystem, a LPSI subsystem, and a charging pump.

Each HPSI and LPSI train includes the piping, instruments, and controls to ensure the availability of an OPERABLE flow path capable of taking suction from the RWT on a SIAS and automatically transferring suction to the containment sump upon a recirculation actuation signal. The charging pump receives inventory from the boric acid tank.

During an event requiring ECCS actuation, a flow path is provided to ensure an abundant supply of water from the RWT to the RCS, via the HPSI and LPSI pumps and their respective supply headers, to each of the four cold leg injection nozzles. In the long-term, this flow path may be switched to take its supply from the containment sump and to supply part of its flow to the RCS hot legs via the pressurizer or the shutdown cooling (SDC) suction nozzles. The acceptable charging pump flow paths are:

a. A boric acid storage tank (Unit 1, Tank No. 12; Unit 2, Tank No. 22), via a boric acid pump (Unit 1, Pump No. 12; Unit 2, Pump No. 22) to the charging pump, and

b. Either boric acid storage tank (Unit 1, Tank Nos. 11 or 12; Unit 2, Tank Nos. 21 or 22) via a gravity feed connection to the charging pump.

The charging pump flow path supplies the RCS via the normal charging lines.

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

In addition for the HPSI pump system to be considered OPERABLE, each HPSI pump system (consisting of a HPSI pump and one of two safety injection headers) must have balanced flows, such that the sum of the flow rates of the three lowest flow legs is > 470 gpm.

APPLICABILITY

In MODEs 1 and 2, and in MODE 3 with RCS pressure ≥ 1750 psia, the ECCS OPERABILITY requirements for the limiting DBA large break LOCA are based on full power operation. Although reduced power would not require the same level of performance, the accident analysis does not provide for reduced cooling requirements in the lower MODEs. The HPSI pump performance is based on the small break LOCA, which establishes the pump performance curve and has less dependence on power. The charging pump performance requirements are based on a small break LOCA when THERMAL POWER is > 80% RTP. In MODE 1, when THERMAL POWER is ≤ 80% RTP, and in MODEs 2 and 3, the charging pumps are not required to be OPERABLE since there is a corresponding decrease in decay heat in these conditions which compensates for the loss of injection from the charging pump assumed in the small break LOCA analyses. The requirements of MODE 2,

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and MODE 3 with RCS pressure ≥ 1750 psia, are bounded by the MODE 1 analysis.

The ECCS functional requirements of MODE 3, with RCS pressure < 1750 psia, and MODE 4 are described in LCO 3.5.3.

In MODEs 5 and 6, unit conditions are such that the probability of an event requiring ECCS injection is extremely low. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7 and LCO 3.4.8. MODE 6 core cooling requirements are addressed by LCO 3.9.4 and LCO 3.9.5.

ACTIONS

A.1

If one or more trains are inoperable and at least 100% of the ECCS flow equivalent to a single OPERABLE ECCS train is available, the inoperable components must be returned to OPERABLE status within 72 hours. The 72 hour Completion Time is based on an Nuclear Regulatory Commission study (Reference 3) using a reliability evaluation and is a reasonable amount of time to effect many repairs.

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

The LCO requires the OPERABILITY of a number of independent subsystems. Due to the redundancy of trains and the diversity of subsystems, the inoperability of one component in a train does not render the ECCS incapable of performing its function. Neither does the inoperability of two different components, each in a different train, necessarily result in a loss of function for the ECCS. The intent of this Condition is to maintain a combination of OPERABLE equipment such that 100% of the ECCS flow equivalent to 100% of a single OPERABLE train remains available. This allows increased flexibility in plant operations when components in opposite trains are inoperable.

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An event accompanied by a loss of offsite power and the failure of an emergency diesel generator can disable one ECCS train until power is restored. A reliability analysis (Reference 3) has shown that the impact with one full ECCS train inoperable is sufficiently small to justify continued operation for 72 hours.

Reference 4 describes situations in which one component, such as a SDC total flow control valve, can disable both ECCS trains. With one or more components inoperable, such that 100% of the equivalent flow to a single OPERABLE ECCS train is not available, the facility is in a condition outside the accident analyses. Therefore, LCO 3.0.3 must be immediately entered.

B.1 and B.2

If the inoperable train cannot be restored to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and pressurizer pressure reduced to < 1750 psia within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.5.2.1

Verification of proper valve position ensures that the flow path from the ECCS pumps to the RCS is maintained. Misalignment of these valves could render both ECCS trains inoperable. Securing these valves in position by interrupting the control signal to the valve operator, ensures that the valves cannot be inadvertently misaligned. These valves are of the type described in Reference 4, which can disable the function of both ECCS trains and invalidate the accident analysis. A 12 hour Frequency is considered reasonable in view of other administrative controls ensuring that a mispositioned valve is an unlikely possibility.

SR 3.5.2.2

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

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

SR 3.5.2.3

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

SR 3.5.2.4

Verification of design flow is a normal test of charging pump performance required by American Society of Mechanical Engineers Code, Section XI. The quarterly Frequency for this test is a Code requirement. This inservice inspection detects component degradation and incipient failures.

SR 3.5.2.5, SR 3.5.2.6, and SR 3.5.2.7

These SRs demonstrate that each automatic ECCS valve actuates to the required position on an actual, or simulated SIAS, and on a recirculation actuation signal; that each ECCS pump starts on receipt of an actual or simulated SIAS; and that the LPSI pumps stop on receipt of an actual or simulated recirculation actuation signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. In order to assure the results of the low temperature overpressure protection analysis remain bounding, whenever flow testing into the RCS is required at RCS temperatures ≤ 365°F (Unit 1), ≤ 301°F (Unit 2), the HPSI pump shall recirculate RCS water (suction from the RWT isolated) or the requirements of LCO 3.4.12, shall be satisfied. The 24 month Frequency is based on the need to perform these surveillance tests under the conditions that apply during a plant outage and the potential for unplanned transients if the surveillance tests were performed with the reactor at power. The 24 month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment. The actuation logic is tested as part of the Engineered Safety Feature Actuation System testing, and equipment performance is monitored as part of the Inservice Testing Program.

SR 3.5.2.8

Periodic inspection of the containment sump ensures that it is unrestricted and stays in proper operating condition. The 24 month Frequency is based on the need to perform this surveillance test under the conditions that apply during an outage, on the need to have access to the location, and on the potential for unplanned transients if the surveillance test were performed with the reactor at power. This

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Frequency is sufficient to detect abnormal degradation and is confirmed by operating experience.

SR 3.5.2.9

Verifying that the SDC System open-permissive interlock is OPERABLE ensures that the SDC suction isolation valves are prevented from being remotely opened when RCS pressure, is at or above, the SDC System design suction pressure of 350 psia. The suction piping of the LPSI pumps, is the SDC component with the limiting design pressure rating. The interlock provides assurance that double isolation of the SDC System from the RCS is preserved whenever RCS pressure, is at or above, the design pressure. The 309 psia value specified in the Surveillance is the actual pressurizer pressure at the instrument tap elevation for PT-103 and PT-103-1 when the SDC System suction pressure is 350 psia. The procedure for this surveillance test contains the required compensation to be applied to this value to account for instrument uncertainties. This surveillance test is normally performed using a simulated RCS pressure input signal. The 24 month Frequency is based on the need to perform this surveillance test under conditions that apply during an outage. The 24 month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) for the equipment.

REFERENCES

1. UFSAR
2. 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plants"
3. Nuclear Regulatory Commission Memorandum to V. Stello, Jr., from R. L. Baer, "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975
4. Inspection and Enforcement Information Notice No. 87-01, "RHR Valve Misalignment Causes Degradation of ECCS in PWRs," January 6, 1987

B 3.5 EMERGENCY CORE COOLING SYSTEM (ECCS)

B 3.5.3 ECCS - Shutdown

BASES

BACKGROUND

The Background Section for B 3.5.2, is applicable to these Bases, with the following modification.

In MODE 3 with pressurizer pressure < 1750 psia and in MODE 4, an ECCS train is defined as one HPSI subsystem. The HPSI flow path consists of piping, valves, and pumps that enable water from the RWT to be injected into the RCS following the accidents described in B 3.5.2.

APPLICABLE SAFETY ANALYSES

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

Due to the stable conditions associated with operation in MODE 3 with RCS pressure < 1750 psia and MODE 4, and the reduced probability of a DBA, the ECCS operational requirements are reduced. Included in these reductions is that certain automatic SIASs are not available. In this MODE, sufficient time exists for manual actuation of the required ECCS to mitigate the consequences of a DBA.

Only one train of ECCS is required for MODE 3 with RCS pressure < 1750 psia and MODE 4. Protection against single failures is not relied on for this MODE of operation.

Emergency Core Cooling System - Shutdown satisfies 10 CFR 50.36(c)(2)(ii), Criterion 3.

LCO

In MODE 3 with pressurizer pressure < 1750 psia and MODE 4, an ECCS subsystem is composed of a single HPSI subsystem. Each HPSI subsystem includes the piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the RWT and transferring suction to the containment sump.

During an event requiring ECCS actuation, a flow path is required to supply water from the RWT to the RCS via the HPSI pumps and their respective supply headers to each of the four cold leg injection nozzles. In the long-term, this flow path may be switched to take its supply from the

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

With RCS pressure < 1750 psia, one HPSI pump is acceptable without single failure consideration, based on the stable reactivity condition of the reactor, and the limited core cooling requirements. The LPSI pumps may therefore be released from the ECCS train for use in SDC. In MODE 3 with RCS cold leg temperature ≤ 365°F (Unit 1), ≤ 301°F (Unit 2), a maximum of one HPSI pump is allowed to be OPERABLE in accordance with LCO 3.4.12.

The LCO is modified by a Note which allows the HPSI train to not be capable of automatically starting on an actuation signal when RCS cold leg temperature is < 385°F (Unit 1), < 325°F (Unit 2), during heatup and cooldown and when < 365°F (Unit 1), < 301°F (Unit 2), during other conditions. This allowance is necessary to ensure low temperature overpressure protection analysis assumptions are maintained. The LCO Note provides a transition period [between 385°F and 365°F (Unit 1), between 325°F and 301°F (Unit 2)] where the OPERABLE HPSI pump will be placed in pull-to-lock on a cooldown and restored to automatic status on heatup (see LCO 3.4.12). At 365°F and less (Unit 1), 301°F and less (Unit 2), the required HPSI pump shall be placed in pull-to-lock and will not start automatically. The HPSI pumps and HPSI header isolation valves are required to be out of automatic when operating within the MODEs of Applicability for the Low Temperature Overpressure Protection System (LCO 3.4.12).

APPLICABILITY

In MODEs 1, 2, and 3 with RCS pressure ≥ 1750 psia, the OPERABILITY requirements for ECCS are covered by LCO 3.5.2.

In MODE 3 with RCS pressure < 1750 psia and in MODE 4, one OPERABLE ECCS train is acceptable without single failure consideration, based on the stable reactivity condition of the reactor, and the limited core cooling requirements.

In MODEs 5 and 6, unit conditions are such that the probability of an event requiring ECCS injection is extremely low. Core cooling requirements in MODE 5 are

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addressed by LCO 3.4.7 and LCO 3.4.8. MODE 6 core cooling requirements are addressed by LCO 3.9.4 and LCO 3.9.5.

ACTIONS

A.1

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

B.1

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

SURVEILLANCE REQUIREMENTS

SR 3.5.3.1

The applicable SR descriptions from B 3.5.2 apply.

REFERENCES

The applicable references from B 3.5.2 apply.

B 3.5 EMERGENCY CORE COOLING SYSTEM (ECCS)

B 3.5.4 Refueling Water Tank (RWT)

BASES

BACKGROUND

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

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

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

This LCO ensures that:

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

b. Sufficient water volume exists in the containment sump to support continued operation of the ESF pumps at the

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time of transfer to the recirculation mode of cooling; and

c. The reactor remains subcritical following a LOCA.

Insufficient water inventory in the RWT could result in insufficient cooling capacity of the ECCS when the transfer to the recirculation mode occurs. Improper boron concentrations could result in a reduction of SDM or excessive boric acid precipitation in the core following a LOCA, as well as excessive caustic stress corrosion of mechanical components and systems inside Containment.

APPLICABLE SAFETY ANALYSES

During accident conditions, the RWT provides a source of borated water to the HPSI, LPSI, containment spray, and charging pumps when level is low in the boric acid tanks. As such, it provides containment cooling and depressurization, core cooling, and replacement inventory, and is a source of negative reactivity for reactor shutdown (Reference 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 and B 3.6.6. These analyses are used to assess changes to the RWT in order to evaluate their effects in relation to the acceptance limits.

The volume limit of 400,000 gallons is based on two factors:

a. Sufficient deliverable volume must be available to provide at least 36 minutes (plus a 10% margin) of full flow from all ESF pumps prior to reaching a low level switchover to the containment sump for recirculation; and

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

When ESF pump suction is transferred to the sump, there must be sufficient water in the sump to ensure adequate net positive suction head for the HPSI and containment spray pumps. The RWT capacity must be sufficient to supply this

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amount of water without considering the inventory added from the SITs or RCS, but accounting for loss of inventory to containment subcompartments and reservoirs due to containment spray operation and to areas outside containment due to leakage from ECCS injection and recirculation equipment.

The 2300 ppm limit for minimum boron concentration was established to ensure that, following a LOCA with a minimum level in the RWT, the reactor will remain subcritical in the cold condition following mixing of the RWT and RCS water volumes with all control rods inserted, except for the control element assembly of highest worth, which is withdrawn from the core. The most limiting case occurs at beginning of core life.

The maximum boron limit of 2700 ppm in the RWT is based on boron precipitation in the core following a LOCA. With the reactor vessel at saturated conditions, the core dissipates heat by pool nucleate boiling. Because of this boiling phenomenon in the core, the boric acid concentration will increase in this region. If allowed to proceed in this manner, a point will be reached where boron precipitation will occur in the core. Post-LOCA emergency procedures direct the operator to establish simultaneous hot and cold leg injection to prevent this condition by establishing a forced flow path through the core regardless of break location. These procedures are based on the minimum time in which precipitation could occur, assuming that maximum boron concentrations exist in the borated water sources used for injection following a LOCA. Boron concentrations in the RWT in excess of the limit could result in precipitation earlier than assumed in the analysis.

The upper limit of 100°F (only required for MODE 1 operation) and the lower limit of 40°F (RWT temperature), are the limits assumed in the accident analysis.

The RWT satisfies 10 CFR 50.36(c)(2)(ii), Criterion 3.

LCO

The RWT ensures that an adequate supply of borated water is available to: cool and depressurize the Containment in the event of a DBA, to cool and cover the core in the event of a

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LOCA, ensure that the reactor remains subcritical following a DBA, and ensure that an adequate level exists in the containment sump to support ESF pump operation in the recirculation mode.

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

APPLICABILITY

In MODEs 1, 2, 3, and 4, the RWT OPERABILITY requirements are dictated by the ECCS and Containment Spray System OPERABILITY requirements. Since both the ECCS and the Containment Spray System must be OPERABLE in MODEs 1, 2, 3, and 4, the RWT must be OPERABLE to support their operation.

Core cooling requirements in MODE 5 are addressed by LCO 3.4.7 and LCO 3.4.8. MODE 6 core cooling requirements are addressed by LCO 3.9.4 and LCO 3.9.5.

ACTIONS

A.1

With RWT boron concentration or borated water temperature not within limits, it must be returned to within limits within eight hours. In this condition neither the ECCS nor the Containment Spray System can perform their design functions; therefore, prompt action must be taken to restore the tank to OPERABLE condition. The allowed Completion Time of eight hours to restore the RWT to within limits was developed considering the time required to change boron concentration or temperature, and that the contents of the tank are still available for injection.

Required Action A.1 only applies to the maximum borated water temperature in MODE 1.

B.1

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

C.1 and C.2

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

SURVEILLANCE REQUIREMENTS

SR 3.5.4.1 and SR 3.5.4.2

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

The SRs are modified by a Note that eliminates the requirement to perform this surveillance test when ambient air temperatures are within the operating temperature limits of the RWT. With ambient temperatures within this range, the RWT temperature should not exceed the limits.

Surveillance Requirement 3.5.4.2 is modified by an additional Note which requires the SR to be met in MODE 1 only. A SR is "met" only when the acceptance criteria are satisfied. Known failure of the requirements of a SR, even without a surveillance test specifically being "performed," constitutes a SR not "met." This reflects the maximum coolant temperature assumptions in the LOCA analysis.

SR 3.5.4.3

Above minimum RWT water volume level shall be verified every seven days. This Frequency ensures that a sufficient initial water supply is available for injection and to support continued ESF pump operation on recirculation.

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Since the RWT volume is normally stable and is provided with a Low Level Alarm, a seven day Frequency is appropriate and has been shown to be acceptable through operating experience.

SR 3.5.4.4

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

REFERENCES

1. UFSAR, Chapters 6, "Engineered Safety Features," and 14, "Safety Analysis"

B 3.5 EMERGENCY CORE COOLING SYSTEM (ECCS)

B 3.5.5 Trisodium Phosphate (TSP)

BASES

BACKGROUND

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

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

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

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

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

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

APPLICABLE SAFETY ANALYSES

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

Trisodium phosphate satisfies 10 CFR 50.36(c)(2)(ii), Criterion 3.

LCO

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

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

APPLICABILITY

In MODEs 1, 2, 3, and 4, the RCS is at elevated temperature and pressure, providing an energy potential for a LOCA. The potential for a LOCA results in a need for the ability to control the pH of the recirculated coolant.

In MODEs 5 and 6, the potential for a LOCA is reduced or non-existent due to the reduced pressure and temperature limitations of these MODEs, and TSP is not required.

ACTIONS

A.1

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

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

B.1 and B.2

If the TSP cannot be restored within limits within the Completion Time of Required Action A.1, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least

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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 in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.5.5.1

Periodic determination of the volume of TSP in Containment must be performed due to the possibility of leaking valves and components in the Containment Building that could cause dissolution of the TSP during normal operation. A Frequency of 24 months is required to determine visually that a minimum of 289.3 cubic feet is contained in the TSP baskets. This requirement ensures that there is an adequate volume of TSP to adjust the pH of the post-LOCA sump solution to a value ≥ 7.0.

The periodic verification is required every 24 months, since access to the TSP baskets is only feasible during outages, and normal fuel cycles are scheduled for 24 months. Operating experience has shown this SR Frequency acceptable, due to the margin in the volume of TSP placed in the Containment Building.

SR 3.5.5.2

Testing must be performed to ensure the solubility and buffering ability of the TSP after exposure to the containment environment. A representative sample of 3.43 ± 0.05 grams of TSP, from one of the baskets in Containment is submerged in 1.0 ± 0.01 liters of water at a boron concentration of 3106 ± 50 ppm, and at the standard temperature of 120 ± 5°F. After four hours without agitation, the solution is decanted and mixed, the temperature adjusted to 77 ± 2°F, and the pH measured. The solution pH should be ≥ 6.0. The representative sample weight is based on the minimum required TSP weight of 14,371 lbm, which at manufactured density corresponds to the minimum volume of 289.3 cubic feet, and maximum possible post-LOCA sump volume of 4,503,500 lbm, normalized to buffer a 1.0 ± 0.01 liter sample. The boron concentration of the test water is representative of the maximum possible boron

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

REFERENCES None

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1 Containment

BASES

BACKGROUND

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

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

The concrete building is required for structural integrity of the Containment Structure under DBA conditions. The steel liner and its penetrations establish the leakage limiting boundary of the Containment Structure. Maintaining the Containment Structure OPERABLE limits the leakage of fission product radioactivity from the Containment Structure to the environment. Surveillance Requirement (SR) 3.6.1.1 leakage rate requirements comply with Reference 1, as modified by approved exemptions.

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

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

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

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2. closed by manual valves, blind flanges, or de-activated automatic valves secured in their closed positions, except as provided in Limiting Condition for Operation (LCO) 3.6.3;

b. Each air lock is OPERABLE, except as provided in LCO 3.6.2;

c. The equipment hatch is closed and sealed.

APPLICABLE SAFETY ANALYSES

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

The DBAs that result in a release of radioactive material within Containment Structure are a loss of coolant accident (LOCA), a main steam line break (SLB), and a control element assembly (CEA) ejection accident (Reference 2, Chapter 14). In the analysis of each of these accidents, it is assumed that Containment Structure is OPERABLE, such that release of fission products to the environment is controlled by the rate of Containment Structure leakage. The Containment Structure was designed with an allowable leakage rate of 0.20% of containment air weight per day (Reference 2, Chapter 5). This leakage rate is defined in Reference 1, as L_a: the maximum allowable containment leakage rate at the calculated maximum peak containment pressure (P_a) of 49.4 psig, which results from the limiting design basis LOCA (Reference 2, Chapter 14).

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

The Containment Structure satisfies 10 CFR 50.36(c)(2)(ii), Criterion 3.

LCO

Containment OPERABILITY is maintained by limiting leakage to $\leq 1.0\ L_a$ (346,000 SCCM), 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.

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Compliance with this LCO will ensure a containment configuration, including an equipment hatch, 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) are not specifically part of the acceptance criteria of Reference 1. Therefore, leakage rates exceeding these individual limits only result in the Containment Structure being inoperable when the leakage results in exceeding the overall acceptance criteria of 1.0 L_a.

APPLICABILITY

In MODEs 1, 2, 3, and 4, a DBA could cause a release of radioactive material into the Containment Structure. 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 Structure is not required to be OPERABLE in MODE 5 to prevent leakage of radioactive material from the Containment Structure. The requirements for the Containment Structure during MODE 6 are addressed in LCO 3.9.3.

ACTIONS

A.1

In the event the Containment Structure is inoperable, the Containment Structure must be restored to OPERABLE status within one hour. The one hour Completion Time provides a period of time to correct the problem commensurate with the importance of maintaining the Containment Structure 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 the Containment Structure is inoperable is minimal.

B.1 and B.2

If the Containment Structure 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

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

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1

Maintaining the Containment Structure OPERABLE requires compliance with the visual examinations and leakage rate test requirements of the Containment Leakage Rate Testing Program. Failure to meet 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, is required to be $\leq 0.6\ L_a$ (207,600 SCCM) for combined Type B and C leakage and $\leq 0.75\ L_a$ (259,500 SCCM) 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. Surveillance Requirement Frequencies are as required by 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.

Additionally, the requirements regarding the Unit 1 containment purge isolation valves must be met.

SR 3.6.1.2

For ungrouted, post-tensioned tendons, this SR ensures that the structural integrity of the Containment Structure will be maintained in accordance with the provisions of the Concrete Containment Tendon Surveillance Program. Testing and Frequency are consistent with the recommendations of Reference 3.

BASES

REFERENCES	1.	10 CFR Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors" Option B, "Performance-Based Requirements"
	2.	Updated Final Safety Analysis Report (UFSAR)
	3.	Regulatory Guide 1.35, Revision 2, "Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containments," January 1976

B 3.6 CONTAINMENT SYSTEMS

B 3.6.2 Containment Air Locks

BASES

BACKGROUND

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

Each air lock is nominally a right circular cylinder, 9 feet-9 inches in diameter for the personnel air lock and 5 feet-9 inches in diameter for the emergency air lock, with a door at each end. The doors are interlocked to prevent simultaneous opening. During periods when the Containment Structure 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 Structure 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 DBA in the Containment Structure. As such, closure of a single door supports the Containment Structure OPERABILITY. Each of the doors contains double gasketed seals and local leakage rate testing capability to ensure pressure integrity. To effect a leak tight seal, the air lock design uses pressure seated doors (i.e., an increase in containment internal pressure results in increased sealing force on each door).

Each personnel air lock is provided with an alarm in the Control Room that actuates when either door or equalizing valve for a personnel air lock is opened. The alarm senses door position from a limit switch located on each door and equalizing valve.

The containment air locks form part of the containment pressure boundary. As such, air lock integrity and leak tightness is essential for maintaining the containment leakage rate within limit in the event of a DBA. Not maintaining air lock integrity or leak tightness may result in a leakage rate in excess of that assumed in the unit safety analysis.

BASES

APPLICABLE SAFETY ANALYSES

The DBAs that result in a release of radioactive material within the Containment Structure are a LOCA, a main SLB, and a CEA ejection accident (Reference 1, Chapter 14). In the analysis of each of these accidents, it is assumed that the Containment Structure is OPERABLE such that release of fission products to the environment is controlled by the rate of containment leakage. The Containment Structure was designed with an allowable leakage rate of 0.20% of containment air weight per day (Reference 1, Chapter 5). This leakage rate is defined in 10 CFR Part 50, Appendix J, Option B, as the maximum allowable containment leakage rate at the calculated peak containment internal pressure, P_a (49.4 psig), following a design basis LOCA. This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air lock.

The containment air locks satisfy 10 CFR 50.36(c)(2)(ii), Criterion 3.

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 the Containment Structure does not exist when the Containment Structure 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 the Containment Structure.

APPLICABILITY

In MODEs 1, 2, 3, and 4, a DBA could cause a release of radioactive material to the containment atmosphere. 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 the Containment Structure. The requirements for the containment air locks during MODE 6 are addressed in LCO 3.9.3.

ACTIONS

The ACTIONS are modified by a Note that allows entry and exit to perform repairs on the affected air lock component. If the outer door is inoperable, then it may be easily accessed for most repairs. It is preferred that the air lock be accessed from inside primary containment by entering through the other OPERABLE air lock. However, if this is not practicable, or if repairs on either door must be performed from the barrel side of the door then it is permissible to enter the air lock through the OPERABLE door, which means there is a short time during which the containment boundary is not intact (during access through the OPERABLE door). The ability to open the OPERABLE door, even if it means the containment boundary is temporarily not intact, is acceptable because of the low probability of an event that could pressurize the Containment Structure during the short time in which the OPERABLE door is expected to be open. After each entry and exit, the OPERABLE door must be immediately closed. If as low as reasonably achievable (ALARA) conditions permit, entry and exit should be via an OPERABLE air lock.

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

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

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

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

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

The Required Actions have been modified by two Notes. Note 1 ensures that only the Required Actions and associated Completion Times of Condition C are required if both doors in the same air lock are inoperable. With both doors in the same air lock inoperable, an OPERABLE door is not available to be closed. Required Actions C.1 and C.2 are the appropriate remedial actions. The exception to 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 seven days under administrative controls if both air locks have an inoperable door. This seven day restriction begins when the second air lock is discovered inoperable. Containment entry may be required to perform Technical Specifications Surveillances and Required Actions, as well as other activities on equipment inside the Containment Structure that are required by Technical Specifications or activities on equipment that support Technical Specifications-required equipment. This Note is not intended to preclude performing other activities (i.e., non-Technical Specifications-required activities) if the Containment Structure was entered, using the inoperable air lock, to perform an allowed activity listed above. This allowance is acceptable due to the low probability of an event that could pressurize the Containment Structure 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 is 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 the Containment Structure under the control of a dedicated individual stationed at the air lock, to ensure that only one door is opened at a time (i.e., the individual performs the function of the interlock).

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

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

With one or more air locks inoperable for reasons other than those described in Conditions 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 Structure 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), the Containment Structure remains OPERABLE, yet only one hour (per LCO 3.6.1) would be provided to restore the air lock door to OPERABLE status prior to requiring a plant shutdown. In addition, even with both doors failing the seal test, the overall containment leakage rate can still be within limits.

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

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at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.2.1

Maintaining containment air locks OPERABLE requires compliance with the leakage rate test requirements of 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 the Containment Structure OPERABILITY testing. The periodic testing requirements verify that the air lock leakage does not exceed the allowed fraction of the overall containment leakage rate. The Frequency is required by 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 of which is applicable to SR 3.6.1.1. This ensures that air lock leakage is properly accounted for in determining the combined Types 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 the Containment Structure OPERABILITY. Thus, the door interlock feature supports the Containment Support OPERABILITY while the air lock is being used for personnel transit into and out of the Containment Structure. Periodic testing of this interlock demonstrates that the interlock will function as

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designed and that simultaneous opening of the inner and outer doors will not inadvertently occur. Due to the purely mechanical nature of this interlock, and given that the interlock mechanism is not normally challenged when the air lock is used for entry and exit (procedures require strict adherence to single door opening), this test is only required to be performed every 24 months. The 24 month Frequency is based on the need to perform this surveillance test under the conditions that apply during a plant outage and the potential for loss of the Containment Structure OPERABILITY if the surveillance test was performed with the reactor at power. The 24 month Frequency for the interlock is justified based on generic operating experience. The 24 month Frequency is based on engineering judgment and is considered adequate given that the interlock is not challenged during use of the air lock.

REFERENCES

1. UFSAR

B 3.6 CONTAINMENT SYSTEMS

B 3.6.3 Containment Isolation Valves

BASES

BACKGROUND

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

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

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

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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 Structure. Therefore, the safety analysis of any event requiring isolation of the Containment Structure is applicable to this LCO.

The DBAs that result in a release of radioactive material within the Containment Structure are a LOCA, a main SLB, and a CEA ejection accident. In the analysis for each of these accidents, it is assumed that containment isolation valves are either closed or function to close within the required isolation time following event initiation. This ensures that potential paths to the environment through containment isolation valves (including containment purge valves) are minimized. The safety analysis assumes that the purge valves are closed at event initiation.

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

The containment isolation valves satisfy 10 CFR 50.36(c)(2)(ii), Criterion 3.

LCO

Containment isolation valves form a part of the Containment Structure boundary. The containment isolation valve safety function is related to minimizing the loss of reactor coolant inventory and establishing the Containment Structure 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 valves covered by this LCO are listed with their associated stroke times in Reference 1.

The normally closed isolation valves are considered OPERABLE when manual valves are closed, automatic valves are de-activated and secured in their closed position, blind flanges are in place, and closed systems are intact. These passive isolation valves or devices are those listed in Reference 1.

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

APPLICABILITY

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

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 that appropriate remedial actions are taken, if necessary, if the affected systems are rendered inoperable by an inoperable containment isolation valve.

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

The fifth Note allows the shutdown cooling isolation valves to be opened when RCS temperature is < 300°F to establish shutdown cooling flow. This Note is required for Operation in MODE 4 to allow shutdown cooling to be established.

A.1 and A.2

In the event one containment isolation valve in one or more penetration flow paths is inoperable, the affected penetration flow path must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic containment isolation valve, a closed manual valve, a blind flange, and a check valve with flow through the valve secured. For penetrations isolated in accordance with Required Action A.1, the device used to isolate the penetration should be the closest available one to the Containment Structure. Required Action A.1 must be completed within the four hour Completion Time. The four hour Completion Time is reasonable, considering the time required to isolate the penetration and the relative importance of supporting the Containment Structure OPERABILITY during MODEs 1, 2, 3, and 4.

For affected penetration flow paths that cannot be restored to OPERABLE status within the four hour Completion Time and that have been isolated in accordance with Required Action A.1, the affected penetration flow paths must be verified to be isolated on a periodic basis. This is necessary to ensure that containment penetrations required to be isolated following an accident and no longer capable of being automatically isolated, will be in the isolation position should an event occur. This Required Action does not require any testing or device manipulation. Rather, it

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involves verification, through a system walkdown, that those isolation devices outside the Containment Structure 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 the Containment Structure, the time period specified as "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is based on engineering judgment and is considered reasonable in view of the inaccessibility of the isolation devices and other administrative controls that will ensure that isolation device misalignment is an unlikely possibility.

Condition A has been modified by a Note indicating that this Condition is only applicable to those penetration flow paths with two containment isolation valves and not a closed system. For penetration flow paths with one or more containment isolation valves and a closed system, Condition C provides appropriate actions.

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

B.1

With two containment isolation valves in one or more penetration flow paths inoperable, the affected penetration flow path must be isolated within one 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 one 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

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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 the Containment Structure and that penetrations requiring isolation following an accident are isolated. The Completion Time of once per 31 days for verifying each affected penetration flow path is isolated, is appropriate, considering the fact that the valves are operated under administrative controls and the probability of their misalignment is low.

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

C.1 and C.2

With one or more containment isolation valves inoperable in one or more penetration flow paths, the inoperable valves must be restored to OPERABLE status or the affected penetration flow path must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, a closed manual valve, and a blind flange. A check valve may not be used to isolate the affected penetration. Required Action C.1 must be completed within the 72 hour Completion Time. The specified time period is reasonable, considering the relative stability of the closed system (hence, reliability) to act as a penetration isolation boundary and the relative importance of supporting the Containment Structure OPERABILITY during MODEs 1, 2, 3, and 4. In the event the affected penetration is isolated in accordance with Required Action C.1, the affected penetration flow path must be verified to be isolated on a periodic basis. This is necessary to assure leak tightness of the Containment Structure and that containment penetrations requiring isolation following an accident are isolated. The Completion Time of once per 31 days for verifying that each affected penetration flow path is isolated, is appropriate considering the valves are operated under administrative controls and the probability of their misalignment is low.

Condition C is modified by a Note indicating that this Condition is only applicable to those penetration flow paths with one or more containment isolation valves and a closed system. This Note is necessary since this Condition is written to specifically address those penetration flow paths in a closed system. Containment isolation valves and their associated penetration numbers are given in Reference 1, Table 5.3. The penetrations with closed systems are listed below.

Penetration No.	Function
1B	Containment Vent Header to Waste Gas
16	Component Cooling Water Inlet
18	Component Cooling Water Outlet
19A	Instrument Air
20A	Nitrogen Supply
20B	Nitrogen Supply
20C	Nitrogen Supply
23	Reactor Coolant Drain Tank Drains
24	Oxygen Sample Line
38	Demineralized Water
44	Fire Protection

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

D.1 and D.2

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

SURVEILLANCE REQUIREMENTS

SR 3.6.3.1

This SR ensures that the containment vent valves are closed as required, or, if open, open for an allowable reason. If a containment vent 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 vent valves are open for pressure control, (ALARA) or air quality considerations for personnel entry, or for surveillance tests that require the valves to be open. The containment vent valves are capable of closing in the environment, following a LOCA. Therefore, these valves are allowed to be open for limited periods of time. The 31 day Frequency is consistent with other containment isolation valve requirements discussed in SR 3.6.3.2.

SR 3.6.3.2

This SR requires verification that each containment isolation manual valve and blind flange located outside the Containment Structure, 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 the containment boundary is within design limits. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those containment isolation valves outside the Containment Structure and capable of being mispositioned are in the correct position. Since verification of valve position for containment isolation valves outside the Containment Structure is relatively easy, the 31 day Frequency is based on engineering judgment, and was chosen to provide added assurance of the correct positions. Containment isolation valves that are open under administrative controls are not required to meet the SR during the time the valves are open. 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, 4 and for ALARA reasons. Therefore, the probability of misalignment of these containment isolation valves, once they have been verified to be in the proper position, is small.

SR 3.6.3.3

This SR requires verification that each containment isolation manual valve and blind flange located inside the Containment Structure, 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 the containment boundary is within design limits. For containment isolation valves inside the Containment Structure, the Frequency of "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is appropriate, since these containment isolation valves are operated under administrative controls and the probability of their misalignment is low. Containment isolation valves that are open under administrative controls are not required to meet the SR during the time that they are open. 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 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, and 3 for ALARA reasons. Therefore, the probability of misalignment of these containment isolation valves, once

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they have been verified to be in their proper position, is small.

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 analysis. The isolation time and Frequency of this SR are in accordance with the Inservice Testing Program.

SR 3.6.3.5

Automatic containment isolation valves close on an isolation signal [containment isolation signal Channels A or B, or safety injection actuation signal (SIAS) Channels A or B] to prevent leakage of radioactive material from the Containment Structure following a DBA. This SR ensures each automatic containment isolation valve will actuate to its isolation position on a containment isolation actuation signal. This surveillance test is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 24 month Frequency was developed considering it is prudent that this SR be performed only during a unit outage, since isolation of penetrations would eliminate cooling water flow and disrupt normal operation of many critical components. Operating experience has shown that these components usually pass this SR when performed on the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. UFSAR, Chapter 5, "Structures"

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4 Containment Pressure

BASES

BACKGROUND

The containment pressure is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a LOCA or main SLB. These limits also prevent the containment pressure from exceeding the containment design negative pressure differential, with respect to the outside atmosphere in the event of the Containment Structure being sealed during low barometric pressure and high temperature, then being exposed to a concurrent cooling of containment atmosphere and a barometric pressure rise.

Containment pressure is a process variable that is monitored and controlled. The containment pressure limits are derived from the input conditions used in the containment functional analyses and the containment structure external pressure analysis. Should operation occur above the upper limits coincident with a DBA, post-accident containment pressures could exceed calculated values. Should containment closure or integrity be set below the lower limits, the external pressure limits may be exceeded during barometric pressure changes.

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 DBA considered for determining the maximum containment internal pressure is the LOCA. A LOCA at 102% RATED THERMAL POWER and + 1.8 psig initial containment pressure results in the highest calculated internal containment pressure (P_a) below the internal design pressure of 50.0 psig. The postulated DBAs are analyzed assuming degraded containment ESF systems (i.e., assuming the loss of one ESF bus, which is the worst case single active failure, resulting in one train of the containment spray and one train of the containment coolers being rendered inoperable). It is this maximum containment pressure that is used to ensure that the licensing basis dose limitations are met.

The initial pressure condition used in the containment analysis was 16.5 psia (1.8 psig). The LCO limit of 1.8 psig ensures that, in the event of an accident, the maximum accident design pressure for the Containment Structure, 50 psig, is not exceeded. If a LOCA occurred while the containment internal pressure was at the LCO value of 1.8 psig, a total pressure below the design value of 50 psig would result.

Containment pressure satisfies 10 CFR 50.36(c)(2)(ii), Criterion 2.

LCO

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

APPLICABILITY

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

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

ACTIONS

A.1

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

BASES

B.1 and B.2

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

SURVEILLANCE REQUIREMENTS

SR 3.6.4.1

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

REFERENCES

None

B 3.6 CONTAINMENT SYSTEMS

B 3.6.5 Containment Air Temperature

BASES

BACKGROUND

The Containment Structure serves to contain radioactive material that may be released from the reactor core following a DBA. The Containment Structure average air temperature is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a LOCA or main 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 Structure by the containment spray and containment cooling during post-accident conditions is dependent on the energy released to the Containment Structure 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 (Reference 1). Operation with containment temperature in excess of the LCO limit violates an initial condition assumed in the accident analysis.

APPLICABLE SAFETY ANALYSES

Containment average air temperature is an initial condition used in the DBA analyses that establishes the containment environmental qualification operating envelope for both pressure and temperature. The limit for containment average air temperature ensures that operation is maintained within the assumptions used in the DBA analysis for Containment. The accident analyses and evaluations considered both LOCAs and main SLBs for determining the maximum peak containment pressures and temperatures. The worst case LOCA generates larger mass and energy releases than the worst case main SLB. Thus, the LOCA event bounds the main SLB event from the containment peak pressure and temperature standpoint. The initial pre-accident temperature inside the Containment Structure was assumed to be 120°F (Reference 1).

BASES

The initial containment average air temperature condition of 120°F resulted in a maximum vapor temperature described in Reference 1. The consequence of exceeding the design temperature for extended periods may be the potential for degradation of the containment structure under accident loads.

Containment average air temperature satisfies 10 CFR 50.36(c)(2)(ii), Criterion 2.

LCO

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

APPLICABILITY

In MODEs 1, 2, 3, and 4, a DBA could cause a release of radioactive material to the Containment Structure. 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 MODEs 5 or 6.

ACTIONS

A.1

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

B.1 and B.2

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

SURVEILLANCE REQUIREMENTS

SR 3.6.5.1

Verifying that containment average air temperature is within the LCO limit ensures that containment operation remains within the limit assumed for the containment analyses. In order to determine the containment average air temperature, an arithmetic average is calculated using measurements taken from the containment dome [1(2)-TI-5309] and the containment reactor cavity [1(2)-TI-5311] temperature indicators selected to provide a representative sample of the overall containment atmosphere. The 24 hour Frequency of this SR is considered acceptable based on the observed slow rates of temperature increase within the Containment Structure as a result of environmental heat sources (due to the large volume of the Containment Structure). Furthermore, the 24 hour Frequency is considered adequate in view of other indications available in the Control Room, including alarms, to alert the operator to an abnormal containment temperature condition.

REFERENCES

1. UFSAR, Section 14.20, "Containment Response"

B 3.6 CONTAINMENT SYSTEMS

B 3.6.6 Containment Spray and Cooling Systems

BASES

BACKGROUND

The Containment Spray and Cooling Systems provide containment atmosphere cooling to limit post-accident pressure and temperature in the Containment Structure to less than the design values. Reduction of containment pressure and the iodine removal capability of the spray, reduce the release of fission product radioactivity from the Containment Structure to the environment, in the event of a DBA, to within limits. The Containment Spray and Cooling Systems are designed to the requirements in Reference 1, Appendix 1C, Criteria, 58, 59, 60, 61, 62, 63, 64, and 65.

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

<u>Containment Spray System</u>

The Containment Spray System consists of two separate trains of equal capacity, each of sufficient capacity to supply approximately 50% of the design cooling requirement. Each train includes a containment spray pump, spray headers, nozzles, valves, and piping. Each train is powered from a separate ESF bus. The refueling water tank (RWT) supplies borated water to the containment spray during the injection phase of operation. In the recirculation mode of operation, containment spray pump suction is transferred from the RWT to the containment sump(s). Each spray system flow path from the containment sump will be via an OPERABLE shutdown cooling heat exchanger.

The Containment Spray System provides a spray of cold borated water into the upper regions of the Containment Structure to reduce containment pressure and temperature and to reduce the concentration of fission products in the containment atmosphere during a DBA. The RWT solution temperature is an important factor in determining the heat removal capability of the Containment Spray System during the injection phase. In the recirculation mode of operation, heat is removed from the containment sump water by the shutdown cooling heat exchangers. Each train of the Containment Spray System provides adequate spray coverage to meet 50% of the system design requirements for containment heat removal and 100% of the iodine removal design bases.

The Containment Spray System is actuated either automatically by a containment spray actuation signal coincident with a SIAS or manually. An automatic actuation starts the two containment spray pumps, and begins the injection phase. The containment spray header isolation valves open upon a containment spray actuation signal. A manual actuation of the Containment Spray System is available on the main control board to begin the same sequence. The injection phase continues until an RWT low level signal is received. The low level for the RWT generates a recirculation actuation signal that aligns valves from the containment spray pump suction to the containment sump. The Containment Spray System in recirculation mode maintains an equilibrium temperature between the containment atmosphere and the recirculated sump water. Operation of the Containment Spray System in the recirculation mode is controlled by the operator in accordance with the Emergency Operating Procedures.

Containment Cooling System

Two trains of containment cooling, each of sufficient capacity to supply approximately 67% of the design cooling requirement, are provided. Two trains with two fan units each are supplied with cooling water from a separate train of service water cooling. Three of the four fans are required to furnish the design cooling capacity. Air is drawn into the coolers through the fans and discharged throughout the Containment Structure.

In post-accident operation following a SIAS, all four Containment Cooling System fans are designed to start automatically in slow speed. Cooling is supplied by the service water cooled coils. The temperature of the service water is an important factor in the heat removal capability of the fan units.

BASES

APPLICABLE SAFETY ANALYSES

The Containment Spray and Cooling Systems limit the temperature and pressure that could be experienced following a DBA. The limiting DBAs considered relative to containment temperature and pressure are LOCA and main SLB. The DBA, LOCA, and main SLB are analyzed using computer codes designed to predict the resultant containment pressure and temperature transients. No DBAs are assumed to occur simultaneously or consecutively. The postulated DBAs are analyzed with regard to containment ESF systems, assuming the loss of one ESF bus, which is the worst case single active failure, resulting in one train of the Containment Spray System and one train of the Containment Cooling System being rendered inoperable.

The analysis and evaluation show that under the worst case scenario, the highest peak containment pressure and temperature are within the design. (See the Bases for Specifications 3.6.4 and 3.6.5 for a detailed discussion.) The analyses and evaluations assume a power level of 102% RATED THERMAL POWER, one containment spray train and one containment cooling train operating, and initial (pre-accident) conditions of 120°F and 16.5 psia. The analyses also assumes a response time delayed initiation, in order to provide a conservative calculation of peak containment pressure and temperature responses.

The modeled Containment Spray System actuation from the containment analysis is based upon a response time associated with exceeding the Containment High-High pressure setpoint coincident with an SIAS to achieve full flow through the containment spray nozzles. The Containment Spray System total response time of 60 seconds includes diesel generator startup (for loss of offsite power), sequencing equipment onto the emergency bus, containment spray pump startup, and spray line filling (Reference 1, Chapter 7).

The performance of the containment cooling train for post-accident conditions is given in Reference 1, Chapter 6. The results of the analysis, is that each train can provide approximately 67% of the required peak cooling capacity during the post-accident condition. The train post-accident cooling capacity under varying containment ambient

BASES

conditions, required to perform the accident analyses, is also shown in Reference 1, Chapter 6.

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

The Containment Spray and Cooling Systems satisfy 10 CFR 50.36(c)(2)(ii), Criterion 3.

LCO

During a DBA, a minimum of one containment cooling train and one containment spray train, is required to maintain the containment peak pressure and temperature, below the design limits (Reference 1, Chapter 6). Additionally, one containment spray train is also required to remove iodine from the containment atmosphere and maintain concentrations below those assumed in the safety analysis. To ensure that these requirements are met, two containment spray trains and two containment cooling trains (all four coolers) must be OPERABLE. Therefore, in the event of an accident, the minimum requirements are met, assuming that the worst case single active failure occurs.

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

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

APPLICABILITY

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

The Containment Spray System is only required to be OPERABLE in MODE 3 with pressurizer pressure ≥ 1750 psia.

In MODE 3 with pressurizer pressure < 1750 psia, and in MODEs 4, 5, and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODEs. Thus, the Containment Spray System is not required to be OPERABLE in MODE 3 with pressurizer pressure < 1750 psia, and the Containment Spray and Cooling Systems are not required to be OPERABLE in MODEs 4, 5, and 6.

ACTIONS

A.1

With one containment spray train inoperable, the inoperable containment spray train must be restored to OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE spray and cooling trains are adequate to perform the iodine removal and containment cooling functions. The 72 hour Completion Time takes into account the redundant heat removal capability afforded by the Containment Spray System, reasonable time for repairs, and the low probability of a DBA occurring during this period.

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

B.1 and B.2

If the inoperable containment spray train cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 3 with pressurizer pressure < 1750 psia within 12 hours. The allowed Completion Time of six hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner, and without challenging plant systems. The extended interval to reach MODE 3 with pressurizer pressure < 1750 psia allows additional time for the restoration of the containment spray train and is reasonable when considering that the driving force for a release of radioactive material from the RCS is reduced in MODE 3.

C.1

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

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

D.1

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

E.1 and E.2

If the Required Actions and associated Completion Times of Conditions C or D of this LCO are not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 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.

F.1

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

SURVEILLANCE REQUIREMENTS

SR 3.6.6.1

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

SR 3.6.6.2

Starting each containment cooling train fan unit from the Control Room and operating it for ≥ 15 minutes ensures that all trains are OPERABLE and that all associated controls are functioning properly. It also ensures that blockage, fan or motor failure, or excessive vibration can be detected and corrective action taken. The 31 day Frequency of this SR was developed considering the known reliability of the fan units and controls, the two train redundancy available, and the low probability of a significant degradation of the containment cooling train occurring between surveillances and has been shown to be acceptable through operating experience.

SR 3.6.6.3

Verifying a service water flow rate of ≥ 2000 gpm to each cooling unit when the full flow service water outlet valves are fully open provides assurance that the design flow rate assumed in the safety analyses will be achieved (Reference 1, Chapter 7). Also considered in selecting this Frequency were the known reliability of the Service Water System, the two train redundancy, and the low probability of a significant degradation of flow occurring between surveillance tests.

SR 3.6.6.4

Verifying that each containment spray pump's developed head at the flow test point is greater than or equal to the required developed head ensures that spray pump performance has not degraded during the cycle. Flow and differential pressure are normal tests of centrifugal pump performance required by Reference 2. Since the containment spray pumps cannot be tested with flow through the spray headers, they are tested on recirculation flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice inspections confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of this SR is in accordance with the Inservice Testing Program.

SR 3.6.6.5 and SR 3.6.6.6

These SRs verify that each automatic containment spray valve actuates to its correct position and that each containment spray pump starts upon receipt of an actual or simulated actuation signal (i.e., the appropriate Engineered Safety Feature Actuation System signal). This SR is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The

24 month Frequency is based on the need to perform these surveillance tests under the conditions that apply during a plant outage and the potential for an unplanned transient if the surveillance tests were performed with the reactor at power. Operating experience has shown that these components usually pass the surveillance tests when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

The surveillance test of containment sump isolation valves is also required by SR 3.5.2.5. A single surveillance test may be used to satisfy both requirements.

SR 3.6.6.7

This SR verifies that each containment cooling train actuates upon receipt of an actual or simulated actuation signal (i.e., the appropriate Engineered Safety Feature Actuation System signal). The 24 month Frequency is based on engineering judgment and has been shown to be acceptable through operating experience. See SR 3.6.6.5 and SR 3.6.6.6, above, for further discussion of the basis for the 24 month Frequency.

SR 3.6.6.8

With the containment spray inlet valves closed and the spray header drained of any solution, low pressure air or smoke can be blown through check valve bonnets. Performance of this SR demonstrates that each spray nozzle is unobstructed and provides assurance that spray coverage of the Containment Structure during an accident is not degraded. Due to the passive design of the nozzle, a test at ten year intervals is considered adequate to detect obstruction of the spray nozzles.

REFERENCES

1. UFSAR

2. American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section XI, "Rules for In-Service Inspection of Nuclear Power Plant Components"

B 3.6 CONTAINMENT SYSTEMS

B 3.6.7 Hydrogen Recombiners

BASES

BACKGROUND

The function of the hydrogen recombiners is to eliminate the potential breach of containment due to a hydrogen-oxygen reaction. Per References 1 and 2 hydrogen recombiners are required to reduce the hydrogen concentration in the Containment Structure following a LOCA. The recombiners accomplish this by recombining hydrogen and oxygen to form water vapor. The vapor remains in the Containment Structure, thus eliminating any discharge to the environment. The hydrogen recombiners are manually initiated since flammability limits would not be reached until several days after a DBA.

Two 100% capacity independent hydrogen recombiners are provided. Each consists of controls located in the Control Room, a power supply, and a recombiner located in the Containment Structure. The recombiners have no moving parts. Recombination is accomplished by heating a hydrogen air mixture above 1150°F. The resulting water vapor and discharge gases are cooled prior to discharge from the recombiner. Air flows through the unit at 100 cfm with natural circulation in the unit providing the motive force. A single recombiner is capable of maintaining the hydrogen concentration in the Containment Structure below the 4.0 volume percent flammability limit. Two recombiners are provided to meet the requirement for redundancy and independence. Each recombiner is powered from a separate ESF bus and is provided with a separate power panel and control panel.

APPLICABLE SAFETY ANALYSES

The hydrogen recombiners provide for controlling the bulk hydrogen concentration in the Containment Structure to less than the lower flammable concentration of 4.0 volume percent following a DBA. This control would prevent a containment-wide hydrogen burn, thus ensuring the pressure and temperature assumed in the analysis are not exceeded and minimizing damage to safety-related equipment located in the Containment Structure. The limiting DBA relative to hydrogen generation is a LOCA.

BASES

Hydrogen may accumulate within the Containment Structure following a LOCA as a result of:

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

b. Radiolytic decomposition of water in the RCS and the containment sump;

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

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

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

The hydrogen recombiners satisfy 10 CFR 50.36(c)(2)(ii), Criterion 3.

LCO

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

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

APPLICABILITY

In MODEs 1 and 2, two hydrogen recombiners are required to control the post-LOCA hydrogen concentration within the Containment Structure below its flammability limit of 4.0 volume percent, assuming a worst case single failure.

In MODEs 3 and 4, both the hydrogen production rate and the total hydrogen produced after a LOCA would be less than that calculated for the MODE 1 LOCA. Also, because of the limited time in these MODEs, the probability of an accident requiring the hydrogen recombiners is low. Therefore, the hydrogen recombiners are not required in MODEs 3 or 4.

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

ACTIONS

A.1

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

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

B.1 and B.2

With two hydrogen recombiners inoperable, the ability to perform the hydrogen control function via alternate capabilities must be verified by administrative means within one hour. The alternate hydrogen control capabilities are provided by the containment vents. The one hour Completion Time allows a reasonable period of time to verify that a loss of hydrogen control function does not exist. In addition, the alternate hydrogen control system capability must be verified every 12 hours thereafter, to ensure its continued availability. Both the initial verification and all subsequent verifications may be performed as an

BASES

administrative check, by examining logs or other information to determine the availability of the alternate hydrogen control system. It does not mean to perform the SRs needed to demonstrate OPERABILITY of the alternate hydrogen control system. If the ability to perform the hydrogen control function is maintained, continued operation is permitted with two hydrogen recombiners inoperable for up to seven days. Seven days is a reasonable time to allow two hydrogen recombiners to be inoperable, because the hydrogen control function is maintained, and because of the low probability of the occurrence of a LOCA that would generate hydrogen in amounts capable of exceeding the flammability limit.

C.1

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

SURVEILLANCE REQUIREMENTS

SR 3.6.7.1

Performance of a system functional test for each hydrogen recombiner ensures that the recombiners are operational and can attain and sustain the temperature necessary for hydrogen recombination. In particular, this SR requires verification that the minimum heater sheath temperature increases to ≥ 1200°F in ≤ 5 hours and is maintained for ≥ 4 hours. Operating experience has shown that these components usually pass the surveillance test when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.7.2

Performance of a CHANNEL CALIBRATION for instrumentation and control circuits in each hydrogen recombiner ensures that the recombiners are operational and can attain the desired temperature necessary for hydrogen recombination. This provides assurance that the hydrogen concentration is maintained below its flammable limit during post-LOCA conditions. This ultimately ensures that either hydrogen recombiner unit is capable of controlling the expected hydrogen generation associated with zirconium-water reactions, radiolytic decomposition of water, and corrosion of metals within the Containment Structure. The 24 month interval is reasonable to ensure the instrumentation and control circuits are calibrated at an interval to accomplish the above requirements.

SR 3.6.7.3

This SR ensures that there are no physical problems that could affect recombiner operation. Since the recombiners are mechanically passive, they are not subject to mechanical failure. The only credible failures involve loss of power, blockage of the internal flow path, missile impact, etc. A visual inspection is sufficient to determine abnormal conditions (i.e., loose wiring or structural connections, deposits of foreign materials, etc.) that could cause such failures. The 24 month Frequency for this SR was developed considering that the incidence of hydrogen recombiners failing the SR in the past is low.

SR 3.6.7.4

This SR requires performance of a resistance-to-ground test for each heater phase, to ensure that there are no detectable grounds in any heater phase. This is accomplished by verifying that the resistance-to-ground for any heater phase is ≥ 10,000 ohms. This test is usually performed after SR 3.6.7.1 to verify the integrity of the heater electrical circuits. The 24 month Frequency for this SR was developed considering that the incidence of hydrogen recombiners failing the SR in the past is low.

BASES

REFERENCES	1.	10 CFR 50.44, "Standards for Combustible Gas Control Systems in Light-Water-Cooled Reactors"
	2.	UFSAR, Appendix 1C, " AEC Proposed General Design Criteria for Nuclear Power Plants," Criteria 62, 63, and 64
	3.	Regulatory Guide 1.7, Revision 1, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident," September 1976

B 3.6 CONTAINMENT SYSTEMS

B 3.6.8 Iodine Removal System (IRS)

BASES

BACKGROUND

The IRS is provided per Reference 1, Appendix 1C, Criteria 62, 63, and 64, to reduce the concentration of fission products released to the containment atmosphere following a postulated accident. The IRS would function together with the Containment Spray and Cooling Systems following a DBA to reduce the potential release of radioactive material, principally iodine, from the Containment Structure to the environment.

The IRS consists of three 50% capacity separate, independent (except for power), and redundant trains. Each train includes a moisture separator, a high efficiency particulate air filter, an activated charcoal adsorber section for removal of radioiodines, a fan, and instrumentation. The moisture separators function to reduce the moisture content of the air stream. The system initiates filtered recirculation of the containment atmosphere following receipt of a SIAS. The system design is described in Reference 1, Section 6.7.

The moisture separator is included for moisture (free water) removal from the gas stream. The moisture separator is important to the effectiveness of the charcoal adsorbers.

Three IRS trains are provided to meet the requirement for separation, independence (except for power), and redundancy. Two trains of the IRS are powered by separate ESF buses. The third IRS train is a swing train that can be aligned to take power from either ESFs bus.

APPLICABLE SAFETY ANALYSES

The DBAs that result in a release of radioactive iodine within the Containment Structure are a LOCA, a main SLB, or a CEA ejection accident. In the analysis for each of these accidents, it is assumed that adequate containment leak tightness exists at event initiation to limit potential leakage to the environment. Additionally, it is assumed that the amount of radioactive iodine release is limited by reducing the iodine concentration in the containment atmosphere.

BASES

The IRS design basis is established by the consequences of the limiting DBA, which is a LOCA. The accident analysis (Reference 1, Section 14.21) assumes that only two trains of the IRS are functional due to a single failure that disables the other train. The accident analysis accounts for the reduction in airborne radioactive iodine provided by the remaining two trains of this filtration system.

The IRS satisfies 10 CFR 50.36(c)(2)(ii), Criterion 3.

LCO

Three separate, independent (except for power), and redundant trains of the IRS are required to ensure that at least two are available, assuming a single failure coincident with a loss of offsite power.

APPLICABILITY

In MODEs 1, 2, 3, and 4, iodine is a fission product that can be released from the fuel to the reactor coolant as a result of a DBA. The DBAs that can cause a failure of the fuel cladding are a LOCA, main SLB, and CEA ejection accident. Because these accidents are considered credible accidents in MODEs 1, 2, 3, and 4, the IRS must be operable in these MODEs to ensure the reduction in iodine concentration assumed in the accident analysis.

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

ACTIONS

A.1

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

a. The availability of the OPERABLE redundant IRS train;

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

BASES

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

B.1

If two IRS trains are inoperable, one must be restored to OPERABLE status within one hour. The one hour Completion Time allows the swing train to be aligned to the appropriate bus to ensure each of the two remaining trains are powered from separate and independent buses. The one hour, also allows time to restore one train to OPERABLE status prior to initiating a plant shutdown. This is reasonable considering that a plant shutdown is a plant transient.

C.1 and C.2

If the IRS 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.

SURVEILLANCE REQUIREMENTS

SR 3.6.8.1

Initiating each IRS train from the Control Room and operating it for $\geq$ 15 minutes ensures that all trains are OPERABLE and that all associated controls are functioning properly. It also ensures that motor failure can be detected for corrective action. The 31 day Frequency was developed considering the known reliability of fan motors and controls, the two train redundancy available, and the iodine removal capability of the Containment Spray System independent of the IRS.

SR 3.6.8.2

This SR verifies that the required IRS filter testing is performed in accordance with the Ventilation Filter Testing Program. The IRS filter tests are in accordance with portions of Reference 2. The Ventilation Filter Testing

BASES

Program includes testing high efficiency particulate air 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 Ventilation Filter Testing Program.

SR 3.6.8.3

The automatic startup test verifies that both trains of equipment start upon receipt of an actual or simulated test signal (Engineered Safety Feature Actuation System). The 24 month Frequency is based on the need to perform this surveillance test under the conditions that apply during a plant outage and the potential for an unplanned transient if the surveillance test were performed with the reactor at power. Operating experience has shown that these components usually pass the surveillance test when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint. Furthermore, the Frequency was developed considering that the system equipment OPERABILITY is demonstrated on a 31 day Frequency by SR 3.6.8.1.

REFERENCES

1. UFSAR
2. Regulatory Guide 1.52, Revision 2, "Design, Testing, and Maintenance Criteria for Postaccident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," March 1978

B 3.7 PLANT SYSTEMS

B 3.7.1 Main Steam Safety Valves (MSSVs)

BASES

BACKGROUND

The primary purpose of the MSSVs is to provide overpressure protection for the secondary system. The MSSVs also provide protection against overpressurizing the reactor coolant pressure boundary by providing a heat sink for the removal of energy from the Reactor Coolant System (RCS) if the preferred heat sink, provided by the condenser and Circulating Water System, is not available.

Eight MSSVs are located on each main steam header, outside the Containment Structure, upstream of the main steam isolation valves (MSIVs), as described in Reference 1, Chapter 10. The MSSV rated capacity passes the full steam flow at 102% RATED THERMAL POWER (100% + 2% for instrument error) with the valves full open. This meets the requirements of Reference 2, Section III, Article NC-7000, Class 2 Components. The MSSV design includes staggered setpoints, according to Table 3.7.1-1 in the accompanying Limiting Condition for Operation (LCO), so that only the number of valves needed will actuate. Staggered setpoints reduce the potential for valve chattering, because of insufficient steam pressure to fully open all valves, following a turbine reactor trip. The MSSVs have "R" size orifices.

APPLICABLE SAFETY ANALYSES

The design basis for the MSSVs comes from Reference 2, Section III, Article NC-7000, Class 2 Components; their purpose is to limit secondary system pressure to ≤ 110% of design pressure when passing 100% of design steam flow. This design basis is sufficient to cope with any anticipated operational occurrence or accident considered Reference 1, Chapter 14.

The events that challenge the MSSV relieving capacity, and thus RCS pressure, are those characterized as decreased heat removal events, and are presented in Reference 1, Section 14.5. Of these, the full power loss of load event is the limiting anticipated operational occurrence. A loss of load isolates the turbine and condenser, and terminates normal feedwater flow to the steam generators. Before delivery of auxiliary feedwater (AFW) to the steam

BASES

generators, RCS pressure reaches peak pressure. The peak pressure is < 110% of the design pressure of 2500 psig, but high enough to actuate the pressurizer safety valves.

The MSSVs satisfy 10 CFR 50.36(c)(2)(ii), Criterion 3.

LCO

This LCO requires all MSSVs to be OPERABLE in compliance with Reference 2, Section III, Article NC-7000, Class 2 Components, even though this is not a requirement of the Design Basis Accident (DBA) analysis. This is because operation with less than the full number of MSSVs requires limitations on allowable THERMAL POWER (to meet Reference 2, Section III, Article NC-7000, Class 2 Components requirements), and adjustment to the Reactor Protective System trip setpoints to meet the transient analysis limits. These limitations are according to those shown in Table 3.7.1-1, Required Action A.2, and Required Action A.3 in the accompanying LCO.

The OPERABILITY of the MSSVs is defined as the ability to open within the setpoint tolerances, relieve steam generator overpressure, and reseat when pressure has been reduced. The OPERABILITY of the MSSVs is determined by periodic surveillance testing in accordance with the Inservice Testing Program. An MSSV is considered inoperable if it fails to open upon demand.

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

A Note is added to Table 3.7.1-2, stating that lift settings for a given steam line are also acceptable, if any two valves lift between 935 and 995 psig, any two other valves lift between 935 and 1035 psig, and the four remaining valves lift between 935 and 1050** psig. Thus, the MSSVs still perform that design basis function properly.

This LCO provides assurance that the MSSVs will perform their designed safety function to mitigate the consequences

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of accidents that could result in a challenge to the reactor coolant pressure boundary.

** The four remaining valves lift between 935 and 1065 psig for Unit 2 only, through Cycle 12.

APPLICABILITY

In MODEs 1, 2, and 3, a minimum of five MSSVs per steam generator are required to be OPERABLE, according to Table 3.7.1-1 in the accompanying LCO, which is limiting and bounds all lower MODEs.

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

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

ACTIONS

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

A.1 and A.2

An alternative to restoring the inoperable MSSV(s) to OPERABLE status is to reduce power so that the available MSSV relieving capacity meets Code requirements for the power level. The number of inoperable MSSVs will determine the necessary level of reduction in secondary system steam flow and THERMAL POWER required by the reduced reactor trip settings of the power level-high channels. The reactor trip setpoint reductions are derived on the following basis:

$$SP = \frac{(X) - (Y)(V)}{X} \times 106.5$$

where:

SP = reduced reactor trip setpoint in percent of RATED THERMAL POWER
V = maximum number of inoperable safety valves per steam line
106.5 = Power Level-High Trip Setpoint
X = Total relieving capacity of all safety valves per steam line in lbs/hour

Y = Maximum relieving capacity of any one safety valve in lbs/hour

Nuclear Regulatory Commission Information Notice 94-60 states that the linear relationship is not always valid; however, the setpoints in Table 3.7.1-1 have been verified by transient analyses.

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

The four hour Completion Time for Required Action A.2 is consistent with A.1. An additional eight hours is allowed to reduce the setpoints in recognition of the difficulty of resetting all channels of this trip function within a period of eight hours. The Completion Time of 12 hours for Required Action A.3 is based on operating experience in resetting all channels of a protective function and on the low probability of the occurrence of a transient, that could result in steam generator overpressure during this period.

B.1 and B.2

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

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1

This Surveillance Requirement (SR) verifies the OPERABILITY of the MSSVs by the verification of each MSSV lift setpoints in accordance with the Inservice Testing Program. Reference 2, Section XI, Article IWV-3500, requires that safety and relief valve tests be performed in accordance

BASES

with Reference 3. According to Reference 3, the following tests are required for MSSVs:

a. Visual examination;

b. Seat tightness determination;

c. Setpoint pressure determination (lift setting);

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

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

The ANSI/American Society of Mechanical Engineers (ASME) Standard requires that all valves be tested every five years, and a minimum of 20% of the valves be tested every 24 months. The ASME Code specifies the activities, as found lift acceptance range, and frequencies necessary to satisfy the requirements. Table 3.7.1-2 defines the lift setting range for each MSSV for OPERABILITY; however, the valves are reset to ± 1% during the surveillance test to allow for drift.

This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. This is to allow testing of the MSSVs at hot conditions. The MSSVs may be either bench tested or tested in situ at hot conditions, using an assist device to simulate lift pressure. If the MSSVs are not tested at hot conditions, the lift setting pressure shall be corrected to ambient conditions of the valve at operating temperature and pressure.

REFERENCES

1. Updated Final Safety Analysis Report (UFSAR)

2. ASME, Boiler and Pressure Vessel Code

3. ANSI/ASME OM-1-1987, Code for the Operation and Maintenance of Nuclear Power Plants, 1987

B 3.7 PLANT SYSTEMS

B 3.7.2 Main Steam Isolation Valves (MSIVs)

BASES

BACKGROUND

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

One MSIV is located in each main steam line outside, but close to, the Containment Structure. The MSIVs are downstream from the MSSVs, atmospheric dump valves (ADVs), and AFW pump turbine steam supplies to prevent their being isolated from the steam generators by MSIV closure. Closing the MSIVs isolates each steam generator from the other, and isolates the turbine, Steam Bypass System, and other auxiliary steam supplies from the steam generators.

The MSIVs close on a steam generator isolation signal generated by low steam generator pressure or on a containment spray actuation signal (CSAS) generated by high containment pressure. The MSIVs fail closed on loss of control or actuation power. The steam generator isolation signal also actuates the main feedwater isolation valves (MFIVs) to close. The MSIVs may also be actuated manually.

A description of the MSIVs is found in Reference 1, Section 10.1.

APPLICABLE SAFETY ANALYSES

The design basis of the MSIVs is established by the containment analysis for the large steam line break (SLB) inside the Containment Structure, as discussed in Reference 1, Section 14.20. It is also influenced by the accident analysis of the SLB events presented in Reference 1, Section 14.14. The design precludes the blowdown of more than one steam generator, assuming a single active component failure (e.g., the failure of one MSIV to close on demand).

The limiting case for main SLB Containment Structure response is hot full power, no loss of offsite power, and failure of a condensate booster pump to trip. This case results in continued feeding of the affected steam generator and maximizes the energy release into the Containment

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Structure. This case does not assume failure of an MSIV; however, an important assumption is both MSIVs are OPERABLE. This prevents blowdown of both steam generators assuming failure of an MSIV to close.

The accident analysis compares several different SLB events against different acceptance criteria. The large SLB outside the Containment Structure upstream of the MSIV is the limiting SLB for offsite dose, although a break in this short section of main steam header has a very low probability. The large SLB inside the Containment Structure at hot full power is the limiting case for a post-trip return to power. The analysis includes scenarios with offsite power available and with a loss of offsite power following turbine trip.

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

a. An HELB inside the Containment Structure. In order to maximize the mass and energy release into the Containment Structure, the analysis assumes steam is discharged into the Containment Structure from both steam generators until closure of the MSIV occurs. After MSIV closure, steam is discharged into the Containment Structure only from the affected steam generator.

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

c. A break downstream of the MSIVs. This type of break will be isolated by the closure of the MSIVs. Events such as increased steam flow through the turbine or the steam bypass valves (e.g., excess load event) will also terminate on closure of the MSIVs.

d. A steam generator tube rupture. For this scenario, closure of the MSIV isolates the affected steam generator from the intact steam generator and minimizes radiological releases. The operator is then required to maintain the pressure of the steam generator with the ruptured tube below the MSSV setpoints, a necessary step toward isolating the flow through the rupture.

e. The MSIVs are also utilized during other events such as a feedwater line break. These events are less limiting so far as MSIV OPERABILITY is concerned.

The MSIVs satisfy 10 CFR 50.36(c)(2)(ii), Criterion 3.

LCO

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

This LCO provides assurance that the MSIVs will perform their design safety function to mitigate the consequences of accidents as described in Reference 1, Chapter 14.

APPLICABILITY

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

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

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

ACTIONS

<u>A.1</u>

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

B.1

If the MSIV cannot be restored to OPERABLE status within eight 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 six hours and Condition C would be entered. The Completion Time is reasonable, based on operating experience, to reach MODE 2, and close the MSIVs in an orderly manner and without challenging unit systems.

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 eight hour Completion Time is consistent with that allowed in Condition A.

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

D.1 and D.2

If the MSIVs cannot be restored to OPERABLE status, or closed, within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating

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experience, to reach the required unit conditions from MODE 2 conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.2.1

This SR verifies that the closure time of each MSIV is < 5.2 seconds. The MSIV closure time is assumed in the accident and containment analyses.

The Frequency for this SR is in accordance with the Inservice Testing Program. The MSIVs are tested during each refueling outage in accordance with Reference 2, and sometimes during other cold shutdown periods. The Frequency demonstrates the valve closure time at least once per refueling cycle. Operating experience has shown that these components usually pass the SR when performed. Therefore, the Frequency is acceptable from a reliability standpoint.

REFERENCES

1. UFSAR
2. ASME, Boiler and Pressure Vessel Code, Section XI, 1989, "Rules for In-Service Inspection of Nuclear Power Plant Components," and ASME Operation and Maintenance Code Part 10, 1987, with 1988 Addenda

B 3.7 PLANT SYSTEMS

B 3.7.3 Auxiliary Feedwater (AFW) System

BASES

BACKGROUND

The AFW System automatically supplies feedwater to the steam generators to remove decay heat from the RCS upon the loss of normal feedwater supply. The AFW pumps take suction through a common suction line from the condensate storage tank (CST) (LCO 3.7.4) and pump to the steam generator secondary side via separate and independent connections, to the AFW header outside the Containment Structure. 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 MSSVs (LCO 3.7.1) or ADVs. If the main condenser is available, steam may be released via the steam bypass valves and the resulting excess water inventory in the hotwell is moved to the backup water supply.

The AFW System consists of, one motor-driven AFW pump and two steam turbine-driven pumps configured into two trains. The motor-driven pump provides 100% of AFW flow capacity; each turbine-driven pump can provide 100% of the required capacity to the steam generators as assumed in the accident analysis, but only one turbine-driven pump is lined up to auto start. The other turbine-driven pump is placed in standby and requires a manual start, when it is needed. The pumps are equipped with a common recirculation line to prevent pump operation against a closed system. The motor-driven AFW pump is powered from an independent Class 1E power supply, and feeds both steam generators.

One pump at full flow is sufficient to remove decay heat and cool the unit to Shutdown Cooling (SDC) System entry conditions.

The steam turbine-driven AFW pumps receive steam from either main steam header upstream of the MSIV. Each of the steam feed lines will supply 100% of the requirements of the turbine-driven AFW pump. The turbine-driven AFW pump supplies a common header capable of feeding both steam generators, with air-operated valves (with controllers powered by AC vital buses) actuated to the appropriate steam

BASES

generator by the Auxiliary Feedwater Actuation System (AFAS).

The AFW System may also supply feedwater to the steam generators during normal unit startup, shutdown, and hot standby conditions although the normal supply is main feedwater (MFW).

The AFW System is designed to supply sufficient water to the steam generator(s) to remove decay heat with steam generator pressure at the setpoint of the MSSVs. Subsequently, the AFW System supplies sufficient water to cool the unit to SDC entry conditions, and steam is released through the ADVs.

The AFW System actuates automatically on low steam generator level by the AFAS, as described in LCO 3.3.4. The AFAS logic is designed to feed either or both steam generators with low levels, but will isolate the AFW System from a steam generator having a significantly lower steam pressure than the other steam generator. The AFAS automatically actuates one AFW turbine-driven pump and associated air-operated valves (with controllers powered by AC vital buses) when required, to ensure an adequate feedwater supply to the steam generators. Air-operated valves with controllers powered by AC vital busses are provided for each AFW line to control the AFW flow to each steam generator.

The AFW System is discussed in Reference 1.

APPLICABLE SAFETY ANALYSES

The AFW System mitigates the consequences of any event with a loss of normal feedwater.

The design basis of the AFW System is to supply water to the steam generator to remove decay heat and other residual heat, by delivering at least the minimum required flow rate to the steam generators at pressures corresponding to the lowest MSSV set pressure plus 3%.

The limiting DBAs and transients for the AFW System are as follows:

a. Main SLB; and

b. Loss of normal feedwater.

The AFW System satisfies 10 CFR 50.36(c)(2)(ii), Criterion 3.

LCO

This LCO requires that two AFW trains be OPERABLE to ensure that the AFW System will perform its design safety function. Three AFW pumps are installed, consisting of one motor-driven and two non-condensing steam turbine-driven pumps. For a shutdown, only one pump is required to be operating, the others are in standby. Upon automatic initiation of AFW, one motor-driven and one turbine-driven pump automatically start.

The AFW System is considered to be OPERABLE when the components and flow paths required to provide AFW flow to the steam generators are OPERABLE. This requires that the motor-driven AFW pump be OPERABLE and capable of supplying AFW flow to both steam generators. The turbine-driven AFW pumps shall be OPERABLE with redundant steam supplies from each of the two main steam lines upstream of the MSIVs and capable of supplying AFW flow to both of the two steam generators. The piping, valves, instrumentation, and controls in the required flow paths shall also be OPERABLE.

The LCO is modified by a Note that allows AFW trains required for Operability to be taken out-of-service for the performance of periodic testing provided that; a dedicated operator(s) is stationed at the local control station(s) with direct communication to the Control Room, and upon completion of the testing the trains are returned to proper status and verified in proper status by independent operator checks. A train consists of one pump and the piping, valves, and controls in the direct flow path. Periodic tests include those tests that are performed in a controlled manner similar to surveillance tests, but not necessarily on the established surveillance test schedule, such as post-maintenance tests. Examples of periodic testing include closing the manual discharge valve for pump total dynamic head testing and logic testing. This Note is necessary because of the AFW pump configuration.

APPLICABILITY

In MODEs 1, 2, and 3, the AFW System is required to be OPERABLE and to function in the event that the MFW is lost.

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In addition, the AFW System is required to supply enough makeup water to replace steam generator secondary inventory and maintain the RCS in MODE 3.

In MODE 4, the AFW System is not required, however, it may be used for heat removal via the steam generator although the preferred method is MFW.

In MODEs 5 and 6, the steam generators are not normally used for decay heat removal, and the AFW System is not required.

ACTIONS

A.1 and A.2

With one of the required steam-driven AFW pumps inoperable, action must be taken to align the remaining OPERABLE steam-driven pump to automatic initiating status. This Required Action ensures that a steam-driven AFW pump is available to automatically start, if required. If the OPERABLE AFW pump is properly aligned, the inoperable steam-driven AFW pump must be restored to OPERABLE status (and placed in either standby or automatic initiating status, depending upon whether the other steam-driven AFW pump is in standby or automatic initiating status) within seven days. The 72 hour and seven day Completion Times are reasonable, based on the redundant capabilities afforded by the AFW System, the time needed for repairs, and the low probability of a DBA event occurring during this period. Two AFW pumps and flow paths remain to supply feedwater to the steam generators. The second Completion Time for Required Action A.2 establishes a limit on the maximum time allowed for any combination of Conditions to be inoperable during any continuous failure to meet this LCO.

The 10 day Completion Time provides a limitation time allowed in this specified Condition after discovery of failure to meet the LCO. The AND connector between seven days and ten days dictates that both Completion Times apply simultaneously, and the more restrictive must be met.

B.1 and B.2

With the motor-driven AFW pump inoperable, action must be taken to align the standby steam-driven pump to automatic initiating status. This Required Action ensures that another AFW pump is available to automatically start, if required. If the standby steam-driven pump is properly aligned, the inoperable motor-driven AFW pump must be restored to OPERABLE status within seven days. The 72-hour and seven day, Completion Times are reasonable, based on the redundant capabilities afforded by the AFW System, the time needed for repairs, and the low probability of a DBA event occurring during this period. Two AFW pumps and one flow path remain to supply feedwater to the steam generators. The second Completion Time for Required Action B.2 establishes a limit on the maximum time allowed for any combination of Conditions to be inoperable during any continuous failure to meet this LCO.

The 10 day Completion Time provides a limitation time allowed in this specified Condition after discovery of failure to meet the LCO. The <u>AND</u> connector between seven days and ten days dictates that both Completion Times apply simultaneously, and more restrictive must be met.

<u>C.1, C.2, C.3, and C.4</u>

With two AFW pumps inoperable, action must be taken to align the remaining OPERABLE pump to automatic initiating status and to verify the other units motor-driven AFW pump is OPERABLE, along with an OPERABLE cross-tie valve, within one hour. If these Required Actions are completed within the Completion Time, one AFW pump must be restored to OPERABLE status within 72 hours. Verifying the other unit's motor-driven AFW pump is OPERABLE provides an additional level of assurance that AFW will be available if needed, because the other unit's AFW can be cross-connected if necessary. The cross-tie valve to the opposite unit is administratively verified OPERABLE by confirming that SR 3.7.3.2 has been performed within the specified Frequency. These one hour Completion Times are reasonable based on the low probability of a DBA occurring during the first hour and the need for AFW during the first hour. The 72 hour completion time to restore one AFW pump to OPERABLE status takes into account the cross-connected capability between units and the unlikelihood of an event occurring in the 72 hour period.

D.1

With one of the required AFW trains inoperable for reasons other than Condition A, B, or C (e.g., flowpath or steam supply valve), 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 pumps. The 72 hour Completion Time is reasonable, based on the redundant capabilities afforded by the AFW System, the time needed for repairs, and the low probability of a DBA event occurring during this period. One AFW train remains to supply feedwater to the steam generators. The second Completion Time for Required Action D.1 establishes a limit on the maximum time allowed for any combination of Conditions to be inoperable during any continuous failure to meet this LCO.

The ten day Completion Time provides a limited time allowed in this specified Condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which Conditions A and B are entered concurrently. The AND connector between 72 hours and ten days dictates that both Completion Times apply simultaneously, and more restrictive must be met.

E.1 and E.2

When the Required Action and associated Completion Time of Condition A, B, C, or D cannot be met 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.

F.1

Required Action F.1 is modified by a Note indicating that all required MODE changes or power reductions are suspended until one AFW train is restored to OPERABLE status.

BASES

With two AFW trains inoperable in MODEs 1, 2, and 3, the unit may be in a seriously degraded condition with only non-safety-related means for conducting a cooldown. In such a condition, the unit should not be perturbed by any action, including a power change, that might result in a trip. However, a power change is not precluded if it is determined to be the most prudent action. The seriousness of this condition requires that action be started immediately to restore one AFW train to OPERABLE status. While other plant conditions may require entry into LCO 3.0.3, the ACTIONS required by LCO 3.0.3 do not have to be completed because they could force the unit into a less safe condition.

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1

Verifying the correct alignment for manual, power-operated, and automatic valves in the AFW water and steam supply flow paths, provides assurance that the proper flow paths exist for AFW operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves are verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This SR does not require any testing or valve manipulations; rather, it involves verification that those valves capable of potentially being mispositioned are in the correct position.

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

SR 3.7.3.2

Cycling each testable, remote-operated valve that is not in its operating position, provides assurance that the valves will perform as required. Operating position is the position that the valve is in during normal plant operation. This is accomplished by cycling each valve at least one cycle. This SR ensures that valves required to function during certain scenarios, will be capable of being properly positioned. The Frequency is based on engineering judgment that when cycled in accordance with the Inservice Testing

Program, these valves can be placed in the desired position when required.

SR 3.7.3.3

Verifying that each AFW pump's developed head at the flow test point is greater than or equal to the required developed head (≥ 2800 ft for the steam-driven pump and ≥ 3100 ft for the motor-driven pump), ensures that AFW pump performance has not degraded during the cycle. Flow and differential head are normal tests of pump performance required by Reference 2. Because it is undesirable to introduce cold AFW into the steam generators while they are operating, this testing is performed on recirculation flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. Performance of inservice testing, discussed in Reference 2, at three 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 an insufficient steam pressure to perform the test.

SR 3.7.3.4

This SR ensures that AFW can be delivered to the appropriate steam generator, in the event of any accident or transient that generates an AFAS signal, by demonstrating that each automatic valve in the flow path actuates to its correct position on an actual or simulated actuation signal (verification of flow-modulating characteristics is not required). This SR is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 24 month Frequency is based on the need to perform this surveillance test under the conditions that apply during a unit outage and the potential for an unplanned transient if the surveillance test were performed with the reactor at power. The 24 month Frequency is acceptable, based on the design reliability and operating experience of the equipment.

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This SR is modified by a Note indicating that the SR should be deferred until suitable test conditions have been established.

SR 3.7.3.5

This SR ensures that the AFW pumps will start in the event of any accident or transient that generates an AFAS signal by demonstrating that each AFW pump starts automatically on an actual or simulated actuation signal. The 24 month Frequency is acceptable, based on the design reliability and operating experience of the equipment.

This SR is modified by a Note. The Note indicates that the SR be deferred until suitable test conditions are established.

SR 3.7.3.6

This SR ensures that the AFW system is capable of providing a minimum nominal flow to each flow leg. This ensures that the minimum required flow is capable of feeding each flow leg. The test may be performed on one flow leg at a time. The SR is modified by a Note which states, the SR is not required to be performed for the AFW train with the turbine-driven AFW pump until 24 hours after reaching 800 psig in the steam generators. The Note ensures that proper test conditions exist prior to performing the test using the turbine-driven AFW pumps. The 24 month Frequency coincides with performing the test during refueling outages.

SR 3.7.3.7

This SR ensures that the AFW System is properly aligned by verifying the flow path to each steam generator prior to entering MODE 2 operation, after 30 days in MODEs 5 or 6. OPERABILITY of AFW flow paths must be verified before sufficient core heat is generated that would require the operation of the AFW System during a subsequent shutdown. The Frequency is reasonable, based on engineering judgment, and other administrative controls to ensure that flow paths remain OPERABLE. To further ensure AFW System alignment, the OPERABILITY of the flow paths is verified following extended outages to determine that no misalignment of valves has occurred. This SR ensures that the flow path from the

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CST to the steam generators is properly aligned. Minimum nominal flow to each flow leg is ensured by performance of SR 3.7.3.6.

REFERENCES

1. UFSAR, Section 10.3
2. ASME, Boiler and Pressure Vessel Code, Section XI, Inservice Inspection, Article IWV-3400

B 3.7 PLANT SYSTEMS

B 3.7.4 Condensate Storage Tank (CST)

BASES

BACKGROUND

The CST provides a safety grade source of water to the steam generators for removing decay and sensible heat from the RCS. The CST provides a passive flow of water, by gravity, to the AFW System (LCO 3.7.3). The steam produced is released to the atmosphere by the MSSVs or the atmospheric dump valves. The AFW pumps operate with a continuous recirculation to the CST.

The component required by this Specification is CST No. 12.

When the MSIVs are open, the preferred means of heat removal is to discharge steam to the condenser by the non-safety grade path of the turbine bypass valves. The condensed steam is returned to the backup water supply (CST No. 11 and CST No. 21) by the condensate pump. This has the advantage of conserving condensate while minimizing releases to the environment.

Because the CST is a principal component in removing residual heat from the RCS, it is designed to withstand earthquakes and other natural phenomena. The CST is designed to Seismic Category I requirements to ensure availability of the feedwater supply. Feedwater is also available from an alternate source.

There is one CST (CST No. 12) shared by Units 1 and 2. A description of the CST is found in Reference 1, Sections 6.3.5.1 and 10.3.2.

APPLICABLE SAFETY ANALYSES

The CST provides cooling water to remove decay heat and to cool down the unit following all events in the accident analysis, discussed in Reference 1, Chapter 14. For anticipated operational occurrences and accidents which do not affect the OPERABILITY of the steam generators, the analysis assumption is generally six hours at MODE 3, steaming through the MSSVs followed by a cooldown to SDC entry conditions at the design cooldown rate.

The limiting event for the condensate volume is the large feedwater line break with a coincident loss of offsite

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power. Single failures that also affect this event include the following:

a. The failure of the diesel generator powering the motor-driven AFW pump to the unaffected steam generator (requiring additional steam to drive the remaining AFW pump turbine); and

b. The failure of the steam driven train (requiring a longer time for cooldown using only one motor-driven AFW pump).

These are not usually the limiting failures in terms of consequences for these events.

The CST satisfies 10 CFR 50.36(c)(2)(ii), Criteria 2 and 3.

LCO

To satisfy accident analysis assumptions, the CST (i.e., CST No. 12) must contain sufficient cooling water for both units to ensure that sufficient water is available to maintain the RCS at MODE 3 for six hours following a reactor trip from 102% RATED THERMAL POWER, assuming a coincident loss of offsite power and the most adverse single failure. In doing this, it must retain sufficient water to ensure adequate net positive suction head for the AFW pumps during the cooldown while in MODE 3, as well as to account for any losses from the steam-driven AFW pump turbine, or before isolating AFW to a broken line.

The CST usable volume required is ≥ 150,000 gallons per unit in the MODE of Applicability, which is equal to 300,000 gallons. The 300,000 gallons of water is enough to provide for decay heat removal and cooldown of both units. By adjusting the feedwater flow to the permissible cooldown rate, decay heat removal and cooldown of both units can be accomplished in six hours. The 300,000 gallons are also adequate to maintain the RCS in MODE 3 for six hours with steam discharge to atmosphere with concurrent and total loss of offsite power, or to remove decay heat from both units for more than ten hours after initiation of cooldown and still maintain normal no-load water level in the steam generators. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

OPERABILITY of the CST is determined by maintaining the tank volume at or above the minimum required volume.

APPLICABILITY

In MODEs 1, 2, and 3, the CST is required to be OPERABLE.

In MODEs 4, 5 and 6, the CST is not required because the AFW System is not required.

ACTIONS

A.1 and A.2

If the CST is not OPERABLE, the OPERABILITY of the backup water supply (CST No. 11 for Unit 1 and CST No. 21 for Unit 2) must 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 manual valves in the flow paths from the backup supply to the AFW pumps are open, and availability of the required volume of water (150,000 gallons) in the backup supply. The CST must be returned to OPERABLE status within seven days, as the backup supply may be performing this function in addition to its normal functions. The four hour Completion Time is reasonable, based on operating experience, to verify the OPERABILITY of the backup water supply. Additionally, verifying the backup water supply every 12 hours is adequate to ensure the backup water supply continues to be available. The seven day Completion Time is reasonable, based on an OPERABLE backup water supply being available, and the low probability of an event requiring the use of the water from the CST occurring during this period.

If the CST volume is less than 300,000 gallons and greater than 150,000 gallons and both units are in the MODE of Applicability, only one unit must enter this condition provided the unit aligns to the OPERABLE backup water supply (CST No. 11 or CST No. 21).

B.1 and B.2

If the CST cannot be restored to OPERABLE status within the associated Completion Time, the affected unit(s) must be placed in a MODE in which the LCO does not apply. To

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achieve this status, the unit(s) 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 plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.4.1

This SR verifies that the CST contains the required usable volume of cooling water. (This volume ≥ 150,000 gallons per unit in the MODE of Applicability.) The 12 hour Frequency is based on operating experience, and the need for operator awareness of unit evolutions that may affect the CST inventory between checks. The 12 hour Frequency is considered adequate in view of other indications in the Control Room, including alarms, to alert the operator to abnormal CST volume deviations.

Although the volume in the CST for each unit is required to be 150,000 gallons, the total combined volume for both units is 300,000 gallons.

REFERENCES

1. UFSAR

B 3.7 PLANT SYSTEMS

B 3.7.5 Component Cooling (CC) System

BASES

BACKGROUND

The CC System provides a heat sink for the removal of process and operating heat from safety-related components during a DBA or transient. During normal operation, the CC System also provides this function for various nonessential components. The CC System serves as a barrier to the release of radioactive byproducts between potentially radioactive systems and the Saltwater (SW) System, and thus to the environment.

The CC System consists of two redundant loops that are always cross-connected. A loop consists of one of three redundant pumps, one of two redundant CC heat exchangers along with a common head tank, associated valves, piping, instrumentation, and controls. The third pump, which is an installed spare, can be powered from either electrical train. The redundant cooling capacity of this system, assuming single active failure, is consistent with the assumptions made in the accident analysis.

During normal operation one loop typically provides cooling water with a maximum CC heat exchanger outlet temperature of 95°F with the redundant loop components in standby. If needed, the redundant loop components can be aligned to supplement the in service loop. While operating on SDC with one loop, the CC heat exchanger outlet temperature may rise to a maximum temperature of 120°F.

Following a loss of coolant accident (LOCA) while recirculating water from the containment sump, the CC heat exchangers are designed to provide a maximum outlet cooling water temperature of 120°F provided one of the following component alignment combinations is met (assumes CC to containment and evaporators is isolated): a) 1 CC pump, 2 CC heat exchangers, and 2 SDC heat exchangers; b) 1 CC pumps, 1 CC heat exchanger, 1 SDC heat exchangers; and c) 2 CC pumps, 2 CC heat exchangers, 1 SDC heat exchangers. In the event of a passive failure of the common portions of the CC loop during a LOCA, the entire system would be lost. The unit can still be maintained in a safe condition since the containment coolers would be utilized in lieu of the spray pumps/shutdown heat exchangers to cool the Containment Structure (Reference 1, Section 9.5.5).

Additional information on the design and operation of the system, along with a list of the components served, is presented in Reference 1, Section 9.5.2.1. The principal safety-related function of the CC System is the removal of decay heat from the reactor via the SDC System heat exchanger. This may utilize the SDC heat exchanger, during a normal or post accident cooldown and shutdown, or the Containment Spray System during the recirculation phase following a LOCA.

APPLICABLE SAFETY ANALYSES

The design basis of the CC System is for it to support a 100% capacity Containment Cooling System (containment spray, containment coolers, or a combination) removing core decay heat 36 minutes after a design basis LOCA. 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 RCS, by the safety injection pumps.

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

The CC System also functions to cool the unit from SDC entry conditions (T_{cold} < 300°F) to T_{cold} < 140°F during normal operations. The time required to cool from 300°F to 140°F is a function of the number of CC and SDC loops operating. One CC loop is sufficient to remove decay heat during subsequent operations with T_{cold} < 140°F. This assumes that a maximum inlet SW temperature occurs simultaneously with the maximum heat loads on the system.

The CC System satisfies 10 CFR 50.36(c)(2)(ii), Criterion 3.

LCO

The CC loops are redundant 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 CC loop is required to provide the minimum heat removal capability assumed in the safety analysis for

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the systems to which it supplies cooling water. To ensure this requirement is met, two CC loops must be OPERABLE. At least one CC loop will operate assuming the worst single active failure occurs coincident with the loss of offsite power. Additionally, the containment cooling function will also operate assuming the worst case passive failure post-recirculation actuation signal (RAS).

A CC loop is considered OPERABLE when the following:

a. The associated pump and common head 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 CC from other components or systems not required for safety may render those components or systems inoperable, but does not affect the OPERABILITY of the CC System.

APPLICABILITY

In MODEs 1, 2, 3, and 4, the CC System is a normally operating system that must be prepared to perform its post accident safety functions, primarily RCS heat removal by cooling the SDC heat exchanger.

In MODEs 5 and 6, the OPERABILITY requirements of the CC System are determined by the systems it supports.

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating the requirement of entry into the applicable Conditions and Required Actions of LCO 3.4.6, for SDC made inoperable by CC. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components.

With one CC loop inoperable, action must be taken to restore OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE CC loop is adequate to perform the heat removal function. The 72 hour Completion Time is based on the redundant capabilities afforded by the OPERABLE loop, and the low probability of a DBA occurring during this period.

B.1 and B.2

If the CC loop cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours.

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

SURVEILLANCE REQUIREMENTS

SR 3.7.5.1

Verifying the correct alignment for manual, power operated, and automatic valves in the CC flow path provides assurance that the proper flow paths exist for CC 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 SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of potentially being mispositioned are in their correct position.

This SR is modified by a Note indicating that the isolation of the CC components or systems may render those components inoperable but does not affect the OPERABILITY of the CC System.

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

SR 3.7.5.2

This SR verifies proper automatic operation of the CC valves on an actual or simulated safety injection actuation signal

(SIAS). The CC System is a normally operating system that cannot be fully actuated as part of routine testing during normal operation. This SR is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 24 month Frequency is based on the need to perform this surveillance test under the conditions that apply during a unit outage and the potential for an unplanned transient if the surveillance test were performed with the reactor at power. Operating experience has shown that these components usually pass the surveillance test when performed at the 24 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

SR 3.7.5.3

This SR verifies proper automatic operation of the CC pumps on an actual or simulated SIAS. The CC System is a normally operating system that cannot be fully actuated as part of routine testing during normal operation. The 24 month Frequency is based on the need to perform this surveillance test under the conditions that apply during a unit outage and the potential for an unplanned transient if the surveillance test were performed with the reactor at power. Operating experience has shown these components usually pass the surveillance test when performed at the 24 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

REFERENCES

1. UFSAR

B 3.7 PLANT SYSTEMS

B 3.7.6 Service Water (SRW) System

BASES

BACKGROUND

The SRW System provides a heat sink for the removal of process and operating heat from safety-related components during a DBA or transient. During normal operation or a normal shutdown, the SRW System also provides this function for various safety-related and non-safety-related components. The safety-related function is covered by this LCO.

The SRW System consists of two separate, 100% capacity safety-related cooling water subsystems. Each subsystem consists of a 100% capacity pump, head tank, two SRW heat exchangers, piping, valves, and instrumentation. A third pump, which is an installed spare, can be powered from either electrical train. The pumps and valves are remote manually aligned, except in the unlikely event of a LOCA. The pumps are automatically started upon receipt of a SIAS and all essential valves are aligned to their post-accident positions.

During normal operation, both subsystems are required, and are independent to the degree necessary to assure the safe operation and shutdown of the plant-assuming a single failure. During shutdown, operation of the SRW System is the same as normal operation, except that the heat loads are reduced. Additional information about the design and operation of the SRW System, along with a list of the components served, is presented in Reference 1, Section 9.5.2.2. In the event of a LOCA, the SRW System automatically realigns to isolate Turbine Building (non-safety-related) loads creating two independent and redundant safety-related subsystems. Service water flow to the spent fuel pool (SFP) cooler and the blowdown heat exchanger is automatically isolated as required for the DBA. Each SRW subsystem will supply cooling water to a diesel generator and two containment air coolers. However, the No. 11 SRW subsystem only supplies two containment air coolers since the No. 1A Diesel Generator is air cooled. Each SRW subsystem is sufficiently sized to remove the maximum amount of heat from the containment atmosphere while maintaining

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the SRW supply temperature to the diesel generator below its design limit.

APPLICABLE SAFETY ANALYSES	The design basis of the SRW System is for it to support a 100% capacity containment cooling system (containment coolers) and to remove core decay heat 36 minutes following a design basis LOCA, as discussed in Reference 1, Section 14.20. 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 RCS by the safety injection pumps. The SRW System is designed to perform its function with a single failure of any active component, assuming the loss of offsite power. The SRW System satisfies 10 CFR 50.36(c)(2)(ii), Criterion 3.
LCO	Two SRW subsystems are required to be OPERABLE to provide the required redundancy to ensure that the system functions to remove post-accident heat loads, assuming the worst single active failure occurs coincident with the loss of offsite power. Additionally, this system will also operate assuming that worst case passive failure post-RAS. An SRW subsystem is considered OPERABLE when: a. The associated pump and head tank are OPERABLE; and b. The associated piping, valves, heat exchanger, and instrumentation and controls required to perform the safety-related function are OPERABLE.
APPLICABILITY	In MODEs 1, 2, 3, and 4, the SRW System is a normally operating system, which is required to support the OPERABILITY of the equipment serviced by the SRW System and required to be OPERABLE in these MODEs. In MODEs 5 and 6, the OPERABILITY requirements of the SRW System are determined by the systems it supports.

ACTIONS

A.1 and A.2

With one SRW heat exchanger inoperable, action must be taken to restore operable status within 7 days. Isolating flow to one associated containment cooling unit will reduce the DBA heat load of the affected SRW subsystem to within the capacity of one SRW heat exchanger, thus ensuring that the SRW temperatures can be maintained within their design limits. This will allow the associated diesel generator (except for 11 SRW which does not cool a diesel generator) to remain operable. In this Condition, the other OPERABLE SRW System is adequate to perform the containment heat removal function. However, the overall reliability is reduced because a single failure in the SRW System could result in loss of SRW containment heat removal function. Required Action A.1 is modified by a Note. The Note indicates that the applicable Conditions of LCO 3.6.6 should be entered for an inoperable containment cooling train. The 7 day Completion Time is based on the redundant capabilities afforded by the OPERABLE subsystem, the Completion Time associated with an inoperable containment cooling unit (3.6.6), and the low probability of a DBA occurring during this time period.

B.1

With one SRW subsystem inoperable, action must be taken to restore OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE SRW System is adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure in the SRW System could result in loss of SRW function. Required Action B.1 is modified by a Note. The Note indicates that the applicable Conditions of LCO 3.8.1, should be entered if the inoperable SRW subsystem results in an inoperable diesel generator. The 72 hour Completion Time is based on the redundant capabilities afforded by the OPERABLE subsystem, and the low probability of a DBA occurring during this time period.

C.1 and C.2

If the SRW subsystem cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours.

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

SURVEILLANCE REQUIREMENTS

SR 3.7.6.1

Verifying the correct alignment for manual, power-operated, and automatic valves in the SRW flow path ensures that the proper flow paths exist for SRW 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 SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of potentially being mispositioned are in the correct position. This SR is modified by a Note indicating that the isolation of the SRW components or systems may render those components inoperable but does not affect the OPERABILITY of the SRW System.

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

SR 3.7.6.2

This SR verifies proper automatic operation of the SRW System valves on an actual or simulated actuation signal (SIAS or CSAS). The SRW System is a normally operating system that cannot be fully actuated as part of normal testing. This surveillance test is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 24 month Frequency is based on the need to perform this surveillance test under the conditions that apply during a unit outage, and the potential for an unplanned transient if the surveillance test were performed with the reactor at power. Operating experience has shown that these components usually pass the surveillance test when performed at the 24 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

SR 3.7.6.3

The SR verifies proper automatic operation of the SRW System pumps on an actual or simulated actuation signal (SIAS or CSAS). The SRW System is a normally operating system that cannot be fully actuated as part of the normal testing during normal operation. Operating experience has shown that these components usually pass the surveillance test when performed at the 24 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

REFERENCES

1. UFSAR

B 3.7 PLANT SYSTEMS

B 3.7.7 Saltwater (SW) System

BASES

BACKGROUND

The SW System provides a heat sink for the removal of process and operating heat from safety-related components during a DBA or transient. During normal operation or a normal shutdown, the SW System also provides this function for various safety-related and non-safety-related components. The safety-related function is covered by this LCO.

The SW System consists of two subsystems. Each subsystem contains one pump. A third pump, which is an installed spare, can be aligned to either subsystem. The safety-related function of each subsystem is to provide SW to two SRW heat exchangers, a CC heat exchanger, and an Emergency Core Cooling System (ECCS) pump room air cooler in order to transfer heat from these systems to the bay. Seal water for the non-safety-related circulating water pumps is supplied by both or either subsystems. The SW pumps provide the driving head to move SW from the intake structure, through the system and back to the circulating water discharge conduits. The system is designed such that each pump has sufficient head and capacity to provide cooling water such that 100% of the required heat load can be removed by either subsystem.

During normal operation, both subsystems in each unit are in operation with one pump running on each header and a third pump in standby. If needed, the standby pumps can be lined-up to either supply header. The SW flow through the SRW and CC heat exchangers is throttled to provide sufficient cooling to the heat exchangers, while maintaining total subsystem flow below a maximum value.

Additional information about the design and operation of the SW System, along with a list of the components served, is presented in Reference 1. During an accident, the SW System is required to remove the heat load from the SRW and ECCS pump room, and from the CC following an RAS.

BASES

APPLICABLE SAFETY ANALYSES

The most limiting event for the SW System is a LOCA. Operation of the SW System following a LOCA is separated into two phases, before the RAS and after the RAS. One subsystem can satisfy cooling requirements of both phases. After a LOCA but before an RAS, each subsystem will cool two SRW heat exchangers and an ECCS pump room air cooler (as required). There is no flow to the CC heat exchangers. When an RAS occurs, flow is initiated to the CC heat exchanger. Flow to each SRW heat exchanger is reduced while the system remains capable of providing the required flow to the ECCS pump room air coolers.

The SW System satisfies 10 CFR 50.36(c)(2)(ii), Criterion 3.

LCO

Two SW subsystems are required to be OPERABLE to provide the required redundancy to ensure that the system functions to remove post-accident heat loads, assuming the worst single active failure occurs coincident with the loss of offsite power. Additionally, this system will also operate assuming the worst case passive failure post-RAS.

An SW subsystem is considered OPERABLE when:

a. The associated pump is OPERABLE; and

b. The associated piping, valves, heat exchangers, and instrumentation and controls required to perform the safety-related function are OPERABLE.

APPLICABILITY

In MODEs 1, 2, 3, and 4, the SW System is a normally operating system, which is required to support the OPERABILITY of the equipment serviced by the SW System and required to be OPERABLE in these MODEs.

In MODEs 5 and 6, the OPERABILITY requirements of the SW System are determined by the systems it supports.

ACTIONS

A.1

With one SW subsystem inoperable, action must be taken to restore OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE SW subsystem is adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure in the SW subsystem could result in loss of SW System function. Required Action A.1 is modified by two Notes. The first Note indicates that the applicable Conditions of LCO 3.8.1 should be entered if the inoperable SW subsystem results in an inoperable emergency diesel generator. The second Note indicates that the applicable Conditions and Required Actions of LCO 3.4.6 should be entered if an inoperable SW subsystem results in an inoperable SDC. The 72 hour Completion Time is based on the redundant capabilities afforded by the OPERABLE train, and the low probability of a DBA occurring during this time period.

B.1 and B.2

If the SW subsystems cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours.

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

SURVEILLANCE REQUIREMENTS

SR 3.7.7.1

Verifying the correct alignment for manual, power-operated, and automatic valves in the SW System flow path ensures that the proper flow paths exist for SW 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 locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This surveillance test does not require any testing or valve manipulation; rather, it involves verification that those valves capable of potentially being mispositioned are in the correct position. This SR is modified by a Note indicating that the isolation of the SW System components or systems may render those components inoperable but does not affect the OPERABILITY of the SW System.

BASES

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

SR 3.7.7.2

This SR verifies proper automatic operation of the SW System valves on an actual or simulated actuation signal (SIAS). The SW System is a normally operating system that cannot be fully actuated as part of the normal testing. This surveillance test is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 24 month Frequency is based on the need to perform this surveillance test under the conditions that apply during a unit outage and the potential for an unplanned transient if the surveillance test were performed with the reactor at power. Operating experience has shown that these components usually pass the surveillance test when performed at the 24 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

SR 3.7.7.3

The SR verifies proper automatic operation of the SW System pumps on an actual or simulated actuation signal (SIAS). The SW System is a normally operating system that cannot be fully actuated as part of the normal testing during normal operation. Operating experience has shown that these components usually pass the surveillance test when performed at the 24 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

REFERENCES

1. UFSAR, Section 9.5.2.3, "Saltwater System"

B 3.7 PLANT SYSTEMS

B 3.7.8 Control Room Emergency Ventilation System (CREVS)

BASES

BACKGROUND

The CREVS provides a protected environment from which operators can control the unit following an uncontrolled release of radioactivity, chemicals, or toxic gas. The CREVS is a shared system providing protection for both Unit 1 and Unit 2.

The CREVS consists of two trains, including redundant outside air intake ducts and redundant emergency recirculation filter trains that recirculate and filter the Control Room air. The CREVS also has and shared equipment, including a shared exhaust to atmosphere duct containing redundant isolation valves, a shared exhaust to atmosphere duct from the kitchen and toilet area of the Control Room containing a single isolation valve, and common supply and return ducts in both the standby and emergency recirculation portions of the system. The shared equipment is considered to be a part of each CREVS train. Each CREVS emergency recirculation filter train consists of a prefilter, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodine), and a fan. Instrumentation which actuates the system is addressed in LCOs 3.3.4 and 3.3.8.

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 into the emergency recirculation mode of operation, closes the unfiltered outside air intake and unfiltered exhaust to atmosphere valves, and aligns the system for emergency recirculation of Control Room air through the redundant trains of HEPA and charcoal filters. The prefilters remove any large particles in the air, and any entrained water droplets present to prevent excessive loading of the HEPA filters and charcoal adsorbers. Either a control room recirculation signal (CRRS) or a SIAS initiates this filtered ventilation of the air supply to the Control Room.

The air entering the Control Room outside air intake is continuously monitored by a radiation detector. Detector output above the setpoint will cause actuation of the CREVS.

BASES

The CREVS operation in maintaining the Control Room habitable is discussed in Reference 1, Section 9.8.2.3.

The redundant emergency recirculation filter train provides the required filtration should an excessive pressure drop develop across the other filter train. Normally open double isolation valves are arranged in series so that the failure of one valve to shut will not result in a breach of isolation from the outside atmosphere, except for the exhaust from the Control Room kitchen and toilet areas. The CREVS is designed in accordance with Seismic Category I requirements.

The CREVS is designed to maintain the Control Room environment 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.

APPLICABLE SAFETY ANALYSES

The CREVS components are generally arranged in redundant safety-related ventilation trains although some equipment is shared between trains.

The CREVS provides automatic airborne radiological protection for the Control Room Operators, as demonstrated by the Control Room accident dose analyses for the most limiting design basis LOCA fission product release presented in Reference 1, Chapter 14.

The CREVS also provides manually actuated airborne radiological protection for the Control Room operations, for the design basis fuel handling accident presented in Reference 1, Chapter 14.

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 (except for one valve in the shared duct between the Control Room and the emergency recirculation filter trains).

The CREVS satisfies 10 CFR 50.36(c)(2)(ii), Criterion 3.

LCO

The CREVS is required to be OPERABLE to ensure that the Control Room is isolated and at least one emergency recirculation filter train is available, assuming a single failure. Total system failure could result in a Control Room Operator receiving a dose in excess of 5 rem whole body dose in the event of a large radioactive release.

The CREVS is considered OPERABLE when the individual components necessary to control operator exposure are OPERABLE. For MODEs 1, 2, 3, and 4, redundancy is required and CREVS is considered OPERABLE when:

a. Both supply fans are OPERABLE;

b. Both recirculation fans are OPERABLE;

c. Both fans included in the emergency recirculation filter trains are OPERABLE;

d. Both HEPA filters and charcoal adsorbers are not excessively restricting flow, and are capable of performing their filtration functions;

e. Ductwork, valves, and dampers are OPERABLE, such that air circulation can be maintained; and

f. The Control Room outside air intake can be isolated for the emergency recirculation mode of operation, assuming a single failure.

The LCO is modified by a Note which indicates that only one CREVS redundant component is required to be OPERABLE during movement of irradiated fuel assemblies, when both units are in MODEs 5 or 6, or defueled. Therefore, with both units in other than MODEs 1, 2, 3, or 4, redundancy is not required for movement of irradiated fuel assemblies and CREVS is considered OPERABLE when:

a. One supply fan is OPERABLE;

b. One recirculation fan is OPERABLE;

c. One fan included in the OPERABLE emergency recirculation filter train is OPERABLE;

d. One HEPA filter and charcoal adsorber are not excessively restricting flow, and are capable of performing their filtration functions; and

e. Associated ductwork, valves, and dampers are OPERABLE, such that air circulation can be maintained and the Control Room can be isolated for the emergency recirculation mode.

When implementing the Note (since redundancy is not required), only one of the two isolation valves in each outside air intake duct is required, and only one of the two isolation valves in the exhaust to atmosphere duct is required. However, the non-operating flow path must be capable of providing isolation of the Control Room from the outside atmosphere.

In addition, the Control Room boundary must be maintained with sufficient integrity to control operator exposure following an accident.

APPLICABILITY

In MODEs 1, 2, 3, and 4, the CREVS must be OPERABLE to limit operator exposure 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.

ACTIONS

A.1

With one or more ducts with one Control Room outside air intake isolation valve inoperable in MODEs 1, 2, 3, or 4, the OPERABLE Control Room outside air intake valve in each affected duct must be closed immediately. This places the OPERABLE Control Room outside air intake isolation valve in each affected duct in its safety function required position.

B.1

With the toilet area exhaust isolation valve inoperable, action must be taken to restore OPERABLE status within 24 hours. In this Condition, the toilet area exhaust cannot be isolated, therefore, the valve must be restored to OPERABLE status. The 24 hour period allows enough time to repair the valve while limiting the time the toilet area is open to the atmosphere. The 24 hour Completion Time is based on the low probability of a DBA occurring during this time period.

C.1

With one exhaust to atmosphere isolation valve inoperable in MODEs 1, 2, 3, or 4, action must be taken to restore OPERABLE status within seven days. In this Condition, the remaining OPERABLE exhaust to atmosphere isolation valve is adequate to isolate the Control Room. However, the overall reliability is reduced because a single failure in the OPERABLE exhaust to atmosphere isolation valve could result in loss of exhaust to atmosphere isolation valve function. The seven day Completion Time is based on the low probability of a DBA occurring during this time period, and the ability of the remaining exhaust to atmosphere isolation valve to provide the required isolation capability.

D.1

With one CREVS train inoperable for reasons other than Conditions A, B, or C in MODEs 1, 2, 3, or 4, action must be taken to restore OPERABLE status within seven days. In this Condition, the remaining OPERABLE CREVS subsystem is adequate to perform Control Room radiation protection function. However, the overall reliability is reduced because a single failure in the OPERABLE CREVS train could result in loss of CREVS function. The seven day Completion Time is based on the low probability of a DBA occurring during this time period, and the ability of the remaining train to provide the required capability.

E.1 and E.2

If the Required Actions and associated Completion Times of Conditions A, B, C, or D are not met in MODEs 1, 2, 3, or 4, the unit must be placed in a MODE that minimizes the accident risk. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

BASES

F.1

During movement of irradiated fuel assemblies, if the Required Action and associated Completion Time of Condition B is not met, or if CREVS is otherwise inoperable the movement of irradiated fuel assemblies must be immediately suspended. This places the unit in a condition that minimizes the accident risk. This does not preclude the movement of fuel assemblies to a safe position.

G.1

If both CREVS trains are inoperable for reasons other than Conditions A, B, or C, or if one or more ducts have two outside air intake isolation valves inoperable, or if two exhaust to atmosphere isolation valves are inoperable, in MODEs 1, 2, 3, or 4, the CREVS may not be capable of performing the intended function and the unit is in a condition outside the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.7.8.1

Standby systems should be checked periodically to ensure that they function properly. Since the environment and normal operating conditions on this system are not severe, testing each required CREVS filter train once every month provides an adequate check on this system.

The 31 day Frequency is based on the known reliability of the equipment, and the two filter train redundancy available.

SR 3.7.8.2

This SR verifies that the required CREVS testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The CREVS filter tests are in accordance with portions of Reference 2. 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 3.7.8.3

This SR verifies each CREVS train starts and operates on an actual or simulated actuation signal (CRRS). This test is conducted on a 24 month Frequency. This Frequency is adequate to ensure the CREVS is capable of starting and operating on an actual or simulated CRRS.

REFERENCES

1. UFSAR
2. Regulatory Guide 1.52, Revision 2, "Design, Testing, and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," March 1978

B 3.7 PLANT SYSTEMS

B 3.7.9 Control Room Emergency Temperature System (CRETS)

BASES

BACKGROUND

The CRETS provides temperature control for the Control Room following isolation of the Control Room. The CRETS is a shared system which is supported by the CREVS, since the CREVS must be operating in the emergency recirculation mode for CRETS to perform its safety function.

The CRETS consists of two independent, redundant trains that provide cooling of recirculated Control Room air. Each train consists of cooling coils, instrumentation, and controls to provide for Control Room temperature control. The CRETS is a subsystem providing air temperature control for the Control Room.

The CRETS is an emergency system, parts of which may also operate during normal unit operations in the standby mode of operation. A single train will provide the required temperature control to maintain the Control Room below 104°F. The CRETS operation to maintain the Control Room temperature is discussed in Reference 1.

APPLICABLE SAFETY ANALYSES

The design basis of the CRETS is to maintain temperature of the Control Room environment throughout 30 days of continuous occupancy.

The CRETS components are arranged in redundant safety-related trains. During emergency operation, the CRETS maintains the temperature below 104°F. A single active failure of a component of the CRETS, assuming a loss of offsite power, does not impair the ability of the system to perform its design function. Redundant detectors and controls are provided for Control Room temperature control. The CRETS is designed in accordance with Seismic Category I requirements. The CRETS is capable of removing sensible and latent heat loads from the Control Room, considering equipment heat loads and personnel occupancy requirements, to ensure equipment OPERABILITY.

The CRETS satisfies 10 CFR 50.36(c)(2)(ii), Criterion 3.

BASES

LCO

Two independent and redundant trains of the CRETS are required to be OPERABLE to ensure that at least one is available, assuming a single failure disables the other train following isolation of the Control Room. Total system failure could result in the equipment operating temperature exceeding limits in the event of an accident requiring isolation of the Control Room.

The CRETS is considered OPERABLE when the individual components that are necessary to maintain the Control Room temperature are OPERABLE. The required components include the cooling coils and associated temperature control instrumentation. In addition, the CRETS must be OPERABLE to the extent that air circulation can be maintained.

For MODEs 1, 2, 3, and 4, redundancy is required and both trains must be OPERABLE. The LCO is modified by a Note which indicates that only one CRETS train is required to be OPERABLE during movement of irradiated fuel assemblies when both units are in MODEs 5 or 6, or defueled. Therefore, with both units in other than MODEs 1, 2, 3, or 4, redundancy is not required for movement of irradiated fuel assemblies and only one CRETS train is required to be OPERABLE.

APPLICABILITY

In MODEs 1, 2, 3, and 4, and during movement of irradiated fuel assemblies, the CRETS must be OPERABLE to ensure that the Control Room temperature will not exceed equipment OPERABILITY requirements following isolation of the Control Room.

ACTIONS

A.1

With one CRETS train inoperable in MODEs 1, 2, 3, or 4, action must be taken to restore OPERABLE status within 30 days. In this Condition, the remaining OPERABLE CRETS train is adequate to maintain the Control Room temperature within limits. The 30 day Completion Time is reasonable, based on the low probability of an event occurring requiring Control Room isolation, consideration that the remaining train can provide the required capabilities, and the alternate safety or non-safety-related cooling means that are available.

B.1 and B.2

If the Required Actions and associated Completion Times of Condition A are not met in MODEs 1, 2, 3, or 4, the unit must be placed in a MODE that minimizes the accident risk. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

C.1

During movement of irradiated fuel assemblies, with the required CRETS train inoperable, action must be taken immediately to suspend activities that could result in a release of radioactivity that might require isolation of the Control Room. This places the unit in a condition that minimizes the accident risk. This does not preclude the movement of fuel to a safe position.

D.1

If both CRETS trains are inoperable in MODEs 1, 2, 3, or 4, the CRETS may not be capable of performing the intended function and the unit is in a condition outside the accident analysis. Therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.7.9.1

This SR verifies each required CRETS train has the capability to maintain Control Room temperature ≤ 104°F for ≥ 12 hours in the recirculation mode. During this test, the backup Control Room air conditioner is to be de-energized. This SR consists of a combination of testing. A 24 month Frequency is appropriate, since significant degradation of the CRETS is slow and is not expected over this time period.

REFERENCES

1. UFSAR, Section 9.8.2.3, "Auxiliary Building Ventilating Systems"

B 3.7 PLANT SYSTEMS

B 3.7.10 Emergency Core Cooling System (ECCS) Pump Room Exhaust Filtration System (PREFS)

BASES

BACKGROUND

The ECCS PREFS filters air from the area of the active ECCS components during the recirculation phase of a LOCA.

The ECCS PREFS consists of two independent and redundant fans, a prefilter, a HEPA filter, and an activated charcoal adsorber section for removal of gaseous activity (principally iodines). Ductwork, valves or dampers, and instrumentation also form part of the system.

The ECCS PREFS operates during normal unit operations. During normal operation flow goes through the pre-filter and HEPA filters, but flow through the charcoal adsorbers is bypassed. During emergency operations, the ECCS PREFS dampers are realigned to initiate filtration. The stream of ventilation air discharges through the system filter trains and out the plant stack. The prefilters remove any large particles in the air to prevent excessive loading of the HEPA filters and charcoal adsorbers.

The ECCS PREFS is discussed in Reference 1, and it may be used for normal, as well as post-accident, atmospheric cleanup functions.

APPLICABLE SAFETY ANALYSES

Emergency Core Cooling System PREFS ensures that radioactive materials leaking from the ECCS equipment within the pump room following a LOCA, are filtered prior to reaching the environment as a layer of defense.

The ECCS PREFS satisfies 10 CFR 50.36(c)(2)(ii), Criterion 3.

LCO

The ECCS PREFS is required to be OPERABLE. The ECCS PREFS is considered OPERABLE when the individual components necessary to maintain the ECCS Pump Room filtration are OPERABLE.

The ECCS PREFS is considered OPERABLE when its associated:

a. Fans are OPERABLE;

b. HEPA filter and charcoal adsorber are not excessively restricting flow and are capable of performing their filtration functions; and

c. Ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained.

APPLICABILITY

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

In MODEs 5 and 6, the ECCS PREFS is not required to be OPERABLE, since the ECCS is not required to be OPERABLE.

ACTIONS

A.1

With one ECCS PREFS exhaust fan inoperable, action must be taken to restore OPERABLE status within seven days.

The seven day Completion Time is reasonable, because one ECCS PREFS exhaust fan remains OPERABLE. It can provide the required flow during this period., Additionally, the probability of a DBA occurring during this time period is low.

B.1

With the ECCS PREFS inoperable for reasons other than Condition A, action must be taken to restore OPERABLE status within 24 hours.

The 24 hour Completion Time is reasonable, based on the low probability of a DBA occurring during this time period.

C.1 and C.2

If the ECCS PREFS cannot be restored to OPERABLE status within the associated Completion Time of Required Action A.1 or B.1, the unit must be in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.10.1

Standby systems should be checked periodically to ensure that they function properly. Since the environment and normal operating conditions on this system are not severe, testing ECCS PREFS once a month provides an adequate check on this system. The ECCS PREFS is tested by starting it from the Control Room and ensuring each exhaust fan discharges through the HEPA filter and charcoal adsorber for ≥ 15 minutes. The 31 day Frequency is based on the known reliability of equipment.

SR 3.7.10.2

This SR verifies that the required ECCS PREFS testing is performed in accordance with the VFTP. The ECCS PREFS filter tests are in accordance with portions of Reference 2. 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.

REFERENCES

1. UFSAR, Section 9.8.2.3, "Auxiliary Building Ventilating Systems"
2. Regulatory Guide 1.52, Revision 2, "Design, Testing, and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," March 1978

B 3.7 PLANT SYSTEMS

B 3.7.11 Spent Fuel Pool Exhaust Ventilation System (SFPEVS)

BASES

BACKGROUND

The SFPEVS filters airborne radioactive particulates and gases from the area of the fuel pool following a fuel handling accident.

The SFPEVS consists of two independent, redundant exhaust fans, a HEPA filter, and an activated charcoal adsorber section consisting of two parallel charcoal adsorber banks for removal of gaseous activity (principally iodines). Ductwork, valves or dampers, and instrumentation also form part of the system. The SFPEVS is supplied power by one non-safety-related power supply.

The SFPEVS is operated during normal unit operations. During normal operation, the charcoal adsorbers are bypassed. When filtration of the air is required (i.e., during movement of irradiated fuel assemblies in the Auxiliary Building), normal air discharges from the fuel handling area in the Auxiliary Building and through the system filter train. The prefilters remove any large particles in the air to prevent excessive loading of the HEPA filters and charcoal adsorbers.

The SFPEVS is discussed in Reference 1 Sections 9.8.2.3 and 14.18, because it may be used for normal, as well as post-accident, atmospheric cleanup functions.

APPLICABLE SAFETY ANALYSES

The SFPEVS is designed to mitigate the consequences of a fuel handling accident in which all rods in the fuel assembly are assumed to be damaged. The analysis of the fuel handling accident is given in Reference 1, Section 14.18. The DBA analysis of the fuel handling accident assumes that the SFPEVS is functional. The accident analysis accounts for the reduction in airborne radioactive material provided by this filtration system. The amount of fission products available for release from the Auxiliary Building is determined for a fuel handling accident. These assumptions and the analysis follow the guidance provided in Reference 2.

The SFPEVS satisfies 10 CFR 50.36(c)(2)(ii), Criterion 3.

LCO

The HEPA filter bank, two charcoal adsorber banks, two exhaust fans, and other equipment listed in the Background Section are required to be OPERABLE and in operation.

The SFPEVS is considered OPERABLE when the individual components necessary to control exposure in the Auxiliary Building are OPERABLE. The SFPEVS is considered OPERABLE when its associated:

a. Fans are OPERABLE;

b. HEPA filter and charcoal adsorber banks are not excessively restricting flow, and are capable of performing their filtration functions; and

c. Ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained.

The SFPEVS is considered in operation when an OPERABLE exhaust fan is in operation, discharging through the OPERABLE HEPA Filter and one OPERABLE charcoal adsorber bank.

APPLICABILITY

During movement of irradiated fuel assemblies in the Auxiliary Building, the SFPEVS is required to be OPERABLE and in operation to mitigate the consequences of a fuel handling accident.

ACTIONS

A.1 and A.2

When one SFPEVS charcoal adsorber bank or one SFPEVS exhaust fan, or both, are inoperable, action must be taken to verify an OPERABLE SFPEVS train is in operation, or movement of irradiated fuel assemblies in the Auxiliary Building must be suspended. One OPERABLE SFPEVS train consists of one OPERABLE exhaust fan able to discharge through the OPERABLE HEPA filter and one OPERABLE charcoal adsorber bank. This ensures the proper equipment is operating for the Applicable Safety Analysis.

B.1

When there is no OPERABLE SFPEVS train or there is no OPERABLE SFPEVS train in operation during movement of irradiated fuel assemblies in the Auxiliary Building, action must be taken to place the unit in a condition in which the LCO does not apply. This Action involves immediately suspending movement of irradiated fuel assemblies in the Auxiliary Building. This does not preclude the movement of fuel to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.7.11.1

The SR requires verification every 12 hours that the SFPEVS is in operation. Verification includes verifying that one exhaust fan is operating and discharging through the HEPA filter bank and one charcoal adsorber bank. The Frequency of 12 hours is sufficient considering that the operators will be focused on the movement of irradiated fuel assemblies within the Auxiliary Building. Thus, if anything were to occur to cause cessation of operation of the SFPEVS, it would be quickly identified.

SR 3.7.11.2

This SR verifies the performance of SFPEVS filter testing in accordance with the VFTP. The SFPEVS filter tests are in accordance with portions of Reference 3. The VFTP includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the VFTP.

SR 3.7.11.3

This SR verifies the integrity of the spent fuel storage pool area. The ability of the spent fuel storage pool area to maintain negative pressure with respect to potentially uncontaminated adjacent areas is periodically tested to verify proper function of the SFPEVS. During operation, the spent fuel storage pool area is designed to maintain a slight negative pressure in the spent fuel storage pool area, with respect to adjacent areas, to prevent unfiltered LEAKAGE.

BASES

This test is conducted on a 24 month Frequency. This Frequency is adequate to ensure the SFPEVS is capable of maintaining a negative pressure.

REFERENCES

1. UFSAR
2. Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors (Safety Guide 25)," March 1972
3. Regulatory Guide 1.52, Revision 2, "Design, Testing, and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," March 1978

B 3.7 PLANT SYSTEMS

B 3.7.12 Penetration Room Exhaust Ventilation System (PREVS)

BASES

BACKGROUND

The PREVS filters air from the penetration room.

The PREVS consists of two independent and redundant trains. Each train consists of a prefilter, a 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 following receipt of a containment isolation actuation signal.

The PREVS is a standby system, which may also operate during normal unit operations. During emergency operations, the PREVS dampers are realigned, and fans are started to initiate filtration. Upon receipt of the actuating Engineered Safety Feature Actuation System signal(s), normal air discharges from the penetration room, and the stream of ventilation air discharges through the system filter trains. The prefilters remove any large particles in the air to prevent excessive loading of the HEPA filters and charcoal adsorbers.

The PREVS is discussed in Reference 1, Section 6.6.2, as it may be used for normal, as well as post-accident, atmospheric cleanup functions.

APPLICABLE SAFETY ANALYSES

The design basis of the PREVS is established by the Maximum Hypothetical Accident. The system is credited with filtering the radioactive material released through the containment vent when the line is open. In such a case, the system restricts the radioactive release to within the acceptance criteria given in Reference 1, Chapter 14. All other containment leakage is assumed to be discharged unfiltered directly to the atmosphere. No credit is taken for any leakage to the penetration room and commensurate dose reduction through the PREVS HEPA and charcoal filters. The analysis of the effects and consequences of a Maximum Hypothetical Accident are presented in Reference 1, Section 14.24.

BASES

Penetration Room Exhaust Ventilation System also provides filtered ventilation of radioactive materials leaking from ECCS equipment within the penetration room following an accident, however, credit for this feature was not assumed in the accident analysis (Reference 1, Section 14.24).

The PREVS satisfies 10 CFR 50.36(c)(2)(ii), Criterion 3.

LCO

Two independent and redundant trains of the PREVS are required to be OPERABLE to ensure that at least one train is available, assuming there is a single failure disabling the other train coincident with a loss of offsite power.

The PREVS is considered OPERABLE when the individual components necessary to control radioactive releases are OPERABLE in both trains. A PREVS train is considered OPERABLE when its associated:

a. Fan is OPERABLE;

b. High efficiency particulate air filter and charcoal adsorber are not excessively restricting flow, and are capable of performing the filtration functions; and

c. Ductwork, valves, and dampers are OPERABLE, and circulation can be maintained.

APPLICABILITY

In MODEs 1, 2, and 3, the PREVS is required to be OPERABLE to mitigate the potential radioactive material release from a Maximum Hypothetical Accident.

In MODEs 4, 5, and 6, the PREVS is not required to be OPERABLE, since the RCS temperature and pressure are low and there is insufficient energy to result in the conditions assumed in the accident analysis.

ACTIONS

A.1

With one PREVS train inoperable, action must be taken to restore OPERABLE status within seven days. During this time period, the remaining OPERABLE train is adequate to perform the PREVS function. The seven day Completion Time is reasonable based on the low probability of a DBA occurring during this time period, and the consideration that the remaining train can provide the required capability.

B.1 and B.2

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

SURVEILLANCE REQUIREMENTS

SR 3.7.12.1

Standby systems should be checked periodically to ensure that they function properly. As the environment and normal operating conditions on this system are not severe, testing each train once every month provides an adequate check on this system.

The test is performed by initiating the system from the Control Room, ensuring flow through the HEPA filter and charcoal adsorber train, and verifying this system operates for ≥ 15 minutes. The 31 day Frequency is based on the known reliability of the equipment and the two train redundancy available.

SR 3.7.12.2

This SR verifies the performance of PREVS filter testing in accordance with the VFTP. The PREVS filter tests are in accordance with portions of Reference 2. The VFTP includes testing the performance of the HEPA filter, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the VFTP.

SR 3.7.12.3

This SR verifies that each PREVS train starts and operates on an actual or simulated actuation signal (Containment

BASES

Isolation Signal). This test is conducted on a 24 month Frequency. This Frequency is adequate to ensure the PREVS is capable of starting and operating on an actual or simulated Containment Isolation Signal.

REFERENCES

1. UFSAR

2. Regulatory Guide 1.52, Revision 2, "Design, Testing, and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," March 1978

B 3.7 PLANT SYSTEMS

B 3.7.13 Spent Fuel Pool (SFP) Water Level

BASES

BACKGROUND	The minimum water level in the SFP 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 SFP design is given in Reference 1, Section 9.7.2, and the SFP Cooling and Cleanup System is given in Reference 1, Section 9.4.1. The assumptions of the fuel handling accident are given in Reference 1, Section 14.18.
APPLICABLE SAFETY ANALYSES	The minimum water level in the SFP meets the assumptions of the fuel handling accident described in Reference 2. The resultant two hour thyroid dose to a person at the exclusion area boundary is within the acceptance criteria given in the UFSAR. Reference 2 assumes that there is 23 ft of water between the top of the damaged fuel bundle and the SFP surface for a fuel handling accident. With 21.5 ft of water above the fuel seated in the spent fuel storage racks, the assumptions of Reference 2 can be used directly, because the analysis assumes the dropping of a fuel assembly onto the floor of the SFP. The SFP water level satisfies 10 CFR 50.36(c)(2)(ii), Criteria 2 and 3.
LCO	The specified water level preserves the assumptions of the fuel handling accident analysis (Reference 1, Section 14.18). As such, it is the minimum required for fuel storage and movement within the fuel storage pool.
APPLICABILITY	This LCO applies during movement of irradiated fuel assemblies in the SFP since the potential for a release of fission products exists.

BASES

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply.

When the initial conditions for an accident cannot be met, steps should be taken to preclude the accident from occurring. When the SFP water level is lower than the required level, the movement of irradiated fuel assemblies in the SFP is immediately suspended. This effectively precludes a spent fuel handling accident from occurring. This does not preclude moving a fuel assembly to a safe position.

If moving irradiated fuel assemblies while in MODEs 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODEs 1, 2, 3, and 4, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.7.13.1

This SR verifies sufficient SFP water is available in the event of a fuel handling accident. The water level in the SFP must be checked periodically. The seven day Frequency is appropriate, because the volume in the pool is normally stable. Water level changes are controlled by unit procedures and are acceptable, based on operating experience.

During refueling operations, the level in the SFP is at equilibrium with that of the refueling canal.

REFERENCES

1. UFSAR
2. Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors (Safety Guide 25)," March 1972

B 3.7 PLANT SYSTEMS

B 3.7.14 Secondary Specific Activity

BASES

BACKGROUND

Activity in the secondary coolant results from steam generator tube outleakage from the RCS. Under steady state conditions, the activity is primarily iodines with relatively short half lives, and thus is an indication of current conditions. During transients, DOSE EQUIVALENT 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 100 gallons per day tube leak (LCO 3.4.13) of primary coolant at the limit of 1.0 μCi/gm (LCO 3.4.15). The main SLB is assumed to result in the release of the noble gas and iodine activity contained in the steam generator inventory, the feedwater, and reactor coolant LEAKAGE. Most of the iodine isotopes have short half lives (i.e., < 20 hours).

APPLICABLE SAFETY ANALYSES

The accident analysis of the main SLB, as discussed in Reference 1, assumes the initial secondary coolant specific activity to have a radioactive isotope concentration of 0.10 μCi/gm DOSE EQUIVALENT I-131. This assumption is used in the analysis for determining the radiological consequences of the postulated accident. The accident analysis, based on this and other assumptions, shows that the radiological consequences of a main SLB do not exceed the acceptance criteria given in Reference 1.

With the loss of offsite power, the remaining steam generator is available for core decay heat dissipation by venting steam to the atmosphere through MSSVs and ADVs. The AFW System supplies the necessary makeup to the steam generator. Venting continues until the reactor coolant

BASES

temperature and pressure have decreased sufficiently for the SDC System to complete the cooldown.

In the evaluation of the radiological consequences of this accident, the activity released from the steam generator connected to the failed steam line is assumed to be released directly to the environment. The unaffected steam generator is assumed to discharge steam and any entrained activity through MSSVs and ADVs during the event.

Secondary specific activity limits satisfy 10 CFR 50.36(c)(2)(ii), Criterion 2.

LCO

As indicated in the Applicable Safety Analyses, the specific activity limit in the secondary coolant system of $\leq 0.10\ \mu Ci/gm$ DOSE EQUIVALENT I-131 to limit the radiological consequences of a DBA to the acceptance criteria given in Reference 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.

APPLICABILITY

In MODEs 1, 2, 3, and 4, the limits on secondary specific activity apply due to the potential for secondary steam releases to the atmosphere.

In MODEs 5 and 6, the steam generators are not being used for heat removal. Both the RCS and steam generators are depressurized, and primary to secondary LEAKAGE is minimal. Therefore, monitoring of secondary specific activity is not required.

ACTIONS

A.1 and A.2

DOSE EQUIVALENT I-131 exceeding the allowable value in the secondary coolant, is an indication of a problem in the RCS, and contributes to increased post-accident doses. If secondary specific activity cannot be restored to within limits in the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.14.1

This SR ensures that the secondary specific activity is within the limits of the accident analysis. A gamma isotope analysis of the secondary coolant, which determines DOSE EQUIVALENT I-131, confirms the validity of the safety analysis assumptions as to the source terms in post-accident releases. It also serves to identify and trend any unusual isotopic concentrations that might indicate changes in reactor coolant activity or LEAKAGE. The 31 day Frequency is based on the detection of increasing trends of the level of DOSE EQUIVALENT I-131, and allows for appropriate action to be taken to maintain levels below the LCO limit.

REFERENCES

1. UFSAR, Chapter 14, "Safety Analysis"

B 3.7 PLANT SYSTEMS

B 3.7.15 Main Feedwater Isolation Valves (MFIVs)

BASES

BACKGROUND

The MFIVs isolate MFW flow to the secondary side of the steam generators following a HELB. The consequences of HELBs occurring in the main steam lines or in the MFW lines downstream of the MFIVs will be mitigated by their closure. Closure of the MFIVs effectively terminates the addition of feedwater to an affected steam generator, limiting the mass and energy release for SLBs/or feedwater line breaks (FWLBs) inside the Containment Structure upstream of the reverse flow check valve, and reducing the cooldown effects for SLBs.

The MFIVs isolate the non-safety-related portions from the safety-related portion of the system. In the event of a secondary side pipe rupture inside the Containment Structure upstream of the reverse flow check valve, the valves limit the quantity of high energy fluid that enters the Containment Structure through the break.

One MFIV is located on each MFW line, outside, but close to, the Containment Structure. The MFIVs are located so that AFW may be supplied to a steam generator following MFIV closure. The piping volume from the valve to the steam generator must be accounted for in calculating mass and energy releases.

The MFIVs close on receipt of a steam generator isolation signal generated by low steam generator pressure. The steam generator isolation signal also actuates the MSIVs to close. The MFIVs may also be actuated manually. In addition, the MFIVs reverse flow check valve inside the Containment Structure is available to isolate the feedwater line penetrating the Containment Structure, and to ensure that the consequences of events do not exceed the capacity of the Containment Cooling System.

A description of the MFIVs operation on receipt of an steam generator isolation signal is found in Reference 1.

BASES

APPLICABLE SAFETY ANALYSES	The design basis of the MFIVs is established by the analysis for the large SLB. It is also influenced by the accident analysis for the large FWLB. Closure of the MFIVs and their bypass valves is also relied on to terminate a steam break for core response analysis and an excess load event upon receipt of a main steam isolation signal on high steam generator level. Failure of an MFIV to close following an SLB or FWLB can result in additional mass and energy to the steam generator's contributing to cooldown. This failure also results in additional mass and energy releases following an SLB or FWLB event. The MFIVs satisfy 10 CFR 50.36(c)(2)(ii), Criterion 3.
LCO	This LCO ensures that the MFIVs will isolate MFW flow to the steam generators. Following an FWLB or SLB, these valves will also isolate the non-safety-related portions from the safety-related portions of the system. This LCO requires that one MFIV in each feedwater line be OPERABLE. The MFIVs are considered OPERABLE when the isolation times are within limits, and are closed on an isolation actuation signal. Failure to meet the LCO requirements can result in additional mass and energy being released to the Containment Structure following an SLB or FWLB inside the Containment Structure. Failure to meet the LCO can also add additional mass and energy to the steam generators contributing to cooldown.
APPLICABILITY	The MFIVs must be OPERABLE whenever there is significant mass and energy in the RCS and steam generators. In MODEs 1, 2, and 3, the MFIVs are required to be OPERABLE in order to limit the amount of available fluid that could be added to the Containment Structure in the case of a secondary system pipe break inside the Containment Structure. In MODEs 4, 5, and 6, steam generator energy is low.

BASES

ACTIONS

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

A.1

With one MFIV inoperable, action must be taken to restore the valve to OPERABLE status within 72 hours.

The 72 hour Completion Time takes into account the isolation capability afforded by the MFW regulating valves, and tripping of the MFW pumps, and the low probability of an event occurring during this time period that would require isolation of the MFW flow paths.

B.1 and B.2

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

SURVEILLANCE REQUIREMENTS

SR 3.7.15.1

This SR ensures the closure time for each MFIV is $\leq$ 80 seconds by manual isolation. The MFIV closure time is assumed in the accident and containment analyses.

The Frequency is in accordance with the Inservice Testing Program. The MFIVs are tested during each refueling outage in accordance with Reference 2, and sometimes during other cold shutdown periods. The Frequency demonstrates the valve closure time at least once per refueling cycle. Operating experience has shown that these components usually pass the surveillance test when performed.

BASES

REFERENCES

1. UFSAR, Section 14.4.2, "Sequence of Events"
2. ASME, Boiler and Pressure Vessel Code, Section XI, 1989, "Rules for In-Service Inspection of Nuclear Power Plant Components," and ASME Operation and Maintenance Code Part 10, 1987, with 1988 Addenda

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources-Operating

BASES

BACKGROUND

The AC sources to the Class 1E Electrical Power Distribution System consist of the offsite power sources starting at the 4.16 kV engineered safety feature (ESF) buses and the onsite diesel generators (DGs). As required by Reference 1, General Design Criteria (GDC) 17, the design of the AC electrical power system has sufficient independence and redundancy to ensure a source to the ESFs assuming a single failure.

The Class 1E AC Distribution System is divided into two redundant load groups so that the loss of one group does not prevent the minimum safety functions from being performed. Each load group has connections to two offsite sources and one Class 1E DG at its 4.16 kV 1E bus.

Offsite power is supplied to the 500 kV Switchyard from the transmission network by three 500 kV transmission lines. Two electrically and physically separated circuits supply electric power from the 500 kV Switchyard to two 13 kV buses and then to the two 4.16 kV ESF buses. A third 69 kV/ 13.8 kV offsite power source that may be manually connected to either 13 kV bus is available from the Southern Maryland Electric Cooperative (SMECO). When appropriate, the Engineered Safety Feature Actuation System (ESFAS) loss of coolant incident and shutdown sequencer for the 4.16 kV bus will sequence loads on the bus after the 69 kV/13.8 kV SMECO line has been manually placed in service. The SMECO offsite power source will not be used to carry loads for an operating unit. A detailed description of the offsite power network and the circuits to the Class 1E ESF buses, is found in Reference 2, Chapter 8.

The required offsite power circuits are the two 13 kV buses (Nos. 11 and 21) which can be powered by:

a. Two 500 kV lines, two 500 kV buses each of which have connections to a 500 kV line that does not pass through the other 500 kV bus and both P-13000 (500 kV/14 kV) transformers; or

b. One 500 kV line, one 500 kV bus, and one associated P-13000 (500 kV/14 kV) transformer, and the 69 kV/ 13.8 kV SMECO line. When the SMECO line is credited as one of the qualified offsite circuits, the disconnect from the SMECO line to Warehouse No. 1 must be open.

In addition, each offsite circuit includes the cabling to and from a 13.8/13.8 voltage regulator, the voltage regulator, 13.8/4.16 kV unit service transformer, the unit service transformer, and one of the two breakers to one 4.16 kV ESF bus. Transfer capability between the two required offsite circuits is by manual means only. The required circuit breaker to each 4.16 kV ESF bus must be from different 13.8/4.16 unit service transformers for the two required offsite circuits. Thus, each unit is able to align one 4.16 kV bus to one required offsite circuit, and the other 4.16 kV bus to the other required offsite circuit.

In some cases, inoperable components in the electrical circuit place both units in Conditions. Examples of these are 13.8 kV bus Nos. 11 or 21, two 500 kV transmission lines, one P-13000 service transformer, or one 500 kV bus. In other cases, inoperable components only place one unit in a Condition, such as an inoperable U-4000 and/or 13.8 kV regulator that feeds a required 4.16 kV bus.

The onsite standby power source to each 4.16 kV ESF bus is a dedicated DG. A DG starts automatically on an safety injection actuation signal or on a 4.16 kV degraded or undervoltage signal. If both 4.16 kV offsite source breakers are open, the DG, after reaching rated voltage and frequency, will automatically close onto the 4.16 kV bus.

In the event of a loss of offsite power to a 4.16 kV 1E bus, if required, the ESF electrical loads will be automatically sequenced onto the DG in sufficient time to provide for safe shutdown for an anticipated operational occurrence (AOO) and to ensure that the containment integrity and other vital functions are maintained in the event of a design bases accident.

Ratings for the No. 1A DG satisfies the requirements of Reference 3 and ratings for the Nos. 1B, 2A, and 2B DGs satisfy the requirements of Reference 4. The continuous service rating for the No. 1A DG is 5400 kW and for the Nos. 1B, 2A, and 2B DGs are 3000 kW.

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in Reference 1, Chapters 6 and 14, assume ESF systems are OPERABLE. The AC electrical power sources are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, RCS, and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for Sections 3.2, 3.4, and 3.6.

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; and

b. A single failure.

The AC sources satisfy 10 CFR 50.36(c)(2)(ii), Criterion 3.

LCO

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

Qualified offsite circuits are those that are described in the Updated Final Safety Analysis Report (UFSAR) and are part of the licensing basis for the unit.

Each offsite circuit must be capable of maintaining rated frequency and voltage and accepting required loads during an accident, while connected to the ESF buses. Loads are immediately connected to the ESF buses when the buses are powered from the 500 kV offsite circuits and, when powered from the 69/13.8 kV SMECO offsite circuit after being manually connected, the loads are sequenced onto the ESF bus utilizing the same sequencer used to sequence the loads onto the DG. The SMECO offsite circuit will not be used to carry loads for an operating unit.

The Limiting Condition for Operation (LCO) requires operability of two out of three qualified circuits between the transmission network and the onsite Class 1E AC Electrical Power Distribution System circuits. These circuits consist of two 500 kV circuits via 500 kV/14 kV and 13.8 kV/4.16 kV transformers and the 69 kV SMECO dedicated source (described in Reference 5) via 69 kV/ 13.8 kV and 13.8 kV/4.16 kV transformers. In addition, each offsite circuit includes one of the two breakers to one 4.16 kV ESF bus. The required circuit breaker to each 4.16 kV ESF bus must be from different 13.8/4.16 unit service transformers for the two required offsite circuits. Thus, each unit is able to align one 4.16 kV bus to one required offsite circuit, and the other 4.16 kV bus to the other required offsite circuit.

Each DG must be capable of starting, accelerating to rated speed and voltage, and connecting to its respective ESF bus on detection of bus undervoltage. This will be accomplished within 10 seconds. Each DG must also be capable of accepting required loads within the assumed loading sequence intervals, and continue to operate until offsite power can be restored to the ESF buses. These capabilities are required to be met from a variety of initial conditions such as DG in standby with the engine hot and DG in standby with the engine at ambient conditions. Additional DG capabilities must be demonstrated to meet required Surveillances, e.g., capability of the DG to reject a load ≥ 500 hp without tripping.

Proper sequencing of loads, including shedding of non-essential loads, is a required function for DG OPERABILITY in MODEs 1, 2, and 3.

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.

The Control Room Emergency Ventilation System (CREVS) Control Room Emergency Temperature System (CRETS), and H_2 Analyzer are shared systems with one train of each system connected to an onsite Class 1E AC electrical power distribution subsystem from each unit. Limiting Condition for Operation 3.8.1.c requires one qualified circuit between the offsite transmission network and the other unit's onsite Class 1E AC electrical power distribution subsystems needed to supply power to the CREVS, CRETS, and H_2 Analyzer to be OPERABLE and one DG from the other unit capable of supplying power to the CREVS, CRETS, and H_2 Analyzer to be OPERABLE. The qualified circuit in LCO 3.8.1.c must be separate and independent (to the extent possible) of the qualified circuit which provides power to the other train of the CREVS, CRETS, and H_2 Analyzer. These requirements, in conjunction with the requirements for the unit AC electrical power sources in LCO 3.8.1.a and LCO 3.8.1.b, ensure that power is available to two trains of the CREVS, CRETS, and H_2 Analyzer.

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

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.

ACTIONS

A.1

To ensure a highly reliable power source remains with the one required LCO 3.8.1.a offsite circuit inoperable, it is necessary to verify the OPERABILITY of the remaining required offsite circuits on a more frequent basis. Since the Required Action only specifies "perform," a failure of

Surveillance Requirement (SR) 3.8.1.1 or SR 3.8.1.2 acceptance criteria does not result in a Required Action not met. However, if a second required circuit fails SR 3.8.1.1 or SR 3.8.1.2, the second offsite circuit is inoperable, and Condition C and/or E, as applicable, for the two offsite circuits inoperable, is entered.

A.2

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

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

a. The train has no offsite power supplying its loads; and

b. A required feature on another train is inoperable.

If at any time during the existence of Condition A (one required LCO 3.8.1.a offsite circuit inoperable) a redundant required feature subsequently becomes inoperable, this Completion Time begins to be tracked.

The Completion Time must be started if it is discovered that there is no offsite power to one train of the onsite Class 1E Electrical Power Distribution System coincident with one or more inoperable required support or supported features (or both) that are associated with the other train that has offsite power. 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 circuits and DGs are adequate to supply electrical power to Train A and Train B of the onsite Class 1E Distribution System. The 24 hour Completion Time takes into account the component OPERABILITY of the redundant counterpart to the inoperable required feature. Additionally, the 24 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

A.3

Consistent with Reference 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 Distribution System.

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

The second Completion Time for Required Action A.3 establishes a limit on the maximum time allowed for any combination of required AC power sources to be inoperable during any single contiguous occurrence of failing to meet LCO 3.8.1.a or LCO 3.8.1.b. If Condition A is entered while, for instance, an LCO 3.8.1.b DG is inoperable, and that DG is subsequently returned OPERABLE, the LCO may already have been not met for up to 72 hours. This could lead to a total of 144 hours, since initial failure to meet LCO 3.8.1.a or LCO 3.8.1.b, to restore the offsite circuit. At this time, a LCO 3.8.1.b DG could again become inoperable, the circuit restored OPERABLE, and an additional 72 hours (for a total of nine days) allowed prior to complete restoration of LCOs 3.8.1.a and 3.8.1.b. The six day Completion Time provides a limit on the time allowed in a specified condition after discovery of failure to meet

LCO 3.8.1.a or LCO 3.8.1.b. This limit is considered reasonable for situations in which Conditions A and B are entered concurrently. The "AND" connector between the 72 hour and six day Completion Time means that both Completion Times apply simultaneously, and the more restrictive Completion Time must be met.

As in Required Action A.2, the Completion Time allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." This will result in establishing the "time zero" at the time that LCO 3.8.1.a or LCO 3.8.1.b was initially not met, instead of at the time Condition A was entered.

B.1

To ensure a highly reliable power source remains with an inoperable LCO 3.8.1.b DG, it is necessary to verify the availability of the offsite circuits on a more frequent basis. Since the Required Action only specifies "perform," a failure of SR 3.8.1.1 or SR 3.8.1.2 acceptance criteria does not result in a Required Action being not met. However, if a circuit fails to pass SR 3.8.1.1 or SR 3.8.1.2, 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 LCO 3.8.1.b DG is inoperable, does not result in a complete loss of safety function of critical systems. These features are designed with redundant safety-related trains. Single train systems are not included. Redundant required feature failures consist of inoperable features with a train, redundant to the train that has an inoperable LCO 3.8.1.b DG.

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

Action, the Completion Time only begins on discovery that both:

a. An inoperable LCO 3.8.1.b DG exists; and

b. A required feature on another train is inoperable.

If at any time during the existence of this Condition (one LCO 3.8.1.b DG inoperable) a required feature subsequently becomes inoperable, this Completion Time begins to be tracked.

Discovering one required LCO 3.8.1.b DG inoperable coincident with one or more inoperable required support or supported features (or both) that are associated with the OPERABLE DGs, 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 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 four hour Completion Time takes into account the OPERABILITY of the redundant counterpart to the inoperable required feature. Additionally, the four 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.

<u>B.3.1 and B.3.2</u>

Required Action B.3.1 provides an allowance to avoid unnecessary testing of OPERABLE DGs. If it can be determined that the cause of the inoperable DG does not exist on the OPERABLE DG(s), SR 3.8.1.3 does not have to be performed. If the cause of inoperability exists on other DG(s), the other DG(s) would be declared inoperable upon discovery and Condition D and/or G of LCO 3.8.1, as applicable, would be entered. Once the failure is repaired, the common cause failure no longer exists and Required Action B.3.1 is satisfied. If the cause of the initial inoperable DG cannot be confirmed not to exist on the remaining DG(s), performance of SR 3.8.1.3 suffices to provide assurance of continued OPERABILITY of the DG(s).

In the event the inoperable DG 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.

Consistent with Reference 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

Consistent with Reference 6, operation may continue in Condition B for a period that should not exceed 72 hours.

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

The second Completion Time for Required Action B.4 establishes a limit on the maximum time allowed for any combination of required AC power sources to be inoperable during any single contiguous occurrence of failing to meet LCO 3.8.1.a or LCO 3.8.1.b. If Condition B is entered while, for instance, an LCO 3.8.1.a offsite circuit is inoperable and that circuit is subsequently returned OPERABLE, the LCO may already have not been met for up to 72 hours. This could lead to a total of 144 hours, since initial failure to meet LCO 3.8.1.a or LCO 3.8.1.b, to restore the DG. At this time, a LCO 3.8.1.a offsite circuit could again become inoperable, the DG restored OPERABLE, and an additional 72 hours (for a total of nine days) allowed prior to complete restoration of LCO 3.8.1.a and LCO 3.8.1.b. The six day Completion Time provides a limit on time allowed in a specified condition after discovery of failure to meet LCO 3.8.1.a or LCO 3.8.1.b. This limit is considered reasonable for situations in which Conditions A and B are entered concurrently. The "AND" connector between the 72 hour and six day Completion Times means that both Completion Times apply simultaneously, and the more restrictive Completion Time must be met.

As in Required Action B.2, the Completion Time allows for an exception to the normal "time zero" for beginning the allowed time "clock." This will result in establishing the "time zero" at the time that LCO 3.8.1.a or LCO 3.8.1.b was initially not met, instead of at the time Condition B was entered.

C.1

Pursuant to LCO 3.0.6, the Distribution System ACTIONS would not be entered even if all AC sources to it, were inoperable resulting in de-energization. Therefore, the Required Actions of Condition C are modified by a Note to indicate that when Condition C is entered with no AC source to any train, the Conditions and Required Actions for LCO 3.8.9, must be immediately entered. This allows Condition C to provide requirements for the loss of the LCO 3.8.1.c offsite circuit and DG without regard to whether a train is de-energized. Limiting Condition for Operation 3.8.9 provides the appropriate restrictions for a de-energized train.

To ensure a highly reliable power source remains with the one required LCO 3.8.1.c offsite circuit inoperable, it is necessary to verify the OPERABILITY of the remaining required offsite circuits on a more frequent basis. Since the Required Action only specifies "perform," a failure of SR 3.8.1.1 or SR 3.8.1.2 acceptance criteria does not result in a Required Action not met. However, if a second required circuit fails SR 3.8.1.1 or SR 3.8.1.2, the second offsite circuit is inoperable, and Condition A and/or E, as applicable, for the two offsite circuits inoperable, is entered.

C.2

Required Action C.2, which only applies if the train cannot be powered from an offsite source, is intended to provide assurance that an event coincident with a single failure of the associated DG will not result in a complete loss of safety function for the CREVS, CRETS, or the H_2 Analyzers. 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. The train has no offsite power supplying its loads; and

b. A train of CREVS, CRETS, or H_2 Analyzer on the other train is inoperable.

If at any time during the existence of Condition C (one required LCO 3.8.1.c offsite circuit inoperable) a train of CREVS, CRETS, or H_2 Analyzer becomes inoperable, this Completion Time begins to be tracked.

Discovering no offsite power to one train of the onsite Class 1E Electrical Power Distribution System coincident with one train of CREVS, CRETS or H_2 Analyzer that is 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 circuits and DGs are adequate to supply electrical power to Train A and Train B of the onsite Class 1E Distribution System. The 24 hour Completion Time takes into account the component OPERABILITY of the redundant counterpart to the inoperable CREVS, CRETS, or H_2 Analyzer. 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.

C.3

Consistent with the time provided in ACTION A, operation may continue in Condition C for a period that should not exceed 72 hours. With one required LCO 3.8.1.c 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 circuits and DGs are adequate to supply electrical power to the onsite Class 1E Distribution System.

If the LCO 3.8.1.c required offsite circuit cannot be restored to OPERABLE status within 72 hours, the CREVS, CRETS, and H_2 Analyzer associated with the offsite circuit must be declared inoperable. The ACTIONS associated with the CREVS, CRETS, and H_2 Analyzer will ensure the appropriate actions are taken. 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.

D.1

Pursuant to LCO 3.0.6, the Distribution System ACTIONS would not be entered even if all AC sources to it, were inoperable resulting in de-energization. Therefore, the Required Actions of Condition D are modified by a Note to indicate that when Condition D is entered with no AC source to any train, the Conditions and Required Actions for LCO 3.8.9 must be immediately entered. This allows Condition D to provide requirements for the loss of the LCO 3.8.1.c offsite circuit and DG without regard to whether a train is de-energized. Limiting Condition for Operation 3.8.9 provides the appropriate restrictions for a de-energized train.

To ensure a highly reliable power source remains with the one required LCO 3.8.1.c DG inoperable, it is necessary to verify the availability of the required offsite circuits on a more frequency basis. Since the Required Action only specifies "perform," a failure of SR 3.8.1.1 or SR 3.8.1.2 acceptance criteria does not result in a Required Action not met. However, if a circuit fails to pass SR 3.8.1.1 or SR 3.8.1.2, it is inoperable. Upon offsite circuit inoperability additional Conditions and Required Actions must then be entered.

D.2

Required Action D.2 is intended to provide assurance that a loss of offsite power, during the period the LCO 3.8.1.c DG is inoperable, does not result in a complete loss of safety function for the CREVS, CRETS, or the H_2 Analyzers. The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." In this Required Action, the Completion Time only begins on discovery that both:

a. An inoperable LCO 3.8.1.c DG exists; and

b. A train of CREVS, CRETS, or H_2 Analyzers on the other train is inoperable.

If at any time during the existence of this Condition (the LCO 3.8.1.c DG inoperable) a train of CREVS, CRETS, or H_2 Analyzer becomes inoperable, this Completion Time begins to be tracked.

Discovering the LCO 3.8.1.c DG inoperable coincident with one train of CREVS, CRETS, or H_2 Analyzer that is associated with the one LCO 3.8.1.b 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 Distribution System. Thus, on a component basis, single failure protection for the CREVS, CRETS, or H_2 Analyzer may have been lost; however, function has not been lost. The four hour Completion Time also takes into account the capacity and capability of the remaining CREVS, CRETS, and H_2 Analyzer train, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

D.3.1 and D.3.2

Required Action D.3.1 provides an allowance to avoid unnecessary testing of OPERABLE DGs. If it can be determined that the cause of the inoperable DG does not exist on the OPERABLE DG(s), SR 3.8.1.3 does not have to be performed. If the cause of inoperability exists on other DG(s), the other DG(s) would be declared inoperable upon discovery and Condition B and/or F of LCO 3.8.1, as applicable, would be entered. Once the failure is repaired, the common cause failure no longer exists and Required Action D.3.1 is satisfied. If the cause of the initial inoperable DG cannot be confirmed not to exist on the remaining DG(s), performance of SR 3.8.1.3 suffices to provide assurance of continued OPERABILITY of the DG(s).

In the event the inoperable DG is restored to OPERABLE status prior to completing either D.3.1 or D.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 D.

Consistent with Reference 6, 24 hours is reasonable to confirm that the OPERABLE DG(s) is not affected by the same problem as the inoperable DG.

D.4

Consistent with the time provided in ACTION B, operation may continue in Condition D for a period that should not exceed 72 hours. In Condition D, the remaining OPERABLE DGs and offsite power circuits are adequate to supply electrical power to the Class 1E Distribution System.

If the LCO 3.8.1.c DG cannot be restored to OPERABLE status within 72 hours the CREVS, CRETS, and H_2 Analyzer associated with this DG must be declared inoperable. The Actions associated with the CREVS, CRETS, and H_2 Analyzer will ensure the appropriate Actions are taken.

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.

E.1 and E.2

Condition E is entered when both offsite circuits required by LCO 3.8.1.a are inoperable, or when the offsite circuit required by LCO 3.8.1.c and one offsite circuit required by LCO 3.8.1.a are concurrently inoperable, if the LCO 3.8.1.a offsite circuit is credited with providing power to the CREVS, CRETS, and H_2 Analyzer.

Required Action E.1 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 Reference 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. Single train features are not included in the list.

The Completion Time for Required Action E.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. Two required offsite circuits are inoperable; and

b. A required feature is inoperable.

If at any time during the existence of Condition E (e.g., two required LCO 3.8.1.a offsite circuits inoperable) and a required feature becomes inoperable, this Completion Time begins to be tracked.

Consistent with Reference 6, operation may continue in Condition E 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 could correspond to a total loss of the immediately accessible offsite power sources.

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

a. The configuration of the redundant AC electrical power system that remains available is not susceptible to a single bus or switching failure; and

b. The time required to detect and restore an unavailable offsite power source is generally much less than that required to detect and restore an unavailable onsite AC source.

With two of the required offsite circuits inoperable, sufficient onsite AC sources are available to maintain the unit in a safe shutdown condition in the event of a DBA or transient. In fact, a simultaneous loss of offsite AC sources, a loss of coolant accident, 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.

Consistent with 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 required within 24 hours, power operation continues in accordance with Condition A or C, as applicable.

F.1 and F.2

Pursuant to LCO 3.0.6, the Distribution System ACTIONS would not be entered even if all AC sources to it were inoperable resulting in de-energization. Therefore, the Required Actions of Condition F are modified by a Note to indicate that when Condition F is entered with no AC source to any train, the Conditions and Required Actions for LCO 3.8.9, must be immediately entered. This allows Condition F to provide requirements for the loss of one required LCO 3.8.1.a offsite circuit and one LCO 3.8.1.b DG without regard to whether a train is de-energized. Limiting Condition for Operation 3.8.9 provides the appropriate restrictions for a de-energized train.

Consistent with Reference 6, operation may continue in Condition F for a period that should not exceed 12 hours.

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

G.1

With two LCO 3.8.1.b DGs inoperable, there are no remaining standby AC sources to provide power to most of the ESF systems. With one LCO 3.8.1.c DG inoperable and the LCO 3.8.1.b DG that provides power to the CREVS, CRETS, and H_2 Analyzer inoperable, there are no remaining standby AC sources to the CREVS, CRETS, and H_2 Analyzers. 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 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.

Consistent with Reference 6, with both LCO 3.8.1.b DGs inoperable, or with the LCO 3.8.1.b DG that provides power to the CREVS, CRETS, and H_2 Analyzer and the LCO 3.8.1.c DG inoperable, operation may continue for a period that should not exceed 2 hours.

H.1 and H.2

If any Required Action and associated Completion Time of Conditions A, B, E, F, or G are 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 six hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

I.1

Condition I corresponds to a level of degradation in which all redundancy in LCO 3.8.1.a and LCO 3.8.1.b AC electrical power supplies has been lost. At this severely degraded level, any further losses in the AC electrical power system will cause a loss of function. Therefore, no additional time is justified for continued operation. The unit is required by LCO 3.0.3 to commence a controlled shutdown.

SURVEILLANCE REQUIREMENTS

The AC sources are designed to permit inspection and testing of all important areas and features, especially those that have a standby function, in accordance with Reference 1, GDC 18. 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 consistent with the recommendations of Reference 3, or Reference 4, and Reference 8.

When the SRs discussed herein specify voltage and frequency tolerances, the following is applicable. The minimum transient output voltage of 3740 V is 90% of the nominal 4160 V output voltage. This value allows for voltage drop to the terminals of 4000 V motors whose minimum operating voltage is specified as 90% or 3600 V. The specified maximum output voltage of 4400 V is equal to the maximum operating voltage specified for 4000 V motors. It ensures that for a lightly loaded distribution system, the voltage at the terminals of 4000 V is no more than the maximum rated operating voltages. The specified minimum and maximum frequencies of the DG are 58.8 Hz and 61.2 Hz, respectively. These values are equal to +/- 2% of the 60 Hz nominal frequency and are the recommendations given in Reference 3.

The SRs are modified by a Note which states that SR 3.8.1.1 through SR 3.8.1.15 are applicable to LCO 3.8.1.a and LCO 3.8.1.b AC Sources. The Note also states that SR 3.8.1.16 is applicable to LCO 3.8.1.c AC sources. This Note clarifies that not all of the SRs are applicable to all the components described in the LCO.

<u>SR 3.8.1.1 and SR 3.8.1.2</u>

These SRs assure 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 Frequency of once within one hour after substitution for a 500 kV circuit and every eight hours thereafter, for SR 3.8.1.1 was established to ensure that the breaker alignment for the SMECO circuit (which does not have Control Room indication) is in its correct position although breaker position is unlikely to change. The seven day Frequency for SR 3.8.1.2 is adequate since the 500 kV circuit breaker position is not likely to change without the operator being aware of it and because its status is displayed in the Control Room.

Surveillance Requirement 3.8.1.1 is modified by a Note which states that this SR is only required when SMECO is being credited for an offsite source. This SR will prevent unnecessary testing on an uncredited circuit.

SR 3.8.1.3 and SR 3.8.1.9

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

To minimize the wear on moving parts that do not get lubricated when the engine is not running, these SRs are modified by a Note (Note 2 for SR 3.8.1.3) to indicate that all DG starts for these surveillance tests may be preceded by an engine prelube period and followed by a warmup period prior to loading by an engine prelube period.

For the purposes of SR 3.8.1.9 testing, the DGs are required to start from standby conditions only for SR 3.8.1.9. Standby conditions for a DG mean the diesel engine coolant and oil are being continuously circulated and temperature is being maintained consistent with manufacturer recommendations.

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

Surveillance Requirement 3.8.1.9 requires that, at a 184 day Frequency, the DG starts from standby conditions and achieves required voltage and frequency within 10 seconds. The minimum voltage and frequency stated in the SR are those necessary to ensure the DG can accept DBA loading while maintaining acceptable voltage and frequency levels. The

10 second start requirement supports the assumptions of the design basis loss of coolant accident analysis in Reference 2, Chapter 14.

Since SR 3.8.1.9 requires a 10 second start, it is more restrictive than SR 3.8.1.3, and it may be performed in lieu of SR 3.8.1.3.

The 31 day Frequency for SR 3.8.1.3 is consistent with Reference 4 and Reference 3. The 184 day Frequency for SR 3.8.1.9 is a reduction in cold testing consistent with Reference 7. This Frequency provides adequate assurance of DG OPERABILITY, while minimizing degradation resulting from testing.

SR 3.8.1.4

This SR verifies that the DGs are capable of synchronizing with the offsite electrical system and accepting loads greater than or equal to 4000 kW for No. 1A DG and greater than or equal to 90% of the continuous duty rating for the remaining DGs. The 90% minimum load limit is consistent with Reference 3 and is acceptable because testing of these DGs at post-accident load values is performed by SR 3.8.1.11. A minimum run time of 60 minutes is required to stabilize engine temperatures, while minimizing the time that the DG is connected to the offsite source.

Although no power factor requirements are established by this SR, the DG is normally operated at a power factor between 0.8 lagging and 1.0. The 0.8 value is the design rating of the machine, while 1.0 is an operational limitation. The 31-day Frequency for this SR is consistent with Reference 3.

This SR is modified by four Notes. Note 1 indicates that the diesel engine runs for this surveillance test may include gradual loading, as recommended by the manufacturer, so that mechanical stress and wear on the diesel engine are minimized. Note 2 states that momentary transients because of changing bus loads do not invalidate this test. Note 3 indicates that this surveillance test shall be conducted on only one DG at a time in order to prevent routinely paralleling multiple DGs and to minimize the potential for effects 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 3.8.1.5

This SR provides verification that the level of fuel oil in the day tank is at or above the level at which fuel oil is automatically added. The level required by the SR is expressed as an equivalent volume in gallons, and is selected to ensure adequate fuel oil for a minimum of one hour of DG operation at full load plus 10%.

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

SR 3.8.1.6

Microbiological fouling is a major cause of fuel oil degradation. There are numerous bacteria that can grow in fuel oil and cause fouling, but all must have a water environment in order to survive. Removal of water from the fuel oil day tanks once every 31 days eliminates the necessary environment for bacterial survival. This is the most effective means of controlling microbiological fouling. In addition, it eliminates the potential for water entrainment in the fuel oil during DG operation. Water may come from any of several sources, including condensation, ground water, rain water, contaminated fuel oil, and from breakdown of the fuel oil by bacteria. Frequent checking for and removal of accumulated water minimizes fouling and provides data regarding the watertight integrity of the fuel oil system. The SR Frequencies are consistent with Reference 8. This SR is for preventive maintenance. The presence of water does not necessarily represent failure of this SR provided the accumulated water is removed during the performance of this surveillance test.

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SR 3.8.1.7

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

The Frequency for this SR is 31 days. The 31-day Frequency corresponds to the design of the fuel transfer system. The design of fuel transfer systems is such that pumps will operate automatically or must be started manually in order to maintain an adequate volume of fuel oil in the day tanks during or following DG testing. In such a case, a 31-day Frequency is appropriate.

SR 3.8.1.8

Under accident and loss of offsite power conditions loads are sequentially connected to the bus by the automatic load sequencer (this SR verifies steps 1 through 5). The sequencing logic controls the permissive and closing signals to breakers to prevent overloading of the DGs due to high motor starting currents. The 10% load sequence time interval tolerance ensures that sufficient time exists for the DG to restore frequency and voltage prior to applying the next load, and that safety analysis assumptions regarding ESF equipment time delays are not violated. The UFSAR provides a summary of the automatic loading of ESF buses.

The Frequency of 31 days is consistent with DG monthly testing and is sufficient to ensure the load sequencer operation as required.

SR 3.8.1.9

See SR 3.8.1.3.

SR 3.8.1.10

Transfer of each 4.16 kV ESF bus power supply from the normal offsite circuit to the alternate offsite circuit demonstrates the OPERABILITY of the alternate circuit distribution network to power the shutdown loads. The 24 month Frequency of the Surveillance is based on engineering judgment, taking into consideration the unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths. Operating experience has shown that these components usually pass the SR when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.8.1.11

This SR provides verification that the DG can be operated at a load greater than predicted accident loads for at least 60 minutes once per 24 months. Operation at the greater than calculated accident loads will clearly demonstrate the ability of the DGs to perform their safety function. In order to ensure that the DG is tested under load conditions that are as close to design conditions as possible, testing must be performed using a DG load greater than or equal to calculated accident load and using a power factor ≤ 0.85. This power factor is chosen to be representative of the actual design basis inductive loading that the DG could experience. In addition, the post-accident load for No. 1A DG is significantly lower than the continuous rating of No. 1A DG. To ensure No. 1A DG performance is not degraded, routine monitoring of engine parameters should be performed during the performance of this SR for No. 1A DG (Reference 9).

This SR is modified by a Note which states that momentary transients due to changing bus loads do not invalidate this test. Similarly, momentary power factor transients above the limit will not invalidate the test. The 24 month Frequency is adequate to ensure DG OPERABILITY and it is consistent with the refueling interval.

SR 3.8.1.12

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 SR demonstrates the DG load response characteristics. This SR is accomplished by 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.

Consistent with References 10, 3, and 4, 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 24 month Frequency is consistent with the Reference 2, Chapter 8.

SR 3.8.1.13

This SR demonstrates that DG non-critical protective functions are bypassed on a required actuation signal. This SR is accomplished by verifying the bypass contact changes to the correct state which prevents actuation of the non-critical function. The non-critical protective functions are consistent with References 3 and 4, and Institute of Electrical and Electronic Engineers (IEEE)-387 and are listed in Reference 2, Chapter 8. Verifying the non-critical trips are bypassed will ensure DG operation during a required actuation. The non-critical trips are bypassed during DBAs and provide an alarm on an abnormal engine condition. A failure of the electronic governor results in the diesel generator operating in hydraulic mode. This alarm provides the operator with sufficient time to react appropriately. The DG availability to mitigate the DBA is more critical than protecting the engine against minor problems that are not immediately detrimental to emergency operation of the DG.

The 24 month Frequency is based on engineering judgment, taking into consideration unit conditions required to perform the surveillance test, and is intended to be consistent with expected fuel cycle lengths. Operating experience has shown that these components usually pass the SR when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint. This Frequency is consistent with Reference 2, Chapter 8.

SR 3.8.1.14

This SR ensures that the manual synchronization and load transfer from the DG to the offsite source can be made and that the DG can be returned to ready-to-load status when offsite power is restored. 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 auto-close signal on bus undervoltage, and the load sequence timers are reset.

The Frequency of 24 months takes into consideration unit conditions required to perform the surveillance test.

SR 3.8.1.15

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 SR demonstrates the DG operation during a loss of offsite power actuation test signal in conjunction with an ESF (i.e., safety injection) 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.

It is not necessary to energize loads which are dependent on temperature to load (i.e., heat tracing, switchgear HVAC compressor, computer room HVAC compressor). Also, it is acceptable to transfer the instrument AC bus to the non tested train to maintain safe operation of the plant during testing. Loads (both permanent and auto connect) < 15 kW do not require loading onto the diesel since these are insignificant loads for the DG.

The Frequency of 24 months takes into consideration unit conditions required to perform the surveillance test and is intended to be consistent with an expected fuel cycle length of 24 months.

This SR is modified by a Note. The reason for the Note is to minimize mechanical wear and stress 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.

SR 3.8.1.16

This SR lists the SRs that are applicable to the LCO 3.8.1.c (SRs 3.8.1.1, 3.8.1.2, 3.8.1.3, 3.8.1.5, 3.8.1.6, and 3.8.1.7). Performance of any SR for the LCO 3.8.1.c will satisfy both Unit 1 and Unit 2 requirements for those SRs. Surveillance Requirements 3.8.1.4, 3.8.1.8, 3.8.1.9, 3.8.1.10, 3.8.1.11, 3.8.1.12, 3.8.1.13, 3.8.1.14, and 3.8.1.15, are not required to be performed for the LCO 3.8.1.c. Surveillance Requirement 3.8.1.10 is not required because this SR verifies manual transfer of AC power sources from the normal offsite circuit to the alternate offsite circuit, but only one qualified offsite circuit is necessary for the LCO 3.8.1.c. Surveillance Requirements 3.8.1.4, 3.8.1.11, and 3.1.8.12 are not required because they are tests that deal with loads. Surveillance Requirement 3.8.1.8 verifies the interval between sequenced loads. Surveillance Requirement 3.8.1.14 verifies the proper sequencing with offsite power. Surveillance Requirement 3.8.1.9 verifies that the DG starts within 10 seconds. These SRs are not required because they do not support the function of the LCO 3.8.1.c to provide power to the CREVS, CRETS, and H_2 Analyzer. Surveillance Requirements 3.8.1.13 and 3.8.1.15 are not required to be performed because these SRs verify the emergency loads are actuated on an ESFAS signal for the Unit in which the test

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is being performed. The LCO 3.8.1.c DG will not start on an ESFAS signal for this Unit.

REFERENCES

1. 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants"
2. UFSAR
3. Regulatory Guide 1.9, Revision 3, "Selection, Design, Qualification, and Testing of Emergency Diesel Generator Units Used as Class 1E Onsite Electric Power Systems at Nuclear Power Plants," July 1993
4. Safety Guide 9, Revision 0, March 1971
5. NRC Safety Evaluation for Amendment Nos. 19 and 5 for Calvert Cliffs Nuclear Power Plant Unit Nos. 1 and 2, dated January 14, 1977
6. Regulatory Guide 1.93, Revision 0, "Availability of Electric Power Sources," December 1974
7. Generic Letter 84-15, Proposed Staff Actions to Improve and Maintain Diesel Generator Reliability, July 2, 1984
8. Regulatory Guide 1.137, Revision 1, "Fuel-Oil Systems for Standby Diesel Generators," October 1979
9. Letter from Mr. D. G. McDonald, Jr. (NRC) to Mr. C. H. Cruse (BGE), dated April 2, 1996, Issuance of Amendments for Calvert Cliffs Nuclear Power Plant, Unit 1 (TAC No. M94030) and Unit 2 (TAC No. M94031)
10. IEEE Standard 308-1991, "IEEE Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations"

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.2 AC Sources-Shutdown

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BACKGROUND	A description of the AC sources is provided in the Bases for LCO 3.8.1.

APPLICABLE SAFETY ANALYSES

The OPERABILITY of the minimum AC sources during MODEs 5 and 6 and during movement of irradiated fuel assemblies ensures that:

a. The unit can be maintained in the shutdown or refueling condition for extended periods;

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and

c. Adequate AC electrical power is provided to mitigate events postulated during shutdown, such as a fuel handling accident.

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

During MODEs 1, 2, 3, and 4, various deviations from the analysis assumptions and design requirements are allowed within the Required Actions. This allowance is in recognition that certain testing and maintenance activities must be conducted provided an acceptable level of risk is not exceeded. During MODEs 5 and 6, performance of a significant number of required testing and maintenance activities is also required. In MODEs 5 and 6, the activities are generally planned and administratively controlled. Relaxations from MODEs 1, 2, 3, and 4 LCO requirements are acceptable during shutdown MODEs based on:

a. The fact that time in an outage is limited. This is a risk prudent goal as well as a utility economic consideration.

b. Requiring appropriate compensatory measures for certain conditions. These may include administrative controls, reliance on systems that do not necessarily meet typical design requirements applied to systems credited in operating MODE analyses, or both.

c. Prudent utility consideration of the risk associated with multiple activities that could affect multiple systems.

d. Maintaining, to the extent practical, the ability to perform required functions (even if not meeting MODEs 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 DG power.

The AC sources satisfy 10 CFR 50.36(c)(2)(ii), Criterion 3.

LCO

One offsite circuit capable of supplying the onsite Class 1E power distribution subsystem(s) of LCO 3.8.10, ensures that all required loads are powered from offsite power. An OPERABLE DG, associated with a distribution system train required to be OPERABLE by LCO 3.8.10, ensures a diverse power source is available to provide electrical power support, assuming a loss of the offsite circuit. Together, OPERABILITY of the required offsite circuit and DG ensures the availability of sufficient AC sources to operate the unit in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents).

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

The DG must be capable of starting, accelerating to rated speed and voltage, connecting to its respective ESF bus, and accepting required loads. The DG must continue to operate until offsite power can be restored to the ESF buses. These capabilities are required to be met from a variety of initial conditions such as DG in standby with the engine hot and DG in standby at ambient conditions.

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

The CREVS and CRETS are shared systems with one train of each system connected to an onsite Class 1E AC electrical power distribution subsystem from each unit. Limiting Condition for Operation 3.8.2.c requires one qualified circuit between the offsite transmission network and the other unit's onsite Class 1E AC electrical power distribution subsystems needed to supply power to the CREVS and CRETS to be OPERABLE. Limiting Condition for Operation 3.8.2.d requires one DG from the other unit capable of supplying power to the required CREVS and CRETS to be OPERABLE, if the DG required by LCO 3.8.2.b is not capable of supplying power to the required CREVS and CRETS. These requirements, in conjunction with the requirements for the unit AC electrical power sources in LCO 3.8.2.a and LCO 3.8.2.b, ensure that offsite power is available to both trains and onsite power is available to one train of the CREVS and CRETS, when they are required to be OPERABLE by their respective LCOs (LCOs 3.7.8 and 3.7.9).

BASES

APPLICABILITY

The AC sources required to be OPERABLE in MODEs 5 and 6 and during movement of irradiated fuel assemblies provide assurance that:

a. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel assemblies;

b. Systems needed to mitigate a fuel handling accident are available;

c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

The AC power requirements for MODEs 1, 2, 3, and 4 are covered in LCO 3.8.1.

ACTIONS

Limiting Condition for Operation 3.0.3 is not applicable while in MODEs 5 or 6. However, since irradiated fuel assembly movement can occur in MODEs 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 MODEs 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODEs 1, 2, 3, or 4, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of irradiated fuel assemblies would not be sufficient reason to require a reactor shutdown.

The ACTIONS have been modified by a second Note stating that performance of Required Actions shall not preclude completion of actions to establish a safe conservative position. This clarification is provided to avoid stopping movement of irradiated fuel assemblies while in a non-conservative position based on compliance with the Required Actions.

<u>A.1</u>

An offsite circuit would be considered inoperable, if it was unavailable to one required ESF train. Although two trains may be required by LCO 3.8.10, the remaining train with offsite power available may be capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS and fuel movement. By the allowance of the option to declare required features inoperable, with no offsite power available, appropriate restrictions will be implemented in accordance with the affected required features LCO's ACTIONS.

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

With the offsite circuit not available to all required trains, the option would still exist to declare all required features inoperable. Since this option may involve undesired administrative efforts, the allowance for sufficiently conservative actions is made. With the required DG inoperable, the minimum required diversity of AC power sources is not available. It is, therefore, required to suspend CORE ALTERATIONS, movement of irradiated fuel assemblies, and operations involving positive reactivity additions. The Required Action to suspend positive reactivity additions does not preclude actions to maintain or increase reactor vessel inventory provided the required SHUTDOWN MARGIN (SDM) is maintained.

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

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

Pursuant to LCO 3.0.6, the Electrical Distribution System's ACTIONS are not entered even if all AC sources to it are inoperable, resulting in de-energization. Therefore, the

BASES

Required Actions of Condition A are modified by a Note to indicate that when Condition A is entered with no AC power to any required ESF bus, the ACTIONS for LCO 3.8.10 must be immediately entered. This Note allows Condition A to provide requirements for the loss of the offsite circuit, whether or not a train is de-energized. Limiting Condition for Operation 3.8.10 provides the appropriate restrictions for the situation involving a de-energized train.

SURVEILLANCE REQUIREMENTS

SR 3.8.2.1 and SR 3.8.2.2

Surveillance Requirements 3.8.2.1 and 3.8.2.2 require the performance of 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. Surveillance Requirement 3.8.1.10 is not required to be met, since only one offsite circuit is required to be OPERABLE. Surveillance Requirements 3.8.1.4, 3.8.1.8, 3.8.1.13, and 3.8.1.15 are related to automatic starting of the DGs for an operating unit, which is not applicable for a shutdown unit. Surveillance Requirement 3.8.1.16 is related to LCO 3.8.2.c and 3.8.2.d AC sources, and is addressed by SR 3.8.2.2.

Surveillance Requirement 3.8.2.1 is modified by a Note. The Note lists SRs not required to be performed in order to preclude de-energizing a required 4.16 kV ESF bus or disconnecting a required offsite circuit during performance of SRs. With limited AC Sources available, a single event could compromise both the required circuit and the DG. It is the intent that these SRs must still be capable of being met, but actual performance is not required during periods when the DG and offsite circuit are required to be OPERABLE. Refer to the corresponding Bases for LCO 3.8.1 for a discussion of each SR.

REFERENCES

None

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.3 Diesel Fuel Oil

BASES

BACKGROUND

The fuel oil storage tanks (FOSTs) contain sufficient capability for the DGs to operate one unit on accident loads and one unit on shutdown loads for seven days. This is discussed in Reference 1, Chapter 8. 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 the storage tanks to the day tank by transfer pumps associated with each DG.

For proper operation of the standby DGs, it is necessary to ensure the proper quality of the fuel oil. Testing to check for water and sediment content, the kinematic viscosity, specific gravity (or API gravity), and impurity level (i.e., total particulates) ensures this quality.

The DG fuel oil system design at Calvert Cliffs supports four emergency DGs and other non-safety DGs. Three of the four emergency DGs, i.e., Nos. 1B, 2A, and 2B, are fueled from two FOSTs, i.e., FOST Nos. 11 and 21, and DG No. 1A is fueled from FOST No. 1A. Fuel Oil Storage Tank Nos. 1A and 21 are enclosed such as to be considered "tornado protected" but FOST No. 11 is not protected. As such, FOST No. 11 is not used as the primary source for the emergency DGs, but rather is used as a backup to support FOST No. 21, if it or the fuel oil it contains becomes degraded.

The operability of FOST Nos. 21 and 11 ensure that at least seven days of fuel oil will be reserved below the internal tank standpipes for operation of one DG on each unit, assuming one unit under accident conditions with a DG load of 3500 kW, and the opposite unit under normal shutdown conditions with a DG load of 3000 kW. Additionally, the operability of FOST No. 21 ensures that in the event of a loss of offsite power, concurrent with a loss of the non-bunkered FOST (tornado/missile event), at least seven days of fuel oil will be available for operation of one DG on each unit, assuming both DGs are loaded to 3000 kW. The operability of the FOST No. 1A ensures that at least seven days of fuel oil is available to support operation of DG No. 1A at 4000 kW.

The operability of the fuel oil day tanks ensures that at least one hour of DG operation is available without makeup to the day tanks, assuming DG No. 1A is loaded to 4000 kW and DG Nos. 1B, 2A, and 2B are loaded to 3500 kW.

APPLICABLE SAFETY ANALYSES

The initial conditions of DBA and transient analyses in Reference 1, and Chapters 6 and 14, assume 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, RCS, and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for LCO Section 3.2, 3.4, and 3.6.

Since diesel fuel oil supports the operation of the standby AC power sources, they satisfy 10 CFR 50.36(c)(2)(ii), Criterion 3.

LCO

Fuel Oil Storage Tank No. 1A is required to contain a minimum of 49,500 gallons of available diesel fuel oil which is a sufficient supply to operate DG No. 1A with accident loads for seven days. Fuel Oil Storage Tank No. 21 is required to contain a minimum of 85,000 gallons of available diesel fuel oil which is a sufficient supply to operate one unit with accident loads and one unit with shutdown loads for seven days. It is also required to meet specific standards for quality. This requirement, in conjunction with an ability to obtain replacement supplies within seven days, supports the availability of DGs required to shut down the reactor and to maintain it in a safe condition for an AOO or a postulated DBA with loss of offsite power. Diesel generator day tank fuel requirements, as well as transfer capability from the FOST to the day tank, are addressed in LCO 3.8.1 and LCO 3.8.2.

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 supports LCO 3.8.1 and LCO 3.8.2, stored diesel fuel oil is required to be within limits when the associated DG is required to be OPERABLE.

For both Unit 1 and Unit 2, the FOST No. 1A associated DG is only DG No. 1A. For Unit 1, the FOST No. 21 associated DGs are DG Nos. 1B and 2B. For Unit 2, the FOST No. 21 associated DGs are DG Nos. 2A and 2B. Alignment does not affect the association of DG and FOST since the individual DG fuel oil day tank provides sufficient volume for the DG to perform its safety function while re-alignment is accomplished, if necessary.

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

In this Condition, the seven day fuel oil supply for a DG is not available. However, fuel oil volume reduction is limited to 6/7 of the required volume which will provide sufficient capacity to operate one DG on one unit on accident loads, and one DG on the other unit on shutdown loads for approximately six days. 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 (approximately six days), the fact that procedures will be initiated to obtain replenishment, and the low probability of an event during this brief period.

Condition A addresses only FOST No. 1A which is "tornado protected" and which contains sufficient fuel for seven days of required operation of DG No. 1A. It supports both Unit 1 and Unit 2 equipment since DG No. 1A provides power for equipment which is shared by both units, e.g., the CREVS.

Condition B addresses only FOST No. 21 which is "tornado protected" and which contains sufficient fuel for seven days of required operation of two DGs. Fuel Oil Storage Tank No. 21 supports both Unit 1 and Unit 2 equipment, but Condition B is written for Unit 1 only to reflect the Unit 1 requirements for DG Nos. 1B and 2B. For an accident, Unit 1 requires either DG No. 1A or both DG Nos. 1B and 2B (since DG No. 2B powers equipment which is redundant to some equipment powered by DG No. 1A, e.g., CREVS). Since DG No. 1A is supported by FOST No. 1A and the redundant required equipment is powered by DG Nos. 1B and 2B which are supported by FOST No. 21, at least one full train of required equipment is supported by a "tornado protected" FOST even with an inoperable FOST or DG. Therefore, low fuel oil volume in FOST No. 21 can be supplemented by the fuel oil volume of an OPERABLE FOST No. 11 to assure the necessary volume. Required Action B.1 requires the combined volume of FOST No. 21 and an OPERABLE FOST No. 11 to be verified to be greater than 6/7 of the required volume within one hour. The Completion Time of one hour is consistent with the time needed to verify through administrative means that the backup FOST is OPERABLE. Required Action B.2 requires the combined volume of FOST No. 21 and an OPERABLE FOST No. 11 to be ≥ 85,000 gallons within 48 hours. In addition, if FOST No. 21 is not restored and FOST No. 11 continues to be relied upon, Required Action B.2 must be repeated every 31 days. This effectively replaces the SR 3.8.3.1 periodic surveillance of available DG fuel oil volume for the inoperable FOST No. 21. Since FOST No. 11 is not required by the LCO, FOST No. 11 may be considered OPERABLE only when the stored fuel oil meets SR 3.8.3.2 and SR 3.8.3.3, and is capable of being delivered to the required DG, i.e., the necessary piping and valves are capable of performing their safety function.

Specific alignment to a particular FOST is not required since the individual DG fuel oil day tank provides sufficient volume for the DG to perform its safety function while re-alignment is accomplished, if necessary. Further, if any fuel oil in FOST No. 11 above the 33,000 gallons reserved for emergency DG use is credited for DG use, appropriate administrative controls must be in place to assure its retention for this purpose.

Condition C also addresses only FOST No. 21 which is "tornado protected" and which contains sufficient fuel for seven days of required operation of two DGs. Fuel Oil Storage Tank No. 21 supports both Unit 1 and Unit 2 equipment, but Condition C is written for Unit 2 only to reflect the Unit 2 requirements for DG Nos. 2A and 2B. For an accident, Unit 2 requires either DG No. 2B or both DG Nos. 1A and 2A (since DG No. 1A powers equipment which is redundant to some equipment powered by DG No. 2B, e.g., CREVS). Unlike Unit 1, at least one full train of required equipment is not supported by a "tornado protected" FOST with an inoperable FOST or DG since most of the redundant required equipment is powered by DG Nos. 2A and 2B which are both supported by FOST No. 21. Therefore, low fuel oil volume in FOST No. 21 can only be supplemented by the fuel oil volume of an OPERABLE FOST No. 11 to assure the necessary volume when the probability for a tornado is sufficiently low. This is reflected in Note 2 for Required Action C.2 which addresses the inoperability of FOST No. 21 from April 1 to September 30. During the time of low tornado probability, the Unit 2 requirements for the inoperability of FOST No. 21 are very similar to the Unit 1 requirements for inoperability of FOST No. 21. It is acceptable for the combined volume of FOST No. 11 and FOST No. 21 to be considered in providing 6/7 of the required volume for the 48 hours allowed by Required Action C.3. Required Action C.1 requires the combined volume of FOST No. 21 and an OPERABLE FOST No. 11 to be verified to be greater than 6/7 of the required volume within one hour. Required Action C.3 then requires the volume of FOST No. 21 to be restored to within volume limits within 48 hours. However, during tornado season, i.e., from April 1 to September 30, the fuel oil volume of FOST No. 11 is not allowed to be credited and the fuel oil seven day volume of

BASES

FOST No. 21 must be restored within two hours as indicated in Required Action C.2. Required Action C.2 is also modified by a Note such that it is only required during the operation of Unit 2 in MODEs 1, 2, 3, or 4 since the unit is already shutdown if it is in another MODE or condition. An OPERABLE FOST No. 11 is determined as described above in the discussion for Condition B.

D.1

This Condition is entered as a result of a failure to meet the acceptance criterion of SR 3.8.3.2. Normally, trending of particulate levels allows sufficient time to correct high particulate levels prior to reaching the limit of acceptability. Poor sample procedures (bottom sampling), contaminated sampling equipment, and errors in laboratory analysis can produce failures that do not follow a trend. Since the presence of particulates does not mean failure of the fuel oil to burn properly in the diesel engine, and particulate concentration is unlikely to change significantly between SR 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 seven day Completion Time allows for further evaluation, resampling, and re-analysis of the DG fuel oil.

E.1

With the new fuel oil properties defined in the Bases for SR 3.8.3.2 not within the required limits, a period of 30 days is allowed for restoring the stored fuel oil properties to within the new fuel oil limits. This period provides sufficient time to test the stored fuel oil to determine that the new fuel oil, when mixed with previously stored fuel oil, remains acceptable, or restore the stored fuel oil properties to within the new fuel oil limits. 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.

F.1

With a Required Action and associated Completion Time not met, or one or more DGs with diesel fuel oil 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. "Associated DG(s)" are identified in the Applicability Bases.

SURVEILLANCE REQUIREMENTS

SR 3.8.3.1

This SR provides verification that there is an adequate inventory of fuel oil in the DG FOSTs to support one unit on accident loads and one unit on shutdown loads for seven days. The seven day period is sufficient time to place the unit in a safe shutdown condition and to bring in replenishment fuel from an offsite location.

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

SR 3.8.3.2

The tests listed below are a means of determining whether new fuel oil is of the appropriate grade (i.e., 2D and 2D low sulfur) and has not been contaminated with substances that would have an immediate, detrimental impact on diesel engine combustion. Note that further references to American Society for Testing Materials (ASTM) 2D fuel oil include both 2D and 2D low sulfur. If results from these tests are within acceptable limits, the fuel oil may be added to the storage tanks without concern for contaminating the entire volume of fuel oil in the storage tanks. These tests are to be conducted prior to adding the new fuel to the storage tank(s), but in no case is the time between receipt of new fuel and conducting the tests to exceed 31 days. The tests, limits, and applicable ASTM Standards are as follows:

a. Sample the new fuel oil in accordance with Reference 3, ASTM D4057-1995;, .

b. Verify in accordance with the tests specified in Reference 3, ASTM D975-1996, that the sample has an absolute specific gravity at 60/60°F of ≥ 0.8155 and ≤ 0.8871, or an American Petroleum Institute gravity at 60°F of ≥ 28° and ≤ 42°, a kinematic viscosity at 40°C of ≥ 1.9 centistokes and ≤ 4.1 centistokes, and a flash point ≥ 125°F; and

c. Verify that the new fuel oil has ≤ 0.05% water and sediment (Reference 3, ASTM D975-1996).

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

Within 31 days following the initial new fuel oil sample, the fuel oil is analyzed to establish that the other properties specified in Reference 2, ASTM D975-1996, Table 1, are met for new fuel oil. 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 SR ensures the availability of high quality fuel oil for the DGs.

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

Particulate concentrations should be determined by gravimetric analysis (based on ASTM D2276-1989) 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. Because the total stored fuel oil volume for DG Nos. 1B, 2A, and 2B is contained in two interconnected tanks, each tank must be considered and tested separately. There is a separate FOST for DG No. 1A.

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 3.8.3.3

Microbiological fouling is a major cause of fuel oil degradation. There are numerous bacteria that can grow in fuel oil and cause fouling, but all must have a water environment in order to survive. Removal of water from the fuel storage tanks once every 92 days eliminates the necessary environment for bacterial survival. This is the most effective means of controlling microbiological fouling. In addition, it eliminates the potential for water entrainment in the fuel oil during DG operation. Water may come from any of several sources, including condensation, ground water, rain water, and contaminated fuel oil, and from breakdown of the fuel oil by bacteria. Frequent checking for and removal of accumulated water minimizes fouling and provides data regarding the watertight integrity of the fuel oil system. The SR Frequencies are established by Reference 3. This SR is for preventative maintenance. The presence of water does not necessarily represent failure of this SR provided the accumulated water is removed during performance of the surveillance test.

REFERENCES

1. UFSAR
2. ASTM Standards
3. Regulatory Guide 1.137, "Fuel-Oil Systems for Standby Diesel Generators," January 1978

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.4 DC Sources-Operating

BASES

BACKGROUND

The station DC sources provide the AC emergency power system with control power. It also provides both motive and control power to selected safety related equipment and preferred AC vital bus power (via inverters). As required by Reference 1, Appendix 1C, Criterion 39, the DC electrical power sources are designed to have sufficient independence, redundancy, and testability to perform their safety functions, assuming a single failure. The DC sources also conform to the recommendations of References 2 and 3.

The 125 VDC electrical power sources consist of four independent and redundant safety related Class 1E DC channels. Each channel consists of one 125 VDC battery, the associated battery charger for each battery, and all the associated control equipment and interconnecting cabling.

During normal operation, the 125 VDC load is powered from the battery chargers with the batteries floating on the system. In cases where momentary loads are greater than the charger capability, or a loss of normal power to the battery charger, the DC load is automatically powered from the station batteries.

The DC channels provide the control power for its associated Class 1E AC power load group, 4.16 kV switchgear, and 480 V load centers. The DC channels also provide a DC source to the inverters, which in turn power the AC vital buses.

The DC sources are described in more detail in the Bases for LCO 3.8.9 and for LCO 3.8.10.

Each battery has adequate storage capacity to carry the required load continuously for at least 2 hours and to carry load duty cycle as discussed in Reference 1, Chapter 8.

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

The batteries for DC channels are sized to produce required capacity at 80% of nameplate rating, corresponding to warranted capacity at end of life cycles and the 100% design demand. Battery size is based on 125% of required capacity. An average voltage of 2.13 V per cell, corresponds to a total minimum voltage output of 125 V per battery (128 V for the reserve battery) as discussed in Reference 1, Chapter 8. The criteria for sizing large lead storage batteries are defined in Reference 4.

Each DC channel has ample power output capacity for the steady state operation of connected loads required during normal operation, while at the same time maintaining its battery bank fully charged. Each battery charger also has sufficient capacity to restore the battery from the design minimum charge to 95% of its fully charged state within 24 hours while supplying normal steady state loads discussed in Reference 1, Chapter 8.

APPLICABLE SAFETY ANALYSES

The initial conditions of DBA and transient analyses in Reference 1, Chapters 6 and 14, assume that ESF systems are OPERABLE. The DC channels provide a normal and emergency DC sources 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 10 CFR 50.36(c)(2)(ii), Criterion 3.

LCO

The DC channels, each channel consisting of one battery, one battery charger, and the corresponding control equipment and interconnecting cabling supplying power to the associated bus, 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 AOO or a postulated DBA. Loss of any DC channel does not prevent the minimum safety function from being performed (Reference 1, Chapter 8).

An OPERABLE DC channel requires the battery and one OPERABLE charger to be operating and connected to the associated DC bus(es).

A battery charger is considered OPERABLE as long as it is receiving power from its normal offsite source and can be connected to a DG within 2 hours following an event.

APPLICABILITY

The DC 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 sources requirement for MODEs 5 and 6 are addressed in the Bases for LCO 3.8.5.

ACTIONS

A.1

Required Action A.1 requires the inoperable battery to be replaced by the reserve battery within four hours when one DC channel is inoperable due to an inoperable battery and the reserve battery is available. The reserve battery is a qualified battery that can replace and perform the required function of any inoperable battery. The four hour Completion Time is acceptable based on the capability of the reserve battery and the time it takes to replace the inoperable battery with the reserve battery while minimizing the time in this degraded condition.

B.1

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

If one of the required DC channels is inoperable for reasons other than Condition A (e.g., inoperable battery, inoperable battery charger(s), or inoperable battery charger and associated inoperable battery), the remaining DC channels have the capacity to support a safe shutdown and to mitigate an accident condition. Since a subsequent worst case single failure would, however, result in the further loss of the 125 VDC channels with attendant loss of ESF functions, continued power operation should not exceed 2 hours. The 2 hour Completion Time is based on Reference 5 and reflects a reasonable time to assess unit status as a function of the inoperable DC channel and, if the DC channel is not restored to OPERABLE status, to prepare to effect an orderly and safe unit shutdown.

C.1 and C.2

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

SURVEILLANCE REQUIREMENTS

SR 3.8.4.1

Verifying battery terminal voltage while on float charge helps to ensure the effectiveness of the charging system and the ability of the batteries to perform their intended function. Float charge is the condition in which the charger is supplying connected loads and the continuous charge required to overcome the internal losses of a battery and maintain the battery in a fully charged state. The voltage requirements are based on the nominal design voltage of the battery (2.13 V per cell average) and are consistent with Reference 6 and the initial state of charge conditions assumed in the battery sizing calculations. The 7 day Frequency is conservative when compared with manufacturer recommendations and Reference 6.

SR 3.8.4.2

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

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

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

SR 3.8.4.3

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

The presence of physical damage or deterioration does not necessarily represent a failure of this SR, provided an evaluation determines that the physical damage or deterioration does not affect the OPERABILITY of the battery (its ability to perform its design function).

The 18 month Frequency is based on engineering judgment. Operating experience has shown that these components usually pass the SR when performed at the 18 month Frequency.

Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.8.4.4 and SR 3.8.4.5

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

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

The 18 month Frequency for these SRs is based on engineering judgment. Operating experience has shown that these components usually pass the SRs when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.8.4.6

This SR requires that each battery charger be capable of supplying 400 amps and 125 V for $\geq$ 30 minutes. These requirements are based on the output rating of the chargers (Reference 1, Chapter 8). According to Reference 7, the battery charger supply is required to be based on the largest combined demands of the various steady state loads and the charging capacity to restore the battery from the design minimum charge state to the fully charged state, irrespective of the status of the unit during these demand occurrences. The minimum required amperes and duration ensures that these requirements can be satisfied. The test is performed while supplying normal DC loads or an equivalent or greater dummy load.

The SR Frequency is acceptable, given the unit conditions required to perform the test and the other administrative controls existing to ensure adequate charger performance during these 24 month intervals. In addition, this Frequency is intended to be consistent with expected fuel cycle lengths.

SR 3.8.4.7

A battery service test is a special test of battery capability, as found and with the associated battery charger disconnected, to satisfy the design requirements (battery duty cycle) of the DC source. The test duration must be ≥ 2 hours and battery terminal voltage must be maintained ≥ 105 volts during the test. The discharge rate and test length should correspond to the design accident load (duty) cycle requirements as specified in Reference 1, Chapter 8. A dummy load simulating the emergency loads of the design duty cycle may be used in lieu of the actual emergency loads.

The SR Frequency of 24 months is consistent with expected fuel cycle lengths.

This SR is modified by a Note. The Note allows the performance of a modified performance discharge test in lieu of a service test. This substitution is acceptable because a modified performance discharge test represents a more severe test of battery capacity than SR 3.8.4.7.

SR 3.8.4.8

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

A battery modified performance discharge test is a simulated duty cycle consisting of just two rates; the one minute rate published for the battery or the largest current load of the duty cycle, followed by the test rate employed for the performance discharge test, both of which envelope the duty

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cycle of the service test. Since the ampere-hours removed by a rated one minute discharge represents a very small portion of the battery capacity, the test rate can be changed to that for the performance test without compromising the results of the performance discharge test. The battery terminal voltage for the modified performance discharge test should remain above the minimum battery terminal voltage specified in the battery performance discharge test for the duration of time equal to that of the performance discharge test.

A modified performance 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. Either the battery performance discharge test or the modified performance discharge test is acceptable for satisfying SR 3.8.4.8; however, only the modified performance discharge test may be used to satisfy SR 3.8.4.8 while satisfying the requirements of SR 3.8.4.7 at the same time.

The acceptance criteria for this SR are consistent with References 6 and 4. These references recommend that the battery be replaced if its capacity is below 80% of the manufacturer rating. A capacity of 80% shows that the battery rate of deterioration is increasing, even if there is ample capacity to meet the load requirements.

The SR Frequency for this test is normally 60 months. If the battery shows degradation, or if the battery has reached 85% of its expected life and capacity is < 100% of the manufacturer's rating, the SR Frequency is reduced to 12 months. However, if the battery shows no degradation but has reached 85% of its expected life, the SR Frequency is only reduced to 24 months for batteries that retain capacity ≥ 100% of the manufacturer's rating. Degradation is indicated, according to Reference 6, when the battery capacity drops by more than 10% relative to its capacity on

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the previous performance test or when it is ≥ 10% below the manufacturer's rating. These Frequencies are consistent with the recommendations in Reference 6.

REFERENCES

1. UFSAR
2. Safety Guide 6, Revision 0, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors," March 1971
3. IEEE Standard 308-1978, "IEEE Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations"
4. IEEE Standard 485-1983, "Recommended Practice for Sizing Large Lead Storage Batteries for Generating Stations and Substations (ANSI)," June 1983
5. Regulatory Guide 1.93, "Availability of Electric Power Sources," December 1974
6. IEEE Standard 450-1995, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications," May 1995
7. Regulatory Guide 1.32, Revision 2, "Criteria for Safety-Related Electric Power Systems for Nuclear Power Plants," February 1977

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.5 DC Sources-Shutdown

BASES

BACKGROUND

A description of the DC sources is provided in the Bases for LCO 3.8.4.

APPLICABLE SAFETY ANALYSES

The initial conditions of DBA and transient analyses in Reference 1, Chapters 6 and 14, assume that ESF systems are OPERABLE. The DC sources provide normal and emergency DC for the DGs, emergency auxiliaries, and control and switching during all MODEs of operation.

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

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

a. The unit can be maintained in the shutdown or refueling condition for extended periods;

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and

c. Adequate DC sources are provided to mitigate events postulated during shutdown, such as a fuel handling accident.

The DC sources satisfy 10 CFR 50.36(c)(2)(ii), Criterion 3.

LCO

The DC channels, each channel consisting of one battery, one battery charger, and the corresponding control equipment and interconnecting cabling within the channel, are required to be OPERABLE to support required trains of distribution systems required OPERABLE by LCO 3.8.10. This ensures the availability of sufficient DC sources to operate the unit in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents).

APPLICABILITY

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

a. Required features needed to mitigate a fuel handling accident are available;

b. Required features necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

c. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

The DC channel requirements for MODEs 1, 2, 3, and 4 are covered in LCO 3.8.4.

ACTIONS

Limiting Condition for Operation 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. Therefore, in either case, the inability to suspend movement of irradiated fuel assemblies would not be sufficient reason to require a reactor shutdown.

The ACTIONS have been modified by a second Note stating that performance of REQUIRED ACTIONS shall not preclude completion of actions to establish a safe conservative position. This clarification is provided to avoid stopping movement of irradiated fuel assemblies while in a non-conservative position based on compliance with the REQUIRED ACTIONS.

<u>A.1, A.2.1, A.2.2, A.2.3, and A.2.4</u>

If two trains are required per LCO 3.8.10, the remaining train with DC power available may be capable of supporting sufficient systems to allow continuation of CORE ALTERATIONS and fuel movement. By allowing the option to declare required features inoperable with the associated DC power source(s) inoperable, appropriate restrictions will be implemented in accordance with the affected required features LCO ACTIONS. In many instances, this option may involve undesired administrative efforts. Therefore, the allowance for sufficiently conservative actions is made (i.e., to suspend CORE ALTERATIONS, movement of irradiated fuel assemblies and operations involving positive reactivity additions). The Required Action to suspend positive reactivity additions does not preclude actions to maintain or increase reactor vessel inventory, provided the required SDM is maintained.

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

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

SURVEILLANCE REQUIREMENTS

SR 3.8.5.1

Surveillance Requirement 3.8.5.1 states that surveillance tests required by SR 3.8.4.1 through SR 3.8.4.8 are applicable in these MODEs. See the corresponding Bases for LCO 3.8.4 for a discussion of each SR.

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

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REFERENCES	1. UFSAR

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.6 Battery Cell Parameters

BASES

BACKGROUND	This LCO delineates the limits on electrolyte temperature, level, individual cell float voltage (ICV), and specific gravity for the DC power source batteries. A discussion of these batteries and their OPERABILITY requirements is provided in the Bases for LCO 3.8.4 and LCO 3.8.5.
APPLICABLE SAFETY ANALYSES	The initial conditions of DBA and transient analyses in Reference 1, Chapters 6 and 14, assume ESF systems are OPERABLE. The DC sources provide 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 channels is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the unit, as discussed in the Bases for LCO 3.8.4 and LCO 3.8.5. Battery cell parameters satisfy Criterion 3 of the NRC Policy Statement.
LCO	Battery cell parameters must remain within acceptable limits to ensure availability of the required DC power to shut down the reactor and maintain it in a safe condition after an AOO or a postulated DBA. Electrolyte limits are conservatively established, allowing continued DC electrical system function even with Category A or B limits not met.
APPLICABILITY	The battery parameters are required solely for the support of the associated DC electrical power subsystems. Therefore, battery electrolyte is only required when the DC power source is required to be OPERABLE. Refer to the Applicability discussion in the Bases for LCO 3.8.4 and LCO 3.8.5.
ACTIONS	The Actions Table is modified by a Note which indicates that separate Condition entry is allowed for each battery. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable DC channel. Complying with the Required Actions for one

inoperable DC channel may allow for continued operation, and subsequent inoperable DC subsystem(s) are governed by separate Condition entry and application of associated Required Actions.

A.1, A.2.1, A.2.2, and A.3

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

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

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

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

B.1

With one or more batteries with one or more battery cell parameters outside the Category C limit for any connected cell, sufficient capacity to supply the maximum expected load requirement is not assured and the corresponding DC channel must be declared inoperable. Additionally, other potentially extreme conditions, such as any Required Action of Condition A and associated Completion Time not met, or average electrolyte temperature of representative cells < 69°F, are also cause for immediately declaring the associated DC channel inoperable.

SURVEILLANCE REQUIREMENTS

SR 3.8.6.1

This SR verifies that Category A battery cell parameters are consistent with Reference 2, which recommends regular battery inspections (at least one per month) including voltage, specific gravity, and electrolyte temperature of pilot cells.

SR 3.8.6.2

The quarterly inspection of specific gravity and voltage is consistent with Reference 2.

SR 3.8.6.3

This Surveillance verification that the average temperature of representative cells is > 69°F is consistent with a recommendation of Reference 2, which states that the temperature of electrolytes in representative cells should be determined on a quarterly basis. The temperature is also high enough to supply the required capacity.

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

Table 3.8.6-1

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

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

The Category A limits specified for electrolyte level are based on manufacturer recommendations and are consistent with the guidance in Reference 2, with the extra 1/4 inch allowance above the high water level indication for operating margin to account for temperatures and charge effects. In addition to this allowance, Footnote (a) to Table 3.8.6-1 permits the electrolyte level to be temporarily above the specified maximum level during and following equalizing charge (i.e., for up to seven days following the completion of an equalize charge), provided it is not overflowing. These limits ensure that the plates suffer no physical damage, and that adequate electron transfer capability is maintained in the event of transient conditions. Reference 2 recommends that electrolyte level readings should be made only after the battery has been at float charge for at least 72 hours.

The Category A limit specified for ICV is ≥ 2.13 V per cell. This value is based on a recommendation of Reference 2, which states that prolonged operation of cells < 2.13 V can reduce the life expectancy of cells.

The Category A limit specified for specific gravity for each pilot cell is ≥ 1.200 (0.015 below the manufacturer fully charged nominal specific gravity or a battery charging current that had stabilized at a low value). This value is characteristic of a charged cell with adequate capacity. According to Reference 2, the specific gravity readings are based on a temperature of 77°F (25°C) and full electrolyte level.

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

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

The Category B limits specified for electrolyte level and ICV are the same as those specified for Category A and have been discussed above. The Category B limit specified for specific gravity for each connected cell is ≥ 1.195 (0.020 below the manufacturer fully charged, nominal specific gravity) with the average of all connected cells ≥ 1.205 (0.010 below the manufacturer fully charged, nominal specific gravity). These values are based on manufacturer's recommendations. The minimum specific gravity value required for each cell ensures a cell with a marginal or unacceptable specific gravity is not masked by averaging cells having higher specific gravities.

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

The Category C limit specified for electrolyte level (above the top of the plates and not overflowing) ensures that the plates suffer no physical damage and maintain adequate electron transfer capability. The Category C limit for ICV is derived from Reference 2 recommendations, which states

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that a cell voltage of 2.07 V or below, under float conditions and not caused by elevated temperature of the cell, indicates internal cell problems and may require cell replacement.

The Category C limit of average specific gravity ≥ 1.195 is based on manufacturer recommendations (0.020 below the manufacturer recommended fully charged, nominal specific gravity). In addition to that limit, it is required that the specific gravity for each connected cell must be no less than 0.020 below the average of all connected cells. This limit ensures that a cell with a marginal or unacceptable specific gravity is not masked by averaging with cells having higher specific gravities.

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

Because of specific gravity gradients that are produced during the recharging process, delays of several days may occur while waiting for the specific gravity to stabilize. A stabilized charging current is an acceptable alternative to specific gravity measurement for determining the state of charge. This phenomenon is discussed in Reference 2. Footnote (c) to Table 3.8.6-1 allows the float charge current to be used as an alternate to specific gravity for up to seven days following a battery equalizing recharge. Within seven days, each connected cell's specific gravity must be measured to confirm the state of charge. Following a minor battery recharge (such as equalizing charge that does not follow a deep discharge) specific gravity gradients are not significant, and confirming measurements may be made in less than seven days.

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REFERENCES

1. UFSAR
2. IEEE Standard 4501995, "IEEE Recommended Practice for Maintenance, Testing, and Replacement Vented Lead-Acid Batteries for Stationary Applications," May 1995

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.7 Inverters-Operating

BASES

BACKGROUND	The inverters are the preferred source of power for the AC vital buses, because of the stability and reliability they achieve. The function of the inverter is to provide AC electrical power to the vital buses. Each inverter has two built-in independent inverters, either one of which can serve as the preferred source of power. In these dual inverters, 120 volt AC power output can be manually switched from one side to the other side. The inverters can be powered from the DC bus which is energized from the station battery and/or battery chargers. The station battery and the inverters provides an uninterruptible power source for the instrumentation and controls for the Reactor Protective System (RPS) and the ESFAS. Specific details on inverters and their operating characteristics are found in Reference 1, Chapter 8.
APPLICABLE SAFETY ANALYSES	The initial conditions of DBA and transient analyses in Reference 1, Chapters 6 and 14, assume ESF 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, RCS, and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for Sections 3.2, 3.4, and 3.6. The OPERABILITY of the inverters is consistent with the initial assumptions of the accident analyses and is based on meeting the design basis of the unit. This includes maintaining required AC vital buses OPERABLE during accident conditions in the event of: a. An assumed loss of all offsite AC electrical power or all onsite AC electrical power; and b. A worst case single failure. Inverters are a part of the distribution system and, as such, satisfy 10 CFR 50.36(c)(2)(ii), Criterion 3.

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LCO

The LCO requires four inverters to be operable, one inverter per AC vital bus. Each AC vital bus can receive power from either side of the dual inverter. Each side of the dual inverter is fully rated, to power the AC vital bus. Therefore, only one side of each dual inverter is required for the inverter to be considered OPERABLE.

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 AOO or a postulated DBA.

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

OPERABLE inverters require the associated vital bus to be powered by either side of the dual inverter with output voltage within tolerances, and power input to the inverter from a 125 VDC station battery. Alternatively, power supply may be from the battery charger as long as the station battery is available as the uninterruptible power supply.

APPLICABILITY

The inverters are required to be OPERABLE in MODEs 1, 2, 3, and 4 to ensure that:

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

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

Inverter requirements for MODEs 5 and 6 are covered in the Bases for LCO 3.8.8.

ACTIONS

A.1

With a required inverter inoperable, its associated AC vital bus becomes inoperable until it is manually re-energized

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from its 120 VAC bus powered by an ESF motor control center through a regulating transformer.

Required Action A.1 is modified by a Note, which states to enter the applicable conditions and Required Actions of LCO 3.8.9, when Condition A is entered with one AC vital bus de-energized. This ensures the vital bus is re-energized within two hours.

Required Action A.1 allows 24 hours to fix the inoperable inverter and return it to service. The 24 hour limit is based upon engineering judgment, taking into consideration the time required to repair an inverter and the additional risk to which the unit is exposed because of the inverter inoperability. This has to be balanced against the risk of an immediate shutdown, along with the potential challenges to safety systems such a shutdown might entail. When the AC vital bus is powered from its constant voltage source, it is relying upon interruptible AC electrical power sources (offsite and onsite). The uninterruptible inverter source to the AC vital buses is the preferred source for powering instrumentation trip setpoint devices.

B.1 and B.2

If the inoperable devices or components cannot be restored to OPERABLE status within the required Completion Time, the 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 six hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.8.7.1

This SR verifies that the inverters are functioning properly with all required circuit breakers closed and AC vital buses energized from the inverter. The verification of proper voltage output ensures that the required power is readily available for the instrumentation of the RPS and ESFAS connected to the AC vital buses. The seven day Frequency

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takes into account the redundant capability of the inverters and other indications available in the Control Room that alert the operator to inverter malfunctions.

REFERENCES

1. UFSAR

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.8 Inverters-Shutdown

BASES

BACKGROUND	A description of the inverters is provided in the Bases for LCO 3.8.7.

APPLICABLE SAFETY ANALYSES

The initial conditions of DBA and transient analyses in Reference 1, Chapters 6 and 14, assume ESF systems are OPERABLE. The DC to AC inverters are designed to provide the required capacity, capability, redundancy, and reliability to ensure the availability of necessary power to the RPS and ESFAS instrumentation and controls so that the fuel, RCS, and containment design limits are not exceeded.

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

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

a. The unit can be maintained in the shutdown or refueling condition for extended periods;

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and

c. Adequate power is available to mitigate events postulated during shutdown, such as a fuel handling accident.

The inverters were previously identified as part of the distribution system and, as such, satisfy 10 CFR 50.36(c)(2)(ii), Criterion 3.

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 AOO or a postulated DBA. The battery powered inverters provide uninterruptible supply of AC electrical power to the AC vital buses even if the 4.16 kV safety buses are de-energized. OPERABILITY of the inverters requires that the vital bus be powered by the inverter. This ensures the availability of sufficient inverter power sources to operate the unit in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents).

APPLICABILITY

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

a. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel in the core;

b. Systems needed to mitigate a fuel handling accident are available;

c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

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

ACTIONS

Limiting Condition for Operation 3.0.3 is not applicable while in MODEs 5 or 6. However, since irradiated fuel assembly movement can occur in MODEs 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 MODEs 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODEs 1, 2, 3, or 4, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of irradiated fuel assemblies would not be sufficient reason to require a reactor shutdown.

<u>A.1, A.2.1, A.2.2, A.2.3, and A.2.4</u>

If two trains are required by LCO 3.8.10, the remaining OPERABLE inverters may be capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS, fuel movement, operations with a potential for draining the reactor vessel, and operations with a potential for positive reactivity additions. The Required Action to suspend

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

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

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

SURVEILLANCE REQUIREMENTS

SR 3.8.8.1

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

REFERENCES

1. UFSAR

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.9 Distribution Systems-Operating

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BACKGROUND

The onsite Class 1E AC, DC, and AC vital bus Electrical Power Distribution Systems are divided into two redundant and independent AC electrical power distribution subsystems and four independent and redundant DC and AC vital bus electrical power distribution subsystems (Reference 1, Chapter 8).

The AC primary Electrical Power Distribution System consists of two 4.16 kV ESF buses, each having at least one separate and independent offsite source of power as well as a dedicated onsite DG source. Each 4.16 kV ESF bus is normally connected to a preferred offsite source. After a loss of the preferred offsite power source to a 4.16 kV ESF bus, the onsite emergency DG supplies power to the 4.16 kV ESF bus. Control power for the 4.16 kV breakers is supplied from the Class 1E batteries. Additional description of this system may be found in the Bases for LCOs 3.8.1 and 3.8.4.

The 480 V system include the safety-related load centers, motor control centers, and distribution panels shown in Table B 3.8.9-1.

The 120 VAC vital buses are divided into four independent and isolated subsystems and are normally supplied from an inverter. The alternate power supply for the vital buses are non-Class 1E 120 VAC Buses fed from a Class 1E ESF motor control center through the regulating transformer, and its use is governed by LCO 3.8.7. Each constant voltage source transformer is powered from a Class 1E AC bus.

There are four independent 125 VDC electrical power distribution subsystems.

The list of all required Distribution Systems-Operating is presented in Table B 3.8.9-1.

APPLICABLE SAFETY ANALYSES

The initial conditions of DBA and transient analyses in Reference 1, Chapters 6 and 14, assume ESF systems are OPERABLE. The AC, DC, and AC vital bus Electrical Power Distribution Systems are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, RCS, and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for Sections 3.2, 3.4, and 3.6.

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

a. An assumed loss of all offsite power or all onsite AC electrical power; and

b. A worst case single failure.

The distribution systems satisfy 10 CFR 50.36(c)(2)(ii), Criterion 3.

LCO

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

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

OPERABLE AC electrical power distribution subsystems require the associated buses, load centers, motor control centers, and distribution panels to be energized to their proper voltages. OPERABLE DC electrical power distribution subsystems require the associated buses to be energized to their proper voltage from either the associated battery or charger. OPERABLE vital bus electrical distribution subsystems require the associated buses to be energized to their proper voltage.

In addition, tie breakers between redundant safety-related AC, DC, and AC vital bus distribution subsystems, if they exist, must be open. This prevents any electrical malfunction in any distribution subsystem from propagating to the redundant subsystem, which could cause the failure of a redundant subsystem and a loss of essential safety function(s). If any tie breakers are closed, the affected redundant electrical distribution subsystems are considered inoperable. This applies to the onsite, safety-related redundant electrical power distribution subsystems.

APPLICABILITY

The electrical 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 distribution subsystem requirements for MODEs 5 and 6 are covered in the Bases for LCO 3.8.10.

ACTIONS

A.1

With one or more required AC buses, load centers, motor control centers, or distribution panels, except AC vital buses, inoperable and a loss of function has not yet 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 required AC buses, load centers, motor control centers, and distribution panels must be restored to OPERABLE status within eight hours.

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

a. The potential for decreased safety if the unit operator's attention is diverted from the evaluations and actions necessary to restore power to the affected train, to the actions associated with taking the unit to shutdown within this time limit; and

b. The potential for an event in conjunction with a single failure of a redundant component in the train with AC power.

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

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

B.1

With one or more AC vital buses inoperable and a loss of Function has not yet occurred, the remaining OPERABLE AC vital buses are capable of supporting the minimum safety functions necessary to shut down the unit and maintain it in the safe shutdown condition. Overall reliability is reduced, however, since an additional single failure could result in the minimum required ESF functions not being supported. Therefore, the AC vital bus must be restored to OPERABLE status within two hours by powering the bus from an associated inverter via DC or the non-Class 1E 120 VAC bus powered by an ESF motor control center through a regulating transformer.

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

This two hour limit is more conservative than Completion Times allowed for the vast majority of components that are without adequate vital AC power. Taking exception to LCO 3.0.2 for components without adequate vital AC power, which would have the Required Action Completion Times shorter than two 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 two hour Completion Time takes into account the importance to safety of restoring the AC vital bus to OPERABLE status, the redundant capability afforded by the other OPERABLE vital buses, and the low probability of a DBA occurring during this period.

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

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

C.1

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

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

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

a. The potential for decreased safety by requiring a change in unit conditions (i.e., requiring a shutdown) while allowing stable operations to continue;

b. The potential for decreased safety by requiring entry into numerous applicable Conditions and Required Actions for components without DC power and not providing sufficient time for the operators to perform the necessary evaluations and actions for restoring power to the affected train; and

c. The potential for an event in conjunction with a single failure of a redundant component.

The two hour Completion Time for DC buses is consistent with Reference 2.

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

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

D.1 and D.2

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

E.1

Condition E corresponds to a level of degradation in the electrical distribution system that causes a required safety function to be lost. When more than one inoperable electrical power distribution subsystem results in the loss of a required function, the plant is in a condition outside the accident analysis. Therefore, no additional time is justified for continued operation. Limiting Condition for Operation 3.0.3 must be entered immediately to commence a controlled shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.8.9.1

This SR verifies that the AC, DC, and AC vital bus Electrical Power Distribution Systems are functioning properly, with the correct circuit breaker alignment. The correct breaker alignment ensures the appropriate separation and independence of the electrical divisions is maintained, and the appropriate voltage is available to each required bus. The verification of proper voltage availability on the buses ensures that the required voltage is readily available for motive as well as control functions for critical system

BASES

loads connected to these buses. The seven day Frequency takes into account the redundant capability of the AC, DC, and AC vital bus electrical power distribution subsystems, and other indications available in the Control Room that alert the operator to subsystem malfunctions.

REFERENCES

1. UFSAR
2. Regulatory Guide 1.93, "Availability of Electric Power Sources," December 1974

Table B 3.8.9-1 (page 1 of 1)
AC and DC Electrical Power Distribution Systems[1]

4160 Volt Emergency Bus No. 11 (Unit 1), No. 21 (Unit 2)
4160 Volt Emergency Bus No. 14 (Unit 1), No. 24 (Unit 2)
480 Volt Emergency Bus No. 11A (Unit 1), No. 21A (Unit 2)
480 Volt Emergency Bus No. 11B (Unit 1), No. 21B (Unit 2)
480 Volt Emergency Bus No. 14A (Unit 1), No. 24A (Unit 2)
480 Volt Emergency Bus No. 14B (Unit 1), No. 24B (Unit 2)
480 Volt Emergency Bus No. 104R (Unit 1), No. 204R (Unit 2)
480 Volt Emergency Bus No. 114R (Unit 1), No. 214R (Unit 2)
120 Volt AC Vital Bus No. 11 (Unit 1), No. 21 (Unit 2)
120 Volt AC Vital Bus No. 12 (Unit 1), No. 22 (Unit 2)
120 Volt AC Vital Bus No. 13 (Unit 1), No. 23 (Unit 2)
120 Volt AC Vital Bus No. 14 (Unit 1), No. 24 (Unit 2)
125 Volt DC Bus No. 11 (Unit 1 and Unit 2)
125 Volt DC Bus No. 12 (Unit 1 and Unit 2)
125 Volt DC Bus No. 21 (Unit 1 and Unit 2)
125 Volt DC Bus No. 22 (Unit 1 and Unit 2)

[1] Each bus of the AC and DC Electrical Power Distribution System is a subsystem.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.10 Distribution Systems-Shutdown

BASES

BACKGROUND

A description of the AC, DC, and AC vital bus Electrical Power Distribution Systems is provided in the Bases for LCO 3.8.9.

The list of all required Distribution Systems-Shutdown is presented in Table B 3.8.10-1.

APPLICABLE SAFETY ANALYSES

The initial conditions of a DBA and transient analyses in Reference 1, Chapters 6 and 14, assume ESF systems are OPERABLE. The AC, DC, and AC vital bus Electrical Power Distribution Systems are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, RCS, and containment design limits are not exceeded.

The OPERABILITY of the AC, DC, and AC vital bus Electrical Power Distribution System is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY.

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

a. The unit can be maintained in the shutdown or refueling condition for extended periods;

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and

c. Adequate power is provided to mitigate events postulated during shutdown, such as a fuel handling accident.

The AC and DC Electrical Power Distribution Systems satisfy 10 CFR 50.36(c)(2)(ii), Criterion 3.

LCO

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

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

APPLICABILITY

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

a. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel in the core;

b. Systems needed to mitigate a fuel handling accident are available;

c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition and refueling condition.

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

ACTIONS

Limiting Condition for Operation 3.0.3 is not applicable while in MODEs 5 or 6. However, since irradiated fuel assembly movement can occur in MODEs 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 MODEs 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODEs 1, 2, 3, or 4, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of irradiated fuel assemblies would not be sufficient reason to require a reactor shutdown.

The ACTIONS have been modified by a second Note stating that performance of Required Actions shall not preclude completion of actions to establish a safe conservative position. This clarification is provided to avoid stopping movement of irradiated fuel assemblies while in a non-conservative position based on compliance with the Required Actions.

<u>A.1, A.2.1, A.2.2, A.2.3, A.2.4, and A.2.5</u>

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

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

Notwithstanding performance of the above conservative Required Actions, a required shutdown cooling (SDC) subsystem may be inoperable. In this case, Required

BASES

Actions A.2.1 through A.2.4 do not adequately address the concerns relating to coolant circulation and heat removal. Pursuant to LCO 3.0.6, the SDC ACTIONS would not be entered. Therefore, Required Action A.2.5 is provided to direct declaring SDC inoperable, which results in taking the appropriate SDC actions. The SDC subsystem(s) declared inoperable and not in operation as a result of not meeting this LCO, may be used if needed. However, the appropriate actions are still required to be taken.

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.

SURVEILLANCE REQUIREMENTS

SR 3.8.10.1

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

REFERENCES

1. UFSAR

Table B 3.8.10-1 (page 1 of 1)

AC and DC Electrical Power Distribution Systems

1	4160 Volt Emergency Bus
1	480 Volt Emergency Bus
2	120 Volt AC Vital Busses
2	125 Volt DC Busses
2	125 Volt Battery Banks (one of which may be the reserve battery) (one associated battery charger per battery bank supplying the required DC busses)

B 3.9 REFUELING OPERATIONS

B 3.9.1 Boron Concentration

BASES

BACKGROUND

The limit on the boron concentrations of the Reactor Coolant System (RCS) and refueling pool 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 that have direct access to the reactor core during refueling.

The soluble boron concentration offsets the core reactivity and is measured by chemical analysis of a representative sample of the coolant in each of the volumes. The refueling boron concentration limit is specified in the Core Operating Limits Report (COLR). Unit procedures ensure the specified boron concentration in order to maintain an overall core reactivity of $k_{eff} \leq 0.95$ during fuel handling, with control element assemblies and fuel assemblies assumed to be in the most adverse configuration (least negative reactivity) allowed by unit procedures. The negative worth of the CEAs may be credited when determining the refueling boron concentrations during full core onloads/offloads only. Unit procedures maintain the number and position of credited CEAs during fuel handling operations.

Reference 1, Appendix 1C, Criterion 27, requires that two independent reactivity control systems of different design principles be provided. One of these systems must be capable of holding the reactor core subcritical under cold conditions. The Chemical and Volume Control System 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 pool. The refueling pool is then flooded with borated water from the refueling water tank into the open reactor vessel by gravity feeding or by the use of the Shutdown Cooling (SDC) System pumps.

BASES

The pumping action of the SDC System in the RCS and the natural circulation due to thermal driving heads in the reactor vessel and the refueling pool mix the added concentrated boric acid with the water in the RCS. The SDC System is in operation during refueling [see Limiting Condition of Operation (LCO) 3.9.4 and LCO 3.9.5], to provide forced circulation in the RCS and assist in maintaining the boron concentrations in the RCS and the refueling pool above the COLR limit.

APPLICABLE SAFETY ANALYSES

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

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

During refueling, the water volume in the spent fuel pool, the transfer tube, the refueling pool, 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 (Reference 1, Chapter 14). A detailed discussion of this event is provided in B 3.1.1.

The RCS boron concentration satisfies 10 CFR 50.36(c)(2)(ii), Criterion 2.

LCO

The LCO requires that a minimum boron concentration be maintained in the RCS and refueling pool while in MODE 6. The boron concentration limit specified in the COLR ensures a core k_{eff} of ≤ 0.95 is maintained during fuel handling

BASES

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

APPLICABILITY

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

ACTIONS

A.1 and A.2

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

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

A.3

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

In determining the required combination of boration flow rate and concentration, there is no unique design basis event that must be satisfied. The only requirement is to restore the boron concentration to its required value as soon as possible. In order to raise the boron concentration as soon as possible, the operator should begin boration with the best source available for unit conditions.

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

BASES

SURVEILLANCE REQUIREMENTS	**SR 3.9.1.1** This Surveillance Requirement (SR) ensures the coolant boron concentration in the RCS and the refueling pool is within the COLR limits. The coolant boron concentration in each volume is determined periodically by chemical analysis. A minimum Frequency of once every 72 hours is therefore a reasonable amount of time to verify the boron concentration of representative samples. The Frequency is based on operating experience, which has shown 72 hours to be adequate.
REFERENCES	1. Updated Final Safety Analysis Report (UFSAR)

B 3.9 REFUELING OPERATIONS

B 3.9.2 Nuclear Instrumentation

BASES

BACKGROUND

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

The installed SRMs are fission chambers operating in the proportional region of the gas filled detector characteristic curve. The detectors monitor the neutron flux in counts per second. The instrument range (shutdown monitors) covers five decades of neutron flux (1E+5 cps) with at least a ± 5% instrument accuracy. The detectors also provide continuous visual indication in the Control Room and an audible indication in the Control Room and Containment to alert operators to a possible dilution accident. The Nuclear Instrumentation System is designed in accordance with the criteria presented in Reference 1, Appendix 1C.

If used, portable detectors should be functionally equivalent to the Nuclear Instrumentation System SRMs.

APPLICABLE SAFETY ANALYSES

Two OPERABLE SRMs are required to provide a signal to alert the operator to unexpected changes in core reactivity such as by a boron dilution accident or an improperly loaded fuel assembly. The safety analysis of the uncontrolled boron dilution accident is described in Reference 1, Chapter 14. The analysis of the uncontrolled boron dilution accident shows that normally available SHUTDOWN MARGIN would be reduced, but there is sufficient time for the operator to take corrective actions.

The SRMs satisfy 10 CFR 50.36(c)(2)(ii), Criterion 3.

LCO

This LCO requires two SRMs OPERABLE to ensure that redundant monitoring capability is available to detect changes in core reactivity.

BASES

APPLICABILITY

In MODE 6, the SRMs must be OPERABLE to determine changes in core reactivity. There is no other direct means available to check core reactivity levels.

In MODEs 2, 3, 4, and 5, the installed source range detectors and circuitry are required to be OPERABLE by LCO 3.3.2.

ACTIONS

A.1 and A.2

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

B.1

With no SRM OPERABLE, action to restore a monitor to OPERABLE status shall be initiated immediately. Once initiated, action shall be continued until an SRM is restored to OPERABLE status.

B.2

With no SRM OPERABLE, there is no direct means of detecting changes in core reactivity. However, since CORE ALTERATIONS and positive reactivity additions are not to be made, the core reactivity condition is stabilized until the SRMs are OPERABLE. This stabilized condition is determined by performing SR 3.9.1.1 to verify that the required boron concentration exists.

The Completion Time of 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 12 hour Frequency is reasonable, considering the low probability of a change in core reactivity during this period.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.2.1

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

The Frequency of 12 hours is consistent with the CHANNEL CHECK Frequency specified similarly for the same instruments in LCO 3.3.1.

SR 3.9.2.2

Surveillance Requirement 3.9.2.2 is the performance of a CHANNEL CALIBRATION every 24 months. This SR is modified by a Note stating that neutron detectors are excluded from the CHANNEL CALIBRATION. This is because generating a meaningful test signal is difficult; the detectors are of simple construction, and any failures in the detectors will be apparent as a change in channel output. This Frequency is the same as that employed for the same channels in the other applicable MODEs.

REFERENCES

1. UFSAR

B 3.9 REFUELING OPERATIONS

B 3.9.3 Containment Penetrations

BASES

BACKGROUND

During CORE ALTERATIONS or movement of irradiated fuel assemblies within the Containment Structure, a release of fission product radioactivity within the Containment Structure 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. In MODE 6, the potential for containment pressurization as a result of an accident is not likely; therefore, requirements to isolate the containment atmosphere 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 design basis accident potential for containment pressurization, the Appendix J leakage criteria and tests are not required.

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

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

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

In other situations, the potential for containment pressurization as a result of an accident is not present, therefore, less stringent requirements are needed to isolate the containment atmosphere from the outside atmosphere. Both containment personnel air lock doors may be open during the movement of irradiated fuel assemblies in containment and during CORE ALTERATIONS; provided one air lock door is OPERABLE, the plant is in MODE 6 with at least 23 ft of water above the fuel, and a designated individual is continuously available to close the air lock door. This individual must be stationed at the Auxiliary Building side of the outer air lock door. OPERABILITY of a containment personnel air lock door requires that the door is capable of being closed, that the door is unblocked, and no cables or hoses are run through the air lock. During CORE ALTERATIONS or movement of irradiated fuel assemblies in containment, the requirement for at least 23 ft of water above the fuel, ensures that there is sufficient time to close the personnel air lock following a loss of SDC before boiling occurs and minimizes activity release after a fuel handling accident. (The water level requirement needs to be stated in this Specification to address an exemption to CORE ALTERATIONS in the Applicability of LCO 3.9.6 regarding control element assembly drive shaft coupling and uncoupling.)

The requirements on containment penetration closure, ensure that a release of fission product radioactivity within the Containment Structure will be restricted to within regulatory limits.

The Containment Purge Valve Isolation System, for the purposes of compliance with LCO 3.9.3, item d.2, includes a 48 inch purge penetration and a 48 inch exhaust penetration. For the purposes of compliance with LCO 3.9.3, the containment vent isolation valves are not considered part of the Containment Purge Valve Isolation System since they may not be capable of being closed automatically. The containment vent, includes a four inch purge penetration and a four inch exhaust penetration. During MODEs 1, 2, 3, and 4, the normal purge and exhaust penetrations are isolated

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(via a blind flange, if installed or by the purge valves). The containment vent valves can be opened intermittently, but are closed automatically by the Engineered Safety Features Actuation System. Neither of the subsystems is subject to a Specification in MODE 5.

In MODE 6, large air exchanges are desired to conduct refueling operations. The normal 48 inch purge system is used for this purpose and all valves are closed by the Engineered Safety Features Actuation System in accordance with LCO 3.3.7.

The containment vent isolation valves are required to be closed during CORE ALTERATIONS or movement of irradiated fuel within Containment. These valves are connected to the penetration room Technical Specification emergency air cleanup systems, which exhaust to the outside atmosphere through high efficiency particulate air and charcoal filters.

The other containment penetrations that provide direct access from containment atmosphere to outside atmosphere must be isolated on at least one side. Isolation may be achieved by an OPERABLE automatic isolation valve, or by a manual isolation valve, blind flange, or equivalent. Equivalent isolation methods must be approved in accordance with appropriate American Society of Mechanical Engineers / American National Standards Institute Codes, and may include use of a material that can provide a temporary ventilation barrier for the other containment penetrations during fuel movements.

APPLICABLE SAFETY ANALYSES

During CORE ALTERATIONS or movement of irradiated fuel assemblies within the Containment Structure, 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 (Reference 1). The fuel handling accident, described in Reference 1, includes dropping a single irradiated fuel assembly which would then rotate to a horizontal position, strike a protruding structure, and rupture the fuel pins. The requirements of LCO 3.9.6, and the minimum decay time of 100 hours prior to CORE ALTERATIONS, ensure that the release of fission product

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radioactivity, subsequent to a fuel handling accident, results in doses that are within the acceptance limits given in Reference 1.

Containment penetrations satisfy 10 CFR 50.36(c)(2)(ii), Criterion 3.

LCO

This LCO limits the consequences of a fuel handling accident in Containment Structure, by limiting the potential escape paths for fission product radioactivity released within Containment. The LCO requires any penetration providing direct access from the Containment Structure atmosphere to the outside atmosphere (including the containment vent isolation valves) 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 the Containment Purge Valve Isolation System. The OPERABILITY requirements for this LCO ensure that the automatic purge and exhaust valve closure times specified in the UFSAR can be achieved, and therefore meet the assumptions used in the safety analysis to ensure releases through the valves are terminated, such that the radiological doses are within the acceptance limit.

Both containment personnel air lock doors may be open under administrative controls during movement of irradiated fuel in Containment and during CORE ALTERATIONS provided that one OPERABLE door is capable of being closed in the event of a fuel handling accident. Note that the 23 ft referred to in the LCO of the Technical Specification is a minimum water level, and water levels greater than 23 ft are acceptable and conservative. The administrative controls consist of a designated individual available immediately outside the personnel air lock to close an OPERABLE air lock door. Should a fuel handling accident occur inside the Containment Structure, one personnel air lock door will be closed following an evacuation of the Containment Structure.

The LCO is modified by a Note which allows the emergency air lock temporary closure device to replace an emergency air lock door. The temporary closure device provides an

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adequate barrier to shield the environment from the containment atmosphere in case of a design basis event.

APPLICABILITY

The containment penetration requirements are applicable during CORE ALTERATIONS or movement of irradiated fuel assemblies within the Containment Structure because this is when there is a potential for a fuel handling accident. In MODEs 1, 2, 3, and 4, containment penetration requirements are addressed by LCO 3.6.1. In MODEs 5 and 6, when CORE ALTERATIONS or movement of irradiated fuel assemblies within the Containment Structure are not being conducted, the potential for a fuel handling accident does not exist. Therefore, under these conditions no requirements are placed on containment penetration status.

ACTIONS

A.1 and A.2

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

SURVEILLANCE REQUIREMENTS

SR 3.9.3.1

This SR demonstrates that each of the containment penetrations required to be in its closed position, is in that position. The surveillance test on the open purge and exhaust valves will demonstrate that the valves are not blocked from closing. Also, the surveillance test will demonstrate that each purge and exhaust valve operator has motive power, which will ensure each valve is capable of being closed by an OPERABLE automatic Containment Purge Valve Isolation System.

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The surveillance test is performed every seven days during CORE ALTERATIONS or movement of irradiated fuel assemblies within the Containment Structure. The surveillance test interval is selected to be commensurate with the normal duration of time to complete fuel handling operations. A surveillance test before the start of refueling operations will provide two or three verifications during the applicable period for this LCO. As such, this SR ensures that a postulated fuel handling accident, that releases fission product radioactivity within the Containment Structure, will not result in a release of fission product radioactivity to the environment in excess of those described in Reference 1.

SR 3.9.3.2

This SR demonstrates that each containment purge and exhaust valve actuates to its isolation position on manual initiation or on an actual or simulated high radiation signal. The once each refueling outage Frequency, maintains consistency with other similar Engineered Safety Features Actuation System instrumentation and valve testing requirements. However, in order to ensure the SR Frequency is satisfied, this surveillance test is typically performed once per refueling outage prior to the start of CORE ALTERATIONS or movement of irradiated fuel assemblies within Containment. In LCO 3.3.7, the Containment Radiation Signal System requires a CHANNEL CHECK every 12 hours and a CHANNEL FUNCTIONAL TEST every 92 days to ensure the channel OPERABILITY during refueling operations. Every 24 months a CHANNEL CALIBRATION is performed. The system actuation response time is demonstrated every 24 months during refueling on a STAGGERED TEST BASIS. Surveillance Requirement 3.6.3.4 demonstrates that the isolation time of each valve is in accordance with the Inservice Testing Program requirements. These surveillance tests performed during MODE 6 will ensure that the valves are capable of closing after a postulated fuel handling accident to limit a release of fission product radioactivity from the Containment Structure.

REFERENCES

1. UFSAR, Section 14.18, "Fuel Handling Incident"

B 3.9 REFUELING OPERATIONS

B 3.9.4 Shutdown Cooling (SDC) and Coolant Circulation-High Water Level

BASES

BACKGROUND

The purposes of the SDC System in MODE 6 are to remove decay heat and other residual heat from the RCS, to provide mixing of borated coolant, to provide sufficient coolant circulation to minimize the effects of a boron dilution accident, and to prevent boron stratification (Reference 1). Heat is removed from the RCS by circulating reactor coolant through the SDC heat exchanger(s), where the heat is transferred to the Component Cooling Water System via the SDC heat exchanger(s). The coolant is then returned to the RCS via the RCS cold leg(s). Operation of the SDC System for normal cooldown or decay heat removal is manually accomplished from the Control Room. The heat removal rate is adjusted by controlling the flow of reactor coolant through the SDC heat exchanger(s) and bypassing the heat exchanger(s). Mixing of the reactor coolant is maintained by this continuous circulation of reactor coolant through the SDC System.

APPLICABLE SAFETY ANALYSES

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

BASES

Shutdown cooling and Coolant Circulation-High Water Level satisfies 10 CFR 50.36(c)(2)(ii), Criterion 2.

LCO

Only one SDC loop is required for decay heat removal in MODE 6, with water level ≥ 23 ft above the top of the irradiated fuel assemblies seated in the reactor vessel. Only one SDC loop is required because the volume of water above the irradiated fuel assemblies seated in the reactor vessel provides backup decay heat removal capability. At least one SDC loop must be OPERABLE and in operation to provide:

a. Removal of decay heat;

b. Mixing of borated coolant to minimize the possibility of a criticality; and

c. Indication of reactor coolant temperature.

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

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

A second Note also allows both SDC loops not to be in operation during the time required for local leak rate testing of Containment Penetration Number 41 pursuant to the requirements of SR 3.6.1.1 or to permit maintenance on valves located in the common SDC suction line. In addition to the requirement in Note 1 regarding control of boron

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concentration, CORE ALTERATIONS are suspended and all containment penetrations providing direct access from containment atmosphere to outside atmosphere must be closed. This allowance is necessary to perform required maintenance and testing.

APPLICABILITY

One SDC loop must be in operation in MODE 6, with the water level $\geq$ 23 ft above the top of the irradiated fuel assemblies seated in the reactor vessel, to provide decay heat removal. The 23 ft level was selected because it corresponds to the 23 ft requirement established for fuel movement in LCO 3.9.6.

Requirements for the SDC System in other MODEs are covered by LCOs in Section 3.4 and Section 3.5. Shutdown cooling loop requirements in MODE 6, with the water level < 23 ft above the top of the irradiated fuel assemblies seated in the reactor vessel, are located in LCO 3.9.5.

ACTIONS

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

A.1

If one required SDC loop is inoperable or not in operation, action shall be immediately initiated and continued until the SDC loop is restored to OPERABLE status and to operation. An immediate Completion Time is necessary for an operator to initiate corrective actions.

A.2

If SDC loop requirements are not met, there will be no forced circulation to provide mixing to establish uniform boron concentrations. Reduced boron concentrations can occur through the addition of water with a lower boron concentration than that contained in the RCS. Therefore, actions that reduce boron concentration shall be suspended immediately. In addition, to ensure compliance with the action is maintained, the charging pumps shall be de-energized and charging flow paths closed as part of Required Action A.2.

BASES

A.3

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

A.4

If SDC loop requirements are not met, all containment penetrations to the outside atmosphere must be closed to prevent fission products, if released by a loss of decay heat event, from escaping the Containment Structure. The four hour Completion Time allows fixing most SDC problems without incurring the additional action of violating the containment atmosphere.

SURVEILLANCE REQUIREMENTS

SR 3.9.4.1

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

REFERENCES

1. UFSAR, Section 9.2, "Shutdown Cooling System"

B 3.9 REFUELING OPERATIONS

B 3.9.5 Shutdown Cooling (SDC) and Coolant Circulation-Low Water Level

BASES

BACKGROUND

The purposes of the SDC System in MODE 6 are to remove decay heat and other residual heat from the RCS, to provide mixing of borated coolant, to provide sufficient coolant circulation to minimize the effects of a boron dilution accident, and to prevent boron stratification (Reference 1). Heat is removed from the RCS by circulating reactor coolant through the SDC heat exchanger(s), where the heat is transferred to the Component Cooling Water System via the SDC heat exchanger(s). The coolant is then returned to the RCS via the RCS cold leg(s). Operation of the SDC System for normal cooldown or decay heat removal is manually accomplished from the Control Room. The heat removal rate is adjusted by controlling the flow of reactor coolant through the SDC heat exchanger(s) and bypassing the heat exchanger(s). Mixing of the reactor coolant is maintained by this continuous circulation of reactor coolant through the SDC System.

APPLICABLE SAFETY ANALYSES

If the reactor coolant temperature is not maintained below 200°F, boiling of the reactor coolant could result. This could lead to inadequate cooling of the reactor fuel due to the resulting loss of coolant in the reactor vessel. Additionally, boiling of the reactor coolant could lead to a reduction in boron concentration in the coolant due to the boron plating out on components near the areas of the boiling activity, and due to the possible addition of water to the reactor vessel with a lower boron concentration than is required to keep the reactor subcritical. The loss of reactor coolant and the reduction of boron concentration in the reactor coolant would eventually challenge the integrity of the fuel cladding, which is a fission product barrier. Two loops of the SDC System are required to be OPERABLE, and one loop is required to be in operation in MODE 6, with the water level < 23 ft above the top of the irradiated fuel assemblies seated in the reactor vessel, to prevent this challenge.

Shutdown cooling and Coolant Circulation-Low Water Level satisfies 10 CFR 50.36(c)(2)(ii), Criterion 2.

BASES

LCO

In MODE 6, with the water level < 23 ft above the top of the irradiated fuel assemblies seated in the reactor vessel, both SDC loops must be OPERABLE. Additionally, one loop of the SDC System must be in operation in order to provide:

a. Removal of decay heat;

b. Mixing of borated coolant to minimize the possibility of a criticality; and

c. Indication of reactor coolant temperature.

An OPERABLE SDC loop consists of an SDC pump, a heat exchanger, valves, piping, instruments, and controls to ensure an OPERABLE flow path and to determine the low end temperature. The flow path starts in one of the RCS hot legs and is returned to the RCS cold legs. Both SDC pumps may be aligned to the Refueling Water Tank to support filling the refueling pool or for performance of required testing.

The LCO is modified by a note that allows one required SDC loop to be replaced by one spent fuel pool cooling loop when it is lined up to provide cooling flow to the irradiated fuel assemblies in the reactor core, and the heat generation rate of the core is below the heat removal capacity of the spent fuel cooling loop.

APPLICABILITY

Two SDC loops are required to be OPERABLE, and one SDC loop must be in operation in MODE 6, with the water level < 23 ft above the top of the irradiated fuel assemblies seated in the reactor vessel, to provide decay heat removal. Requirements for the SDC System in other MODEs are covered by LCOs in Section 3.4. MODE 6 requirements, with a water level ≥ 23 ft above the irradiated fuel assemblies seated in the reactor vessel, are covered in LCO 3.9.4.

ACTIONS

A.1 and A.2

If one SDC loop is inoperable, action shall be immediately initiated and continued until the SDC loop is restored to OPERABLE status and is in operation, or until the water level is ≥ 23 ft above the irradiated fuel assemblies seated in the reactor vessel. When the water level is established at ≥ 23 ft above the irradiated fuel assemblies seated in the reactor vessel, the Applicability will change to that of LCO 3.9.4, and only one SDC loop is required to be OPERABLE and in operation. An immediate Completion Time is necessary for an operator to initiate corrective actions.

B.1

If no SDC loop is in operation or no SDC loops are OPERABLE, there will be no forced circulation to provide mixing to establish uniform boron concentrations. Reduced boron concentrations can occur by the addition of water with lower boron concentration than that contained in the RCS. Therefore, actions that reduce boron concentration shall be suspended immediately. In addition, to ensure compliance with the action is maintained, the charging pumps shall be de-energized and charging flow paths closed as part of Required Action B.1.

B.2

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

B.3

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

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

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.5.1

This SR demonstrates that one SDC loop is operating and circulating reactor coolant. The flow rate is determined by the flow rate necessary to provide sufficient decay heat removal capability and to prevent thermal and boron stratification in the core. This SR also demonstrates that the other SDC loop is OPERABLE.

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

Verification that the required loops are OPERABLE and in operation ensures that loops can be placed in operation as needed, to maintain decay heat and retain forced circulation. The Frequency of 12 hours is considered reasonable, since other administrative controls are available and have proven to be acceptable by operating experience.

SR 3.9.5.2

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

SR 3.9.5.3

Verification that the required pump and valves are OPERABLE ensures that an additional SDC loop can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the required pump and valves. The Frequency of seven days

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is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

REFERENCES

1. UFSAR, Section 9.2, "Shutdown Cooling System"

B 3.9 REFUELING OPERATIONS

B 3.9.6 Refueling Pool Water Level

BASES

BACKGROUND	The movement of irradiated fuel assemblies or performance of CORE ALTERATIONS, except during latching and unlatching of control rod drive shafts, within the Containment Structure requires a minimum water level of 23 ft above the top of the irradiated fuel assemblies seated in the reactor vessel. During refueling this maintains sufficient water level in the fuel transfer canal, the refueling pool, and the spent fuel pool. Sufficient water is necessary to retain iodine fission product activity in the water in the event of a fuel handling accident (References 1 and 2). Sufficient iodine activity would be retained to limit offsite doses from the accident to within the acceptance criteria given in Reference 2.
APPLICABLE SAFETY ANALYSES	During core alterations and during movement of irradiated fuel assemblies, the water level in the refueling pool is an initial condition design parameter in the analysis of the fuel handling accident in the Containment Structure, postulated by Reference 1. A minimum water level of 23 ft (Regulatory Position C.1.c of Reference 1) allows a decontamination factor of 100 (Regulatory Position C.1.g of Reference 1) to be used in the accident analysis for iodine. This relates to the assumption that 99% of the total iodine released from the pellet to cladding gap of all the dropped fuel assembly rods, is retained by the refueling pool water. The fuel pellet to cladding gap is assumed to contain 10% of the total fuel rod iodine inventory (Reference 1). The fuel handling accident analysis inside the Containment Structure 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 (Reference 2). Refueling pool water level satisfies 10 CFR 50.36(c)(2)(ii), Criterion 2.

BASES

LCO

A minimum refueling pool water level of 23 ft above the irradiated fuel assemblies seated in the reactor vessel is required to ensure that the radiological consequences of a postulated fuel handling accident inside the Containment Structure are within the acceptable limits given in Reference 2.

APPLICABILITY

LCO 3.9.6 is applicable during CORE ALTERATIONS, except during coupling and uncoupling of control element assembly drive shafts, and when moving irradiated fuel assemblies in the Containment Structure. The LCO minimizes the possibility of a fuel handling accident in the Containment Structure that is beyond the assumptions of the safety analysis. If irradiated fuel is not present in the Containment Structure, 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.

ACTIONS

A.1 and A.2

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

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

SURVEILLANCE REQUIREMENTS

SR 3.9.6.1

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

BASES

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

REFERENCES

1. Regulatory Guide 1.25, Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors (Safety Guide 25), March 1972
2. UFSAR, Section 14.18, "Fuel Handling Incident"