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Vol. 2

Standard Technical Specifications Babcock and Wilcox Plants

Bases (Sections 2.0–3.3)

Draft Report for Comment

Issued by the
U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

January 1991

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STANDARD TECHNICAL SPECIFICATIONS
BABCOCK AND WILCOX PLANTS

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PREFACE

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

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

Comments should be submitted no later than March 15, 1991, in accordance with the following guidance: The exact wording of each proposed change should be marked in pen and ink on copies of all the affected pages of DRAFT NUREG-1430, "Standard Technical Specifications, Babcock and Wilcox Plants." Each proposed change should be numbered. Each proposed change should be accompanied with a separate technical justification, cross referenced to the applicable proposed change on the marked up pages.

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

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B 2.0 SAFETY LIMITS

B 2.1.1 Reactor Core Safety Limits (SLs)

BASES

BACKGROUND

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

Centerline fuel melting occurs when the local LHR, or power peaking, in a region of the fuel is high enough to cause the fuel-centerline temperature to reach the melting point of the fuel. Expansion of the pellet upon centerline melting may cause the pellet to stress the cladding to the point of failure, allowing an uncontrolled release of activity to the reactor coolant.

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

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BASES (continued)

BACKGROUND (continued)

The proper functioning of the Reactor Protection System (RPS) and steam generator (SG) safety valves prevents 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 that the hot fuel rod in the core does not experience DNB (this is referred to hereafter as the 95/95 DNB criterion); and

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

The RPS setpoints (Ref. 2), in combination with all the 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 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 are provided by the trip setpoints for the following functions:

a. RCS high-pressure reactor trip;

b. RCS low-pressure reactor trip;

c. Nuclear overpower reactor trip;

d. RCS variable low pressure;

e. Reactor coolant pump-to-power;

f. SG safety valves;

(continued)

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

g. Nuclear overpower RCS flow and axial power imbalance; and

h. Loss of main feedwater pumps.

The limitation that the average enthalpy in the hot leg be less than or equal to the enthalpy of saturated liquid also ensures that the ΔT measured by instrumentation (used in the protection system design as a measure of the core power) is proportional to core power.

The SL represents a design requirement for establishing the RPS trip setpoints identified previously. LCO 3.2.3, "Axial Power Imbalance Operating Limits," or the assumed initial condition of the safety analyses (as indicated in the FSAR, Ref. 2) provide more restrictive limits that ensure the SLs are not exceeded.

SAFETY LIMITS

SL 2.1.1.1, SL 2.1.1.2, and SL 2.1.1.3 ensure that the minimum DNBR is not less than the safety analyses limit and that fuel centerline temperature stays below the melting point, or the average enthalpy in the hot leg is less than or equal to the enthalpy of saturated liquid or the exit quality is within the limits defined by the DNBR correlation. In addition, the SL 2.1.1.3 shows the pressure-temperature operating region that keeps the reactor from reaching an SL when operating at ≤ 100% steady-state power, and it defines the safe operating region from brittle-fracture concerns.

The SLs are observed by monitoring the process variable AXIAL POWER IMBALANCE to ensure that the core operates within the fuel design criteria. AXIAL POWER IMBALANCE protective limits are provided in the CORE OPERATING LIMITS REPORT (COLR). The trip setpoints are derived by adjusting the measurement-system-independent AXIAL POWER IMBALANCE protective limit given in the COLR to allow for measurement system observability and instrumentation errors.

Operation within these limits is assured by compliance with the AXIAL POWER IMBALANCE protective limits preserved by the RPS setpoints in LCO 3.3.1, as specified in the COLR.

(continued)

BASES (continued)

APPLICABILITY

SL 2.1.1.1, SL 2.1.1.2, and SL 2.1.1.3 only apply 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 SG safety valves or automatic protection actions serve to prevent RCS heatup to 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 and LCO 3.3.2.

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

SAFETY LIMIT VIOLATIONS

2.2.1 and 2.2.2

If SL 2.1.1.1, SL 2.1.1.2, or SL 2.1.1.3 is violated, the requirement to go to MODE 3 places the plant in a MODE in which these SLs are not applicable.

The allowed Completion Time of 1 hour recognizes the importance of bringing the plant to a mode of operation where these SLs are not applicable. Also, the Completion Time of 1 hour ensures that the probability of an accident occurring during this period is minimal. The allowed Completion Time of 15 minutes to restore RCS pressure and temperature to within limits implies immediacy.

2.2.5

If SL 2.1.1.1, SL 2.1.1.2, or SL 2.1.1.3 is violated, the NRC Operations Center must be notified within 1 hour. This is in accordance with 10 CFR 50.72 (Ref. 3).

2.2.6

If SL 2.1.1.1, SL 2.1.1.2, or SL 2.1.1.3 is violated, the appropriate senior management of the nuclear plant and the utility shall be notified within 24 hours. This 24-hour period provides time for plant operators and staff to take the appropriate immediate action and assess the condition of the plant before reporting to senior management.

(continued)

(continued)

BASES (continued)

SAFETY LIMIT VIOLATIONS (continued)

2.2.7

If SL 2.1.1.1, SL 2.1.1.2, or SL 2.1.1.3 is violated, a Licensee Event Report shall be prepared and submitted within 30 days to the NRC, the senior management of the nuclear plant, and the utility Vice-President—Nuclear Operations. This requirement is in accordance 10 CFR 50.73 (Ref. 4).

2.2.8

If SL 2.1.1.1, SL 2.1.1.2, or SL 2.1.1.3 is violated, restart of the unit shall not commence until authorized by the NRC. This requirement ensures the NRC that all necessary reviews, analyses, and actions are completed before the unit begins its restart to normal operation.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix A, General Design Criterion 10, "Reactor Design," 1988.
2. [Unit Name] FSAR, Section [ ], "[Title]."
3. Title 10, Code of Federal Regulations, Part 50.72, "Immediate Notification Requirements for Operating Nuclear Power Reactors."
4. Title 10, Code of Federal Regulations, Part 50.73, "Licensee Event Report System."

B 2.0 SAFETY LIMITS

B 2.1.2 Reactor Coolant System (RCS) Pressure Safety Limit (SL)

BASES

BACKGROUND

The SL on RCS pressure protects the integrity of the RCS against overpressurization. In the event of fuel-cladding failure, fission products are released into the reactor coolant. The RCS then serves as the primary barrier in preventing the release of fission products into the atmosphere. By establishing an upper limit on RCS pressure, the continued integrity of the RCS is ensured. According to 10 CFR 50, Appendix A, GDC 14, "Reactor Coolant Pressure Boundary," and GDC 15, "Reactor Coolant System Design" (Ref. 1), the reactor coolant pressure boundary (RCPB) design conditions are not exceeded during normal operation nor during anticipated operational occurrences (AOOs). GDC 28 (Ref. 1), "Reactivity Limits," specifies that reactivity accidents including rod ejection do not result in damage to the RCPB greater than limited local yielding.

The design pressure of the RCS is 2500 psia. During normal operation and AOOs, the RCS pressure is kept from exceeding the design pressure by more than 10% in order to remain in accordance with Section III of the American Society of Mechanical Engineers (ASME) Code (Ref. 2). As an assurance of system integrity, all RCS components are hydrostatically tested at 125% of design, as specified in the ASME Code requirements prior to initial operation when there is no fuel in the core. If repairs or replacements that require a full hydrostatic test of the RCS are made, the fuel would have to be completely offloaded before it exceeded the maximum pressure specified in this SL. Removing fuel from the vessel precludes fission products from entering the reactor coolant.

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

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

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

The RCS pressurizer safety valves are sized to prevent system pressure from exceeding the design pressure by more than 10%, as specified in Section III of the ASME Code for Nuclear Power Plant Components (Ref. 2). The transient that establishes the required relief capacity, and hence 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 Protection System (RPS) trip setpoints (Ref. 3), together with the settings of the MSSVs (Ref. 4), provide pressure protection for normal operation and AOOs. The reactor high-pressure trip setpoint is specifically set to provide protection against overpressurization (Ref. 5). The safety analyses for both the high-pressure trip and the RCS pressurizer safety valves are performed using conservative assumptions relative to pressure control devices.

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

a. Pressurizer power-operated relief valves (PORVs);

b. Steam line relief valve;

c. Steam Dump System;

d. RCS;

e. Pressurizer Level Control System; and

f. Pressurizer spray valve.

(continued)

BASES (continued)

SAFETY LIMITS

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

APPLICABILITY

SL 2.1.2 applies in MODES 1 through 5 because this SL could be approached or exceeded in these modes during overpressurization events. The SL is not applicable in MODE 6 since the reactor vessel head closure bolts are not fully tightened, making it impossible to pressurize the RCS.

SAFETY LIMIT VIOLATIONS

2.2.3

If the RCS pressure SL is violated when the reactor is in MODE 1 or 2, the requirement is to restore compliance within 15 minutes and be in MODE 3 within 1 hour.

If the RCS pressure SL is violated while in MODE 1 or 2, the reactor vessel temperature is well above the transition temperature, at which the reactor vessel metal goes from being ductile to being nonductile. Given that the reactor vessel metal is ductile, a pressure increase above 110% of design pressure does not represent nearly the challenge to the RCS integrity it would present if the reactor vessel were in a non-ductile state therefore 15 minutes to restore pressure implies immediacy.

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

2.2.4

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

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BASES (continueo,

SAFETY LIMIT VIOLATIONS (continued)

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

2.2.5

If the RCS pressure SL is violated, the NRC Operations Center must be notified within 1 hour. This is in accordance with 10 CFR 50.72 (Ref. 7).

2.2.6

If the RCS pressure SL is violated, the appropriate senior management of the nuclear plant and the utility shall be notified within 24 hours. This 24-hour period provides time for the plant operators and staff to take the appropriate immediate action and assess the condition of the plant before reporting to the senior management.

2.2.7

If the RCS pressure SL is violated, a Licensee Event Report shall be prepared and submitted within 30 days to the NRC, the senior management of the nuclear plant, and the utility Vice-President—Nuclear Operations. This requirement is in accordance with 10 CFR 50.73 (Ref. 8).

2.2.8

If the RCS pressure SL is violated, restart of the unit shall not commence until authorized by the NRC. This requirement ensures the NRC that all necessary reviews, analyses, and actions are completed before the unit begins its restart to normal operation.

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BASES (continued)

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix A, General Design Criterion 14, "Reactor Coolant Pressure Boundary"; General Design Criterion 15, "Reactor Coolant System Design"; and General Design Criterion 28, "Reactivity Limits," 1988.

2. American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," Article NB-7000, "Protection Against Overpressure."

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

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

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

6. USAS B31.1, Standard Code for Pressure Piping, American Society of Mechanical Engineers, 1967.

7. Title 10, Code of Federal Regulations, Part 50.72, "Immediate Notification Requirements for Operating Nuclear Power Reactors."

8. Title 10, Code of Federal Regulations, Part 50.73, "Licensee Event Report System."

B 3.0 APPLICABILITY

B 3.0 Limiting Conditions for Operation (LCO) Applicability

BASES

LCO 3.0.1 LCO 3.0.2 LCO 3.0.3 LCO 3.0.4 and LCO 3.0.5	LCO 3.0.1, LCO 3.0.2, LCO 3.0.3, LCO 3.0.4, and LCO 3.0.5 establish the general requirements applicable to all specifications unless otherwise stated. This includes specifications regarding the programs in Section 5.7.4, and "Programs and Manuals," as well as LCOs contained in Sections 3.1 through 3.9.
LCO 3.0.1	LCO 3.0.1 establishes the requirement to meet LCOs when the unit is in the MODES or other specified Conditions of the Applicability statement of each specification.
LCO 3.0.2	LCO 3.0.2 establishes that upon discovery of a failure to meet an LCO, the associated ACTIONS shall be met. The Completion Time of each Required Action for an ACTIONS Condition is applicable from the point in time it is discovered that an inoperable situation exists (i.e., that the LCO is not met) associated with a Condition. Following this discovery, the associated 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. Concurrent entry into all applicable ACTIONS Conditions is a requirement to be followed in each specification. The Required Action(s) of each Condition entered must be completed within the specified Completion Time(s). There are two basic types of Required Actions. The first type of Required Action has an associated time limit in which the entered Condition must be corrected. This time limit is the Completion Time to place required equipment in operation, or 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 facility in a MODE or Condition in which the specification no longer applies. (Whether

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BASES (continued)

LCO 3.0.2 (continued)

stated as a Required Action or not, correction of the entered Condition is the first action that is to be considered upon entering ACTIONS.) The second type of Required Action specifies the remedial measures that permit continued operation of the facility that is not further restricted by the Completion Time. In this case, conformance to the Required Actions provides an acceptable level of safety for continued operation. This type of Required Action is common throughout the Technical Specifications (TS).

This Specification establishes that performance of the Required Actions within the specified Completion Times constitutes compliance with the TS. It also establishes, however, that completing the performance of the Required Actions is not required when an LCO is met within the associated Completion Time, unless otherwise specified in the individual specifications. This is equivalent to stating that correction of an ACTIONS Condition prior to the expiration of the specified Completion Time(s) makes it unnecessary to continue or complete the performance of the associated Required Action(s).

This specification is written for the more general case in which more than one of the stated Conditions are concurrently applicable. As each Condition is resolved, the Required Action(s) for that Condition no longer need be performed.

A Condition once entered or once applicable is resolved either by completing corrective measures such that it no longer exists or by placing the facility outside the Applicability of the LCO.

The nature of some Required Actions necessitates that, once begun, their performance must be completed even though the associated Conditions are resolved. The individual LCO's ACTIONS specify the Required Actions where this is the case. An example of this is in LCO 3.8.1, "AC Sources—Operating."

The above discussion about not having to complete the performance of Required Actions once the corresponding Conditions have been resolved also applies to the category

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BASES (continued)

LCO 3.0.2 (continued)

of Conditions that state, "Required Actions and associated Completion Times not met."

Usually, the Required Action for a Condition of this type is to go to an inapplicable MODE or other specified Condition. The performance of such a shutdown Required Action may be suspended if the LCO Required Action that was not performed is completed or if the LCO is restored. If the shutdown had proceeded to the point where a MODE change had occurred, however, returning to the previously applicable MODE or specified Condition is not allowed by LCO 3.0.4, unless otherwise specified.

It is possible in some LCOs (but unlikely) to enter and exit two or more ACTION's Conditions repeatedly, in such a manner that facility operation could continue indefinitely without ever having restored the LCO (i.e., the facility is always in at least one of the Conditions). Because of the risk associated with extended facility operation with certain LCOs unmet, Specification 1.3 limits such operation to the longer of the specified Completion Times for the Conditions that are concurrently entered. This limitation does not apply to Conditions where the associated Required Actions, if met, permit continued operation for an unlimited period of time.

The Completion Times of the Required Actions are also applicable when a system or component is removed from service intentionally. The reasons for intentionally relying on the ACTIONS include, but are not limited to, performance of surveillances, preventive maintenance, corrective maintenance, or investigation of operational problems. Entering ACTIONS for these reasons must be done in a manner that does not compromise safety. It is not intended that intentional entry into ACTIONS be made for operational convenience. Intentional entry into ACTIONS Conditions with shutdown Required Actions (i.e., Actions requiring a change in MODE) is strongly discouraged and should be considered only in extreme circumstances. This is to limit routine voluntary removal of redundant equipment from service in lieu of other alternatives that would not result in redundant equipment being inoperable. Individual specifications may specify a time limit for performing an SR when equipment is removed from service or bypassed for testing. In such a case, the Completion Times of the

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BASES (continued)

LCO 3.0.2 (continued)

Required Actions are applicable when this time limit expires, if the SR has not been completed. When a change in MODE or other specified Condition is required to comply with Required Actions, the facility may enter a MODE or other specified condition in which a new specification becomes applicable. Upon the new specification becoming applicable, immediately enter all ACTIONS Conditions that apply, unless otherwise specified. The Completion Times of the associated Required Actions would apply from the point in time that the new specification became applicable.

LCO 3.0.3

LCO 3.0.3 establishes the Required 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 facility 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 facility. Sometimes, possible combinations of Conditions are such that going to 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 facility 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 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. Intentional entry into LCO 3.0.3 for operational convenience constitutes noncompliance with the TS. Under suitable circumstances, intentional entry into LCO 3.0.3 for corrective action or repairs may be justified, but prior notification of the NRC should be considered.

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BASES (continued)

LCO 3.0.3 (continued)

After entering LCO 3.0.3, 1 hour is allowed to prepare for an orderly shutdown before initiating a change in facility 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 higher-numbered MODES of operation permit the shutdown to proceed in a controlled and orderly manner that is well within the specified maximum cool-down rate and within the capabilities of the facility, assuming that only the minimum required equipment is OPERABLE. This reduces thermal stresses on components of the Reactor Coolant System (RCS) and the potential for a plant upset that could challenge safety systems under conditions to which this Specification applies. The use and interpretation of specified times to complete the actions of LCO 3.0.3 shall be consistent with the discussion of Specification 1.3, "Completion Times."

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

a. The LCO is now met;

b. Remedial measures have restored the facility to an LCO Condition for which the Required Actions have now been performed, where such ACTIONS permit operation in that Condition for either a limited or unlimited period of time; or

c. Remedial measures have restored the facility to a Condition for which the Completion Times of the Required Action(s) have not expired. For example, while in MODE 1, one of the two Emergency Core Cooling System (ECCS) trains is declared inoperable. The corresponding Condition for one inoperable train is entered and 72 hours are allowed to restore the train to OPERABLE status. Then, the second train is declared inoperable at a time 24 hours into the Completion Time. Since no ACTIONS Condition is provided for both trains being inoperable, LCO 3.0.3 must be entered. If one of the trains is made OPERABLE while still in MODE 1, for example, at time 30 hours (6 hours into LCO 3.0.3), then the shutdown may be halted and operation can continue in the

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BASES (continued)

LCO 3.0.3 (continued)

Condition of one train being inoperable. In this example, that would mean operation for another 42 hours. If the train is restored to OPERABLE status after going to MODE 2, 3, or 4, operation could continue only in the MODE that the facility is in when LCO 3.0.3 is exited. This is because LCO 3.0.4 does not permit MODE changes when the LCO is not met.

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

In MODES 1, 2, 3, and 4, LCO 3.0.3 provides Required Actions for Conditions not stated in other specifications. The requirements of LCO 3.0.3 do not apply in MODES 5 and 6 because the facility is already in the most restrictive Condition in which LCO 3.0.3 would require the facility to be placed. 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. [This must be verified by review of all LCOs when finalized.]

The exceptions to LCO 3.0.3 are provided in instances where requiring a facility shutdown, in accordance with LCO 3.0.3, would not provide appropriate remedial measures for the associated condition of the facility. These exceptions are addressed in the individual specifications.

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BASES (continued)

LCO 3.0.4

LCO 3.0.4 establishes limitations on changes in MODES or other specified Conditions in the Applicability when an LCO is not met. It precludes placing the facility in a different MODE or other specified condition when the following exists:

a. The requirements of an LCO in the MODE or other specified Condition to be entered are not met; and

b. Continued noncompliance with these requirements would eventually result in a shutdown to comply with the Required Actions.

Compliance with Required Actions that permit continued operation of the facility for an unlimited period of time in an applicable MODE or other specified Condition provides an acceptable level of safety for continued operation. Therefore, in such cases, entry into a MODE or other Condition specified in the Applicability is 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 good practice in restoring systems or components to OPERABLE status before facility startup.

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.

Exceptions to LCO 3.0.4 are stated in the individual specifications. Exceptions may apply to all the ACTIONS or to a specific Required Action of a specification. While entering or changing MODES or other specified conditions during operation of the facility in an ACTIONS Condition, as permitted by LCO 3.0.4 or where an exception to LCO 3.0.4 is stated, the ACTIONS define the remedial measures that must be taken. Surveillances do not have to be performed on the associated inoperable equipment (or on variables outside the specified limits), as permitted by SR 3.0.1. Therefore, a MODE change in this situation does not violate SR 3.0.4 for those Surveillances that do not have to be performed due to the associated inoperable equipment, etc. SRs must, however, be met to demonstrate OPERABILITY prior to declaring the affected equipment OPERABLE (or variable within limits) and the associated LCOs met.

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BASES (continued)

LCO 3.0.5

Special tests and operations are required at various times over the facility's life to demonstrate performance characteristics, to perform maintenance activities, and to perform special evaluations. Because TS normally preclude these tests and operations, special test exceptions (STEs) allow specified requirements to be changed or suspended under controlled conditions. STEs are included in applicable sections of the specifications. Unless otherwise specified, all other TS requirements remain unchanged and in effect as applicable. This will ensure that all appropriate requirements of the MODE or other specified Condition not directly associated with or required to be changed or suspended to perform the special test or operation will remain in effect.

The Applicability of an STE LCO represents a Condition not necessarily in compliance with the normal requirements of the TS. Compliance with STE LCOs is optional.

A special test may be performed either under the provisions of the appropriate STE LCO or the other applicable TS requirements. If it is desired to perform the special test under the provisions of the STE LCO, the requirements of the STE LCO shall be followed. This includes the SRs specified in the STE LCO.

Some of the STE LCOs require that one or more of the LCOs for normal operation be met (i.e., meeting the STE LCO requires meeting the specified normal LCOs). The Applicability, ACTIONS, and SRs of the specified normal LCOs, however, are not required to be met in order to meet the STE LCO when it is in effect. This means that, upon failure to meet a specified normal LCO, the associated ACTIONS of the STE LCO apply, in lieu of the ACTIONS of the normal LCO. Exceptions to the above do exist. There are instances when the Applicability of the specified normal LCO must be met, where its ACTIONS must be taken, where certain of its Surveillances must be performed, or where all of these requirements must be met concurrently with the requirements of the STE LCO.

Unless the SRs of the specified normal LCOs are suspended or changed by the STE LCO, those SRs that are necessary to meet the specified normal LCOs must be met prior to performing

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BASES (continued)

LCO 3.0.5 (continued)

the special test. During the conduct of the special test, those Surveillances need not be performed unless specified by the ACTIONS or SRs of the STE LCO.

ACTIONS for STE LCOs provide appropriate remedial measures upon failure to meet the STE LCO. Upon failure to meet these ACTIONS, suspend the performance of the special test and enter the ACTIONS for all LCOs that are then not met. Entry into LCO 3.0.3 may possibly be required, but this determination should not be made by considering only the failure to meet the ACTIONS of the STE LCO.

B 3.0 APPLICABILITY

B 3.0 Surveillance Requirement (SR) Applicability

BASES

SR 3.0.1, SR 3.0.2, SR 3.0.3, and SR 3.0.4	SR 3.0.1, 3.0.2, 3.0.3, and 3.0.4 establish the general requirements applicable to all specifications unless otherwise stated. This includes specifications regarding the programs in Section 5.7.4, "Programs and Manuals," as well as specifications contained in Sections 3.1 through 3.9.

SR 3.0.1

SR 3.0.1 establishes the requirement that SRs must be met during the MODES or other specified Conditions in the Applicability of the LCO, unless otherwise specified in the individual SRs. This specification ensures that Surveillances are performed to verify the OPERABILITY of systems and components, and that variables are within specified limits. Failure to meet an 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 SRs are being met; or

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

Surveillances do not have to be performed when the facility is in a MODE or other specified Condition for which the associated LCO is not applicable, unless otherwise specified. The SRs associated with a special test exception (STE) are only applicable when the STE is used as an allowable exception to the requirements of a specification.

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BASES (continued)

SR 3.0.1 (continued)

Surveillances, including Surveillances invoked by Required Actions, do not have to be performed on inoperable equipment because the ACTIONS define the remedial measures that apply. SRs have to be met in accordance with SR 3.0.2 prior to returning equipment to OPERABLE status.

Upon completion of maintenance, appropriate post-maintenance testing (which usually includes Surveillance testing) is required to declare equipment OPERABLE. Post-maintenance testing may not be possible in the MODE or Condition that the facility is in when the maintenance is completed because the necessary facility parameters have not been established. In these situations, proceeding to the appropriate appl.cable MODE or other specified Condition may be allowed as an exception to SR 3.0.4, provided that such an exception is stated in the requirements of the affected equipment's LCO. Such exceptions to SR 3.0.4 are permitted, provided that the post-maintenance and Surveillance testing to demonstrate OPERABILITY of the equipment has been satisfactorily completed to the extent possible and provided that the equipment is not otherwise suspected of being incapable of performing its intended function. Once the necessary facility parameters have been established, completion of the excepted tests must be accomplished to demonstrate OPERABILITY of the equipment.

SR 3.0.2

SR 3.0.2 establishes the requirements for meeting the specified Frequency for SRs, the Required Actions that call for the performance of a Surveillance, and any Required Action with a Completion Time that requires the periodic performance of an action on a "once per..." interval.

SR 3.0.2 permits a 25% extension of the interval specified in the Frequency or periodic Completion Time. This provides flexibility to Surveillance scheduling by providing the opportunity for consideration of plant operating conditions that may not be suitable for conducting the Surveillance (e.g., transient conditions or other ongoing Surveillance or maintenance activities).

The 25% extension does not significantly degrade the assurance of reliability obtained by performing the Surveillance at its specified Frequency. This recognizes

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BASES (continued)

SR 3.0.2 (continued)

that the most probable result of any particular Surveillance being performed is the verification of conformance with the SRs. The exceptions to SR 3.0.2 are those Surveillances for which the 25% 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 a Surveillance with a Frequency of "in accordance with 10 CFR 50, Appendix J, and approved exemptions." The requirements of regulations take precedence over the Technical Specifications (TS). The TS cannot extend a test interval specified in the regulations. Therefore, there would be a Note in the Frequency stating, "Provisions of SR 3.0.2 are not applicable."

As stated in SR 3.0.2, the 25% extension also does not apply to the initial portion of a periodic Completion Time. The initial performance of the Required Action, whether it is a particular Surveillance or some other remedial action, is considered a single action with a single Completion Time. One reason for not allowing the 25% 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 to ensure that specified limits or conditions of the LCO are met.

The previous Standard Technical Specifications (STS) also contained a specification that permitted the 25% extension, but restricted the combined time interval for any three consecutive Surveillance intervals to 3.25 times the specified interval. Generic Letter 89-14 (Ref. 1) encouraged licensees to request license amendments to remove the 3.25 restriction, because the NRC staff concluded that the removal would result in a greater benefit to safety. This line-item improvement to the STS did not extend the Applicability of the 25% extension to intervals associated with LCO Required Actions (including Required Actions to perform Surveillances) specified for periodic performance. The NRC staff subsequently concluded, however, that extending the applicability of the 25% extension to periodic Completion Times, as SR 3.0.2 does, was also justified because the reasons for doing so were essentially the same as the reasons that originally justified the 25% extension

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BASES (continued)

SR 3.0.2 (continued)

(i.e., flexibility for scheduling the performance of Surveillances, etc.). Extending periodic Completion Time intervals for performing Surveillances or repetitive remedial actions specified by ACTIONS can result in a benefit to safety when the performance is due at a time that is not suitable because of plant operating conditions, for example.

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

SR 3.0.3

SR 3.0.3 establishes the option to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a Surveillance has not been completed within the specified Frequency. A delay period of up to 24 hours applies from the point in time that it is discovered that the Surveillance has not been performed, in accordance with SR 3.0.2, and not at the time that the specified Frequency was not met. This 24-hour delay period was approved by the NRC as a line-item improvement to the STS in Generic Letter 87-09 (Ref. 2). The length of the delay period in SR 3.0.3 differs from the 24-hour allowance in the generic letter. SR 3.0.3 limits it to 24 hours or the specified Surveillance interval, whichever is shorter. Although the 24-hour allowance is not applicable to all the cases apparently provided for in the generic letter, the intent of the generic letter was to only allow the specified Surveillance interval in which to complete a missed Surveillance when the Frequency is less than 24 hours.

This delay period provides an adequate time limit to complete Surveillances that have been missed. This delay period provides the opportunity to complete a Surveillance that otherwise could not be completed before compliance with ACTIONS would be required and when compliance with such ACTIONS would then preclude completion of the Surveillance.

The basis for this delay period includes consideration of facility conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, and the safety significance of the delay in completing the

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BASES (continued)

SR 3.0.3 (continued)

Surveillance. The delay period is considered appropriate for balancing the risk associated with delaying completion of the Surveillance for this period against the risk associated with the potential for a plant transient and challenge to safety systems when the alternative is a shutdown to comply with ACTIONS before the Surveillance can be completed.

SR 3.0.3 differs from the position taken in Generic Letter 87-09 in one other respect. Unlike the generic letter, SR 3.0.3 authorizes the delay-period option for performance of missed Surveillances without respect to the duration of the Completion Time associated with the LCO Condition that would otherwise be entered.

When a Surveillance with a Frequency based not on time intervals, but upon specified facility Conditions or operational situations, is discovered not to have been performed when specified, SR 3.0.3 allows the full 24-hour delay period in which to perform the Surveillance.

An additional application of SR 3.0.3 is to establish a time limit for completion of Surveillances that become applicable as a consequence of MODE changes imposed by Required Actions, when such Surveillances could not be completed prior to entering the applicable MODE or other specified Condition either because there was insufficient time or because plant Conditions were not suitable for performance of the Surveillance.

The provisions of SR 3.0.3 exist because it is recognized that the most probable result of the performance of a particular Surveillance is the verification of conformance with the SRs and that a facility shutdown entails some risk that ought to be avoided unless a shutdown is actually warranted. Implementation of the provisions of SR 3.0.3, however, does not imply that a violation of SR 3.0.1 has not occurred, except in situations where SRs become applicable as a consequence of MODE changes imposed by Required Actions, as described above.

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 optional and is expected only under extreme circumstances.

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BASES (continued)

SR 3.0.3 (continued)

If a Surveillance 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 is failed within the delay period, then the equipment is inoperable, or the variable is outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon the failure of the Surveillance.

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

SR 3.0.4

SR 3.0.4 establishes the requirement that all SRs associated with an LCO and all applicable Section 5.7.4 program requirements must be met before entry into a MODE or other specified Condition in the Applicability of the LCO. Thus, prior to entry into an applicable MODE or other specified Condition, all of the SRs associated with all of the LCOs applicable in that MODE or Condition must be met.

This specification ensures that requirements on system and component OPERABILITY and variable limits that are necessary for safe operation of the facility are met before entry into an applicable MODE or other specified Condition to which the requirements apply. This specification applies to changes in MODES or other specified Conditions in the Applicability associated with facility shutdown as well as startup.

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.

Exceptions to SR 3.0.4 are needed in several situations. Because the concerns of each situation are not the same, the conditions under which the exceptions are permitted are different. Briefly, these situations are as follows:

a. When there is insufficient time to complete a Surveillance prior to the associated LCO becoming

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BASES (continued)

SR 3.0.4 (continued)

applicable as a result of complying with ACTIONS, the provisions of SR 3.0.3 apply; and

b. When an individual exception to SR 3.0.4 is stated in the individual specification:

1. if the Surveillance is required to be performed, after entry into an applicable MODE or other specified Condition, because the specified Surveillance interval expired, and there is no other reason to suspect that the affected equipment (or variable) is inoperable (or outside limits), then a Completion Time of 12 hours is specified.

   Unless otherwise stated, performance of the Surveillance is not required if the specified Surveillance interval has not expired.

2. if the Surveillance is required by the specified Frequency to be performed every time the LCO becomes applicable, then, unless an alternative Completion Time is specified, the 12-hour limit applies.

3. if the Surveillance must be performed for the additional purpose of restoring the affected equipment (or variable) to OPERABLE status (or to within limits), upon entering an applicable MODE or other specified Condition, the associated ACTIONS of the LCO must be entered, unless specified otherwise in the individual specification. The ACTIONS specify the Completion Time allowed.

A more detailed discussion of these situations follows.

If unable to complete a Surveillance prior to its becoming applicable because Required Actions in an LCO affected changes in MODES or other specified Conditions, then upon entering the applicable MODE or other specified Condition, a delay period within which to complete the Surveillance is allowed, as specified in SR 3.0.3. This use of the provisions of SR 3.0.3 is an exception to SR 3.0.4 that applies only when an exception to SR 3.0.4 is not

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BASES (continued)

SR 3.0.4 (continued)

provided in the individual specification, as discussed below. The exception of SR 3.0.3 is not intended to be used consecutively with exceptions to SR 3.0.4 stated in the individual specifications.

Individual exceptions to SR 3.0.4 are usually stated with the SRs. These exceptions are provided to permit performance of Surveillance testing that otherwise would be prevented by compliance with SR 3.0.4. The prerequisite conditions for such a Surveillance (usually specified in the Surveillance test procedure) require entry into an applicable MODE or specified Condition in order to perform or complete the Surveillance test. If an exception to SR 3.0.4 is stated in an individual specification, a Completion Time of 12 hours, which begins upon entering the pre-requisite MODE or Condition, is specified by SR 3.0.4 for performing the Surveillance when the specified Surveillance interval has expired (including the 25% extension), unless otherwise specified. It is expected that the performance of such Surveillances will commence soon after entry into the prerequisite MODE or other specified Condition. Use of the entire 12-hour Completion Time interval is expected to occur infrequently. The 12 hours provide sufficient operational flexibility, so the 25% extension allowed by SR 3.0.2 is not needed and therefore does not apply.

This 12-hour Completion Time applies when there is no reason to conclude that the affected equipment is inoperable, or the variable is outside specified limits other than the expiration of the Surveillance interval specified by the Frequency. If still within the Surveillance interval, the Surveillance is still considered to be met and does not have to be performed solely because its LCO becomes Applicable. The 12-hour Completion Time also applies to those Surveillances that are specified to be performed only one time after the prerequisite conditions have been just established (i.e., Surveillances that do not have a periodic Frequency specified). If 12 hours is not an appropriate Completion Time for a Surveillance that has an exception to SR 3.0.4 stated in the individual specification, then the stated exception to SR 3.0.4 specifies an alternative Completion Time, which should be followed. If an alternative Completion Time is not specified, then the

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BASES (continued)

SR 3.0.4 (continued)

12-hour Completion Time applies. In the event the Surveillance is failed, compliance with the ACTIONS of the LCO is required.

The 12-hour Completion Time does not apply when performance of the Surveillance is necessary to establish the affected equipment's OPERABILITY as follows:

a. The equipment was declared inoperable for reasons other than the Surveillance interval expired; or

b. It is necessary to establish that the affected variable is restored to within limits after the variable was known to be outside limits.

In such situations, prior to entering a MODE or other specified Condition in the Applicability of the LCO, appropriate measures must be taken to provide reasonable assurance that the affected equipment or variable is able to meet the requirements of the Surveillance. For example, post-maintenance testing of equipment may not demonstrate OPERABILITY of the equipment with as much assurance as the Surveillance testing does, but it could be an appropriate measure to provide assurance that the Surveillance will be passed. In some cases, appropriate measures could include partial or complete performance of the Surveillance using suitably revised acceptance criteria, if necessary.

It must be emphasized that entry into an applicable MODE or specified Condition, when the affected equipment is known to be inoperable or when the affected variable is known to be outside specified limits, is not permitted by any exception to SR 3.0.4 that is stated in an individual specification. There must first be a reasonable expectation that performance of the Surveillance will establish that the equipment is OPERABLE or that the variable is within specified limits. At the time the associated LCO becomes applicable (because of entry into an applicable MODE or specified Condition from a non-applicable MODE or Condition), the ACTIONS of the LCO must be entered for the Condition corresponding to the affected equipment or variable being inoperable or outside specified limits. The SR must be met and the entered Conditions corrected prior to

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BASES (continued)

SR 3.0.4 (continued)

expiration of the specified Completion Time. Any associated Required Actions other than the Action to restore the equipment to OPERABLE status or to return the variable to within the specified limits must be accomplished within the specified Completion Times until the entered Condition is corrected. In the event the Surveillance is failed, compliance with the ACTIONS of the LCO is required. The Completion Time clock (that began when the LCO became applicable and is associated with the Required Action to correct the entered Condition) does not reset upon failure of the Surveillance.

REFERENCES

1. NRC Generic Letter 89-14, "Line-Item Improvements in Technical Specifications - Removal of 3.25 Limit on Extending Surveillance Intervals," August 21, 1989.

2. NRC Generic Letter 87-09, "Sections 3.0 and 4.0 of the Standard Technical Specifications (STS) on the Applicability of Limiting Conditions for Operation and Surveillance Requirements," June 4, 1987.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.1 SHUTDOWN MARGIN (SDM)

BASES

BACKGROUND

The reactivity control system must be redundant and capable of holding the reactor core subcritical when shutdown under cold conditions (GDC 26, Ref. 1). Maintenance of the SDM ensures that postulated reactivity events will not damage the fuel. SDM requirements provide sufficient reactivity margin to ensure that acceptable fuel design limits will not be exceeded for normal shutdown and anticipated operational occurrences (AOOs). As such, in MODES 1 and 2 the SDM defines the degree of subcriticality which would be obtained immediately following the insertion or scram of all safety and regulating rods, assuming the single rod cluster assembly of highest reactivity worth is fully withdrawn. In MODES 3, 4, and 5, the SDM specified continues to provide for adequate shutdown capability and acceptable fuel design limits for potential accidents initiated from shutdown conditions.

The system design requires that two independent reactivity control systems be provided and that one of these systems be capable of maintaining the core subcritical under cold conditions. These requirements are provided by the use of movable control assemblies and soluble boric acid in the Reactor Coolant System (RCS). The CONTROL ROD System can compensate for the reactivity effects of the fuel and water temperature changes accompanying power level changes over the range from full-load to no-load. In addition, the CONTROL ROD System together with the Boration System provides SDM during power operation and is capable of making the core subcritical rapidly enough to prevent exceeding acceptable fuel damage limits, assuming that the rod of highest reactivity worth remains fully withdrawn.

The soluble Boron System can compensate for fuel depletion during operation and all xenon burnout reactivity changes and maintain the reactor subcritical under cold conditions.

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BASES (continued)

BACKGROUND (continued)

During power operation, SDM control is ensured by operating with the safety rods fully withdrawn (LCO 3.1.5) and the regulating rods within the limits of LCO 3.2.1. When in the shutdown and refueling MODES, the SDM requirements are met by adjustments to the RCS boron concentration.

APPLICABLE SAFETY ANALYSES

The minimum required SDM is assumed as an initial condition in safety analysis. The safety analysis (Ref. 2) establishes an SDM that ensures that specified acceptable fuel design limits are not exceeded for normal operation and AOOs with assumption of the highest worth rod stuck out on scram.

The acceptance criteria for SDM are that specified acceptable fuel design limits are maintained by ensuring:

a. The reactor can be made subcritical from all operating conditions and transients and Design Basis Events (DBEs);

b. The reactivity transients associated with postulated accident conditions are controllable with acceptable limits (departure from nucleate boiling ratio (DNBR), fuel centerline temperature limits for AOOs, and $\leq$ 280 cal/gm energy deposition for the rod ejection accident); and

c. The reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

The most limiting accident for the SDM requirements is based on a main steam line break (MSLB) as described in the accident analysis (Ref. 2). The increased steam flow resulting from a pipe break in the main steam system causes an increased energy removal from the affected steam generator (SG), and consequently the RCS. This results in a reduction of the reactor coolant temperature. The resultant coolant shrinkage causes a reduction in pressure. In the presence of a negative moderator temperature coefficient, this cooldown causes an increase in core reactivity. As

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

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 SG boils dry, thus terminating RCS heat removal and cooldown. Following the MSLB, a post-trip return to power may occur, however no fuel damage occurs as a result of the post-trip return to power and the 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 must also protect against:

a. Inadvertent boron dilution;

b. An uncontrolled rod withdrawal from a subcritical or low power condition;

c. Startup of an inactive reactor coolant pump (RCP); and

d. Rod ejection.

Each of these events is discussed below.

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

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

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

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

The withdrawal of rods from subcritical 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 rods also produces a time dependent redistribution of core power.

SDM satisfies Criterion 2 of the NRC Interim Policy Statement. Even though it is not directly observed from the control room, SDM is considered an initial condition process variable because it is periodically monitored to provide assurance that the unit is operating within the bounds of accident analysis assumptions.

LCO

The accident analysis has shown that the required SDM is sufficient to avoid unacceptable consequences to the fuel or RCS as a result of the events addressed above. Shutdown boron concentration requirement assumes the highest worth rod is stuck in the fully withdrawn position to account for a postulated inoperable or untrippable rod prior to reactor shutdown.

SDM is a core design condition that can be ensured during operation through control rod positioning (control and shutdown banks) and through the soluble boron concentration. To ensure that SDM is behaving as anticipated so that the acceptance criteria are met, the SDM is evaluated during SR 3.1.1.1 and appropriate actions are taken as necessary.

The MSLB (Ref. 2) and the boron dilution (Ref. 3) accidents are the most limiting analyses which established the SDM value of the LCO.

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BASES (continued)

LCO (continued)

For MSLB accidents, if the LCO is violated, there is a potential to exceed the DNBR limit and to exceed 10 CFR 100 limits. SDM is a core physics design condition that can be ensured during operation through rod positioning (control and safety rods) and through the soluble boron concentration. To ensure that SDM is behaving as anticipated so that the acceptance criteria are met, the SDM is evaluated during SR 3.1.1.1 and appropriate actions are taken as necessary when the SDM is not within the required limit. For the boron dilution accident, if the LCO is violated, then the minimum required time assumed for operator action to terminate dilution may no longer be applicable.

APPLICABILITY

In MODES 1, 2, 3, 4, and 5, the SDM requirements are applicable to provide sufficient negative reactivity to meet the assumptions of the safety analysis discussed above. In MODE 6, the shutdown reactivity requirements are given in LCO 3.9.1, "Boron Concentration."

ACTIONS

A.1

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

In the determination of the required combination of boration flow rate and boron concentration, there is no unique DBF that must be satisfied. It is imperative to raise the boron concentration of the RCS as soon as possible.

Therefore, the operator should begin boration with the best source available for the plant conditions. Some of the possible sources of boron originate from either the boric acid storage tank (BAST), whose minimum concentration of boron is [11600]ppm, or the borated water storage tank (BWST), whose minimum concentration of boron is [2270]ppm. These sources include:

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BASES (continued)

ACTIONS (continued)

a. Makeup flow through makeup pumps from makeup tank: Makeup pumps are rated at [300]gpm at [2400]psig. Boron concentration of the makeup tank varies with the time in life and the concentration in the RCS;

b. Makeup flow through makeup pumps from BWST: Makeup pumps are rated at [300]gpm at [2400] psig;

c. Makeup flow through makeup pumps from BAST: Makeup pumps are rated at [300]gpm at [2400] psig;

d. High pressure injection through makeup pumps from BWST: Makeup pumps are rated at [500]gpm at [600]psig;

e. Decay heat flow throu decay heat pumps from BWST: Decay heat pumps are ted at [3000]gpm at [100] psig;

f. Low pressure injection through decay heat pumps from BWST: Decay heat pumps are rated at [3000]gpm at [100] psig; and

g. Boric acid through boric acid pumps from BAST: Boric acid pumps are rated at [25]gpm at [100] psig.

In determining the boration flow rate, it should be remembered that the most difficult time in core life to increase the RCS boron concentration is at beginning of cycle when the boron concentration may approach or exceed [2000]ppm.

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1

In MODES 1 and 2, SDM is verified by observing that the requirements of LCO 3.1.5, "Safety Rod Insertion Limits," and LCO 3.2.1, "Regulating Rod Insertion Limits," are met. However, in the event that a rod is known to be untrippable, SDM verification must account for the worth of the untrippable rod as well as another rod of maximum worth.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

In MODES 3, 4, and 5, the SDM is verified by performing a reactivity balance calculation, considering the listed reactivity effects:

a. RCS boron concentration;

b. Regulating rod position;

c. RCS average temperature;

d. Fuel burnup based on gross thermal energy generation;

e. Xenon concentration;

f. Samarium concentration; and

g. Isothermal Temperature Coefficient (ITC).

Using the ITC accounts for Doppler reactivity in this calculation because the reactor is subcritical and the fuel temperature will be changing at the same rate as the RCS. The 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, including a boron concentration analysis, and complete the calculation.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix A, General Design Criterion 26, "Reactivity Control System Redundancy and Capability."

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

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

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.2 Reactivity Balance

BASES

BACKGROUND

Per GDC 26, 28, and 29 (Ref. 1), reactivity shall be controllable such that subcriticality is maintained under cold conditions and acceptable fuel design limits are not exceeded during normal operation and anticipated operational occurrences. Therefore, reactivity balance is used as a measure of the predicted versus measured core reactivity during power operation. The periodic confirmation of core reactivity is necessary to ensure that safety analyses of design basis transients and accidents remain valid. A large reactivity difference could be the result of unanticipated changes in fuel, or control rod worth, or operation at conditions not consistent with those assumed in the predictions of core reactivity, and which could potentially result in a loss of SHUTDOWN MARGIN (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 assuring the reactor can be brought safely to cold, subcritical conditions.

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

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(continued)

BASES (continued)

BACKGROUND (continued)

reactivity is within design expectations, and that the calculation models used to generate the safety analysis within design expectations, and that the calculation models used to generate the safety analysis are adequate.

In order to achieve the required fuel cycle energy output, the uranium enrichment in the new fuel loading and 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 RATED THERMAL POWER (RTP) and moderator temperature, the excess positive reactivity is compensated by burnable absorbers (if any), control rods, whatever neutron poisons (mainly xenon and samarium) are present in the fuel, and the RCS boron concentration.

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

APPLICABLE SAFETY ANALYSES

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

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

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

reactivity balance provides additional assurance that the nuclear methods provide an accurate representation of the core reactivity.

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

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

Reactivity balance provides an additional assurance that SDM is maintained within the limits. Thus, reactivity balance satisfies Criterion 2 of the NRC Interim Policy Statement.

LCO

This specification is provided to ensure that core reactivity behaves as expected in the long term and to ensure that significant reactivity anomalies will be investigated. Long-term core reactivity behavior is a

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BASES (continued)

LCO (continued)

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

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

APPLICABILITY

In MODE 1, most of the control rods are withdrawn and steady-state operation is typically achieved. Under these conditions, the comparison between predictions and measurements provides an effective measure of the reactivity balance. In MODE 2, control rods are typically being withdrawn during a startup. In MODES 3, 4, and 5, all control rods are fully inserted and therefore the reactor is in the least reactive state where monitoring core reactivity is not necessary. 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 and a SDM demonstration is required during the first startup following operations which could have altered core reactivity (e.g., fuel movement or control rod replacement or shuffling).

(continued)

BASES (continued)

ACTIONS

A.1

Should an anomaly develop between measured and predicted core reactivity, an evaluation of the core design and safety analysis is performed. In practice, larger deviations in core reactivity (greater than 0.5% Δ k/k) are generally cause for concern, and evaluation of both core conditions and the core design are performed to determine the cause of the deviation.

When a reactivity deviation is noted, the evaluation of core conditions typically includes the following steps:

a. Core conditions and the input to calculational models are verified to be consistent;

b. Shutdown capability from both the CONTROL RODS and the Boron Injection System is determined to be adequate;

c. A core power distribution map is obtained to evaluate peaking factors;

d. OPERABILITY of all CONTROL RODS is verified; and

e. Physical changes in the fuel or boron content of the RCS are considered.

An evaluation of the core design and safety analysis typically includes the following steps:

a. Reactivity worth calculations of boron, the CONTROL RODS, xenon, and samarium are reviewed;

b. The moderator and fuel temperature coefficient calculations are reviewed and verified to be within the bounds of the safety analysis;

c. The fuel depletion calculations are reviewed to determine that the calculated core burnup is appropriate; and

d. The calculation models are reviewed to verify that they are adequate for representation of the core conditions.

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BASES (continued)

ACTIONS (continued)

Reactivity anomalies are generally investigated when they are small, so that the evaluations are in progress before the 1% Δ k/k reactivity limit for a deviation is reached, and corrective measures may be defined. The required Completion Time of 72 hours is based on operating experience and the low probability of a Design Basis Accident occurring during this period. Also, it allows sufficient time to assess the physical condition of the reactor and complete an evaluation of the core design and safety analysis.

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

B.1

The unit must be placed in a MODE in which the LCO does not apply if the core reactivity cannot be restored to within the 1% Δ k/k limit by the methods discussed in Required Action A.1 and the associated Completion Time. This is done by placing the unit in at least MODE 3 within 6 hours. If the SDM for MODE 3 is not met, then boration required by SR 3.1.1.1 would occur. The allowed Completion Time is reasonable, based on operating experience related to the time required, to reach the required plant conditions from full power in an orderly manner and without challenging plant systems.

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BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1

Core reactivity is verified by periodic comparisons of measured and predicted RCS boron concentrations. The comparison is made considering that other core conditions are fixed or stable including CONTROL RODS position, moderator temperature, fuel temperature, fuel depletion, xenon concentration, and samarium concentration. The surveillance is performed prior to entering MODE 1 as an initial check on core conditions and design calculations at BOC. A Note is included in the SR to indicate that the normalization of predicted core reactivity to the measured value must take place within the first 60 effective full power days (EFPDs) 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 EFPDs after the initial 60 EFPDs after entering MODE 1 is acceptable based on the slow rate of core changes due to fuel depletion and the presence of other indicators (quadrant power tilt, etc.) for prompt indication of an anomaly. Another Note is included in the SRs to indicate that the provisions of SR 3.0.4 are not applicable for changing MODES.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix A, General Design Criterion 26, "Reactivity Control System Redundancy and Capability"; General Design Criterion 28, "Reactivity Limits"; General Design Criterion 29, "Protection Against Anticipated Operational Occurrences."

2. [Unit Name] FSAR, Section [ ], "[Accident Analysis]."

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.3 Moderator Temperature Coefficient (MTC)

BASES

BACKGROUND

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

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

MTC values are predicted at selected burnups during the safety evaluation analysis and are confirmed to be acceptable by measurements. Both initial and reload cores are designed so that the beginning of cycle (BOC) MTC is less than zero when THERMAL POWER is [95%] of RATED THERMAL POWER (RTP) or greater. 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 (lumped burnable poison assemblies) to yield an MTC at 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 designed to achieve high burnups or with changes to other characteristics are evaluated to ensure the MTC does not exceed the EOC limit.

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APPLICABLE SAFETY ANALYSES

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

The acceptance criteria for the specified MTC are:

a. The MTC values must remain within the bounds of those used in the accident analysis (Ref. 2); and

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

Accidents that cause core overheating (either 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 CONTROL ROD 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 positive MTC is a [rod withdrawal accident from zero power, also referred to as a startup accident (Ref. 4)].

Accidents that cause core overcooling must be evaluated for results when the MTC is most negative. The event which 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 CONTROL ROD assemblies inserted except the most reactive one. Even if the reactivity increase produces slightly subcritical conditions, a large fraction of core

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

power may be produced through the effects of subcritical neutron multiplication.

MTC 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 300 ppm. The measured value may be extrapolated to project the EOC value, in order to confirm reload design predictions.

MTC satisfies Criterion 2 of the NRC Interim Policy Statement. Even though it is not directly observed and controlled from the control room, MTC is considered an initial condition process variable because of its dependence on boron concentration.

LCO

LCO 3.1.3 requires the MTC to be within specified limits in the CORE OPERATING LIMITS REPORTS (COLR)(Ref. 5) 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 of +0.9E-4 (%Δk/k)/°F on positive MTC when THERMAL POWER is less than 95% of RTP assures that core overheating accidents will not violate the accident analysis assumptions. The requirement for a negative MTC when THERMAL POWER is 95% of RTP or greater ensures that core operation will be stable. The negative MTC limit for EOC specified in the COLR ensures that core overcooling accidents will not violate the accident analysis assumptions. MTC 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 conditions of the LCO can only be ensured through measurement. The surveillance checks at BOC and EOC on MTC provide confirmation that the MTC is behaving as anticipated so that the acceptance criteria are met.

APPLICABILITY

In MODE 1, the limits on MTC must be maintained to assure that any accident initiated from THERMAL POWER operation will not violate the design assumptions of the accident

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BASES (continued)

APPLICABILITY (continued)

analysis. In MODE 2, the limits must also be maintained to ensure startup and subcritical accidents (such as the uncontrolled CONTROL ROD assembly 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 Design Basis Accidents (DBAs) using the MTC as an analysis assumption are initiated from these MODES. However, the variation of MTC with temperature in MODES 3, 4, and 5 for DBAs initiated in MODES 1 and 2 is accounted for in the subject accident analysis. The variation of MTC with temperature assumed in the safety analysis is accepted as valid once the BOC and MOC measurements are used for normalization.

ACTIONS

A.1

MTC is a function of the fuel and fuel cycle design and cannot be controlled directly once their designs have been implemented in the core. If MTC exceeds its limits, the reactor must be placed in MODE 3 with a minimum SHUTDOWN MARGIN. 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 which would require a MTC value within the LCO limits, and the length of time required to reach MODE 3 conditions from full power in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.3.1

The SRs for measurement of the MTC at the beginning and end of each fuel cycle provide for confirmation of the limiting MTC values. The MTC changes slowly from most positive (least negative) to most negative value during fuel cycle operation as the RCS boron concentration is reduced with fuel depletion. The requirement for measurement prior to initial operation above 5% of RTP satisfies the confirmatory check on the most positive (least negative) MTC value. The requirement for measurement within 7 effective full power days (EFPDs) after reaching an equilibrium boron concentration of 300 ppm for RTP satisfies the confirmatory

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

check on most MTC value. The measurement is performed at any THERMAL POWER equivalent to an RCS boron concentration of 300 ppm (for steady-state operation at RTP with all CONTROL RODS fully withdrawn) so that the projected EOC MTC may be evaluated before the reactor actually reaches the EOC condition. MTC values may be extrapolated and compensated to permit direct comparison to the specified MTC limits.

Th : is modified by a Note that states that SR 3.0.4 is not applicable for entering MODE 2. Although this surveillance is applicable in MODE 2, the reactor must be critical before the surveillance can be completed. Therefore, entry into the applicable MODE prior to accomplishing the surveillance is necessary.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix A, General Design Criterion 11, "Reactor Inherent Protection."

2. [Unit Name] FSAR, Section [14], "[Safety Analysis]."

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

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

5. [Unit Name] Core Operating Limits Report, "[Title]."

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.4 CONTROL ROD Alignment Limits

BASES

BACKGROUND

The OPERABILITY of the CONTROL RODS (safety rods and regulating rods) are initial assumptions in all safety analyses which assume rod insertion upon reactor trip. Maximum rod misalignment is an initial assumption in the safety analysis which directly affects core power distributions and assumptions of available SHUTDOWN MARGIN (SDM).

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

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

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

CONTROL RODS are moved by their control rod drive mechanisms (CRDMs). Each CRDM moves its rod one step [$\frac{3}{4}$ inch for one revolution of the leadscrew] at a time but at varying rates (steps per minute) depending on the signal output from the Rod Control System.

The CONTROL RODS are arranged into rod groups that are radially symmetric. Therefore, movement of the CONTROL RODS does not introduce radial asymmetries in the core power distribution. The safety rods provide required reactivity

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BASES (continued)

BACKGROUND (continued)

worth for immediate reactor shutdown upon a reactor trip. The regulating rods provide reactivity (power level) control during normal operation and transients, and their movement is normally governed by the Automatic Control System.

The axial position of safety rods and regulating rods is indicated by two separate and independent systems, which are the relative position indicator transducers, and the absolute position indicator transducers (see LCO 3.1.7).

The relative position indicator transducer is a potentiometer that is driven by electrical pulses from the Rod Control System that moves the rods. There is one counter for each group of rods. Individual rods in a group all receive the same signal to move and should, therefore, all be at the same position indicated by the group counter for that group. The relative position indicator system is considered highly precise (one rotation of the leadscrew is $\frac{3}{4}$ inch in rod motion). If a rod does not move for each demand pulse, the counter will still count the pulse and incorrectly reflect the position of the rod.

The Absolute Position Indicator System provides a highly accurate indication of actual CONTROL ROD position, but at a lower precision than the step counters. This system is based on inductive analog signals from a series of reed switches spaced along a tube with a center-to-center distance of [3.75 inches], which is [6 steps].

APPLICABLE SAFETY ANALYSES

CONTROL ROD misalignment and inoperability accidents are analyzed in the safety analysis (Ref. 3). The acceptance criteria for addressing CONTROL ROD inoperability or misalignment are that:

a. There be no violations of:

   1. specified acceptable fuel design limits,

   2. centerline fuel temperature,

   3. Reactor Coolant System (RCS) pressure boundary damage; and

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

b. The core must remain subcritical after accident transients.

Three types of misalignment or inoperability are distinguished. During movement of a CONTROL ROD group, one rod may stop moving while the other rods in the group continue. This condition may cause excessive power peaking. The second type of misalignment occurs if one rod fails to insert upon a reactor trip and remains stuck fully withdrawn. This condition requires an evaluation to determine that sufficient reactivity worth is held in the CONTROL RODS to meet the SDM requirement with the maximum worth rod stuck fully withdrawn. If a CONTROL ROD is stuck in the fully withdrawn position, its worth is added to the SDM, since the safety analysis does not take two stuck rods into account. The third type of misalignment occurs when one rod drops partially or fully into the reactor core. This event causes an initial power reduction followed by a return towards the original power due to positive reactivity feedback from the negative moderator temperature coefficient. Increased peaking during the power increase may result in excessive local linear heat rates (LHR).

The accident analysis and reload safety evaluations define regulating rod insertion limits that ensure the required SDM can always be achieved if the maximum worth CONTROL ROD is stuck fully withdrawn (Ref. 4). If a CONTROL ROD is stuck in or dropped in, continued operation is permitted if the increase in local LHR is within the design limits. The Required Action statements in the LCOs provide conservative reductions in THERMAL POWER and verification of SDM to ensure continued operation remains within the bounds of the safety analysis (Ref. 5).

Continued operation of the reactor with a misaligned or dropped CONTROL ROD is allowed if the NUCLEAR HEAT FLUX HOT CHANNEL FACTOR ($F_Q(Z)$) and the NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR ($F^N_{\Delta H}$) are verified to be within their limits in the CORE OPERATING LIMITS REPORT (COLR) (Ref. 6). When a CONTROL ROD is misaligned, the assumptions that are used to determine the regulating rod insertion limits, AXIAL POWER SHAPING ROD (APSR) insertion limits, AXIAL POWER IMBALANCE limits, and QUADRANT POWER TILT (QPT) limits are not preserved. Therefore, the limits may not preserve the design peaking factors, and $F_Q(Z)$ and $F^N_{\Delta H}$ must be verified

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

directly by incore mapping. Section B 3.2, Power Distribution Limits, contains more complete discussion of the relation of $F_Q(Z)$ and $F^N_{\Delta H}$ to the operating limits.

The CONTROL ROD group alignment limits of Specification 3.1.4 are directly related to power peaking and SDM. Power peaking and SDM are process variables that satisfy Criterion 2 of the NRC Interim Policy Statement, since they represent initial conditions input to the plant safety analysis. In addition, the CONTROL RODS satisfy Criterion 3, since they actuate to mitigate transients that challenge the integrity of a fission-product barrier.

LCO

The limits on CONTROL ROD group alignment, safety rod insertion, and APSR alignment, together with the limits on regulating rod insertion, APSR insertion, AXIAL POWER IMBALANCE, and QPT, assure the reactor will operate within the fuel design criteria. The Required Actions in these LCOs assure that deviations from the alignment limits will either be corrected or that THERMAL POWER will be adjusted so that excessive local LHRs will not occur, and that the requirements on SDM and ejected rod worth are preserved.

The limit for individual CONTROL ROD misalignment is 6.5% (9 inches) deviation from the group average position. This value is established based on the distance between reed switches, with additional allowances for uncertainty in the absolute position indicator amplifiers, group maximum or minimum synthesizer, and asymmetric alarm or fault detector outputs. The position of an inoperable rod is not included in the calculation of the rod group average position.

CONTROL RODS are OPERABLE when they can meet the surveillance requirements of this LCO and can be inserted and withdrawn to meet the alignment limits, sequence and overlap withdrawal requirements, rod drop times, and position indication requirements.

[For this facility, an OPERABLE Relative Position Indicator System and Absolute Position Indicator System constitute the following:]

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BASES (continued)

LCO (continued)

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

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

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

APPLICABILITY

The requirements on CONTROL ROD 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 and alignment of rods has the potential to affect the safety of the plant. In MODES 3, 4, 5, and 6, the alignment limits do not apply because the CONTROL RODS are bottomed and the reactor is shutdown and not producing fission power. In the shutdown MODES, the OPERABILITY of the safety and regulating rods has the potential to affect the required SDM, but this effect can be compensated for by an increase in the boron concentration of the RCS. See LCO 3.1.1 for SDM in MODES 3, 4, and 5, and LCO 3.9.1 for boron concentration requirements during refueling.

ACTIONS

(Refer to Figure B 3.1.4-1)

A.1

If a CONTROL ROD is inoperable but trippable or misaligned beyond the specified alignment limit, the first preference is usually to restore it to OPERABLE status within the alignment requirements. A misaligned CONTROL ROD can usually be moved and is still trippable. If the rod can be realigned within its limits within the 1-hour Completion Time, local xenon redistribution during this short interval will not be significant, and operation may proceed without further restriction.

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(continued)

BASES (continued)

ACTIONS (continued)

A.2

Alignment of the inoperable or misaligned CONTROL ROD may be accomplished by either moving the single CONTROL ROD to the group average position or by moving the remainder of the group to the position of the single inoperable or misaligned CONTROL ROD. Either action can be used to restore the CONTROL RODS to a radially symmetric pattern. However, this must be done without violating the CONTROL ROD sequence, overlap, and insertion limits of LCO 3.2.1 ("Regulating Rod Insertion Limits") given in the COLR (Ref. 6). THERMAL POWER must also be restricted, as necessary, to the value allowed by the insertion limits of LCO 3.2.1. The required Completion Time of 1 hour is acceptable because local xenon redistribution during this short interval will not cause a significant increase in LHR. This required Completion Time is more conservative than that required for restoration of the regulating rods to within their limits given in LCO 3.2.1. This option is not available if a safety rod is misaligned, since the limits of LCO 3.1.5 would be violated. It is acceptable to operate with one CONTROL ROD assembly in the fully withdrawn position since this is consistent with all safety analysis and core design calculations.

A.3.1.1

Compliance with Required Actions A.3.1.1 through A.3.5 allows for continued power operation with one CONTROL ROD inoperable but trippable or misaligned from its group average position. These Required Actions comprise the final alternate for Condition A.

If realignment of the CONTROL ROD to the group average or alignment of the group to the misaligned CONTROL ROD is not completed within 1 hour (Required Actions A.1 or A.2 not met), the rod should be considered inoperable. Since the rod may be inserted farther than the group average insertion for a long time, SDM must be evaluated. Assuring the SDM meets the minimum requirement within 1 hour is adequate to determine that further degradation of the SDM is not occurring.

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BASES (continued)

ACTIONS (continued)

A.3.1.2

Restoration of the required SDM requires increasing the RCS boron concentration, since the CONTROL ROD may remain misaligned and not be providing its normal negative reactivity on tripping. RCS boration must occur as described in Section B 3.1.1. The required Completion Time of 1 hour to initiate boration is reasonable based on the time required for potential xenon redistribution, the low probability of an accident occurring, and the steps required to complete the action. This allows the operator sufficient time to align the required valves and start the boric acid pumps. Boration will continue until the required SDM is restored.

A.3.2

Reduction of THERMAL POWER to ≤ 60% of the ALLOWABLE THERMAL POWER ensures that local LHR increases due to a misaligned rod will not cause the core design criteria to be exceeded (Ref. 7). The required Completion Time of 2 hours allows the operator sufficient time to reduce THERMAL POWER.

A.3.3

Reduction of the nuclear overpower trip setpoint to 70% of ALLOWABLE THERMAL POWER after THERMAL POWER has been reduced to 60% of ALLOWABLE THERMAL POWER maintains both core protection and an operating margin at reduced power similar to that at RATED THERMAL POWER (RTP) (Ref. 7). The required Completion Time of 10 hours allows the operator 8 additional hours after completion of the THERMAL POWER reduction in Required Action A.3.2. This allows adequate time to adjust the trip setpoint.

A 3.4

The existing CONTROL ROD configuration must not cause an ejected rod to exceed the limit of 0.65 %Δk/k at RTP or 1.00 %Δk/k at zero power (Ref. 8). This evaluation may require a computer calculation of the maximum ejected rod worth based on non-standard configurations of the CONTROL ROD groups. The evaluation must determine the ejected rod worth for the remainder of the fuel cycle to ensure a valid evaluation, should fuel cycle conditions at some later time

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(continued)

BASES (continued)

ACTIONS (continued)

become more bounding than those at the time of the rod misalignment. The required Completion Time of 73 hours is acceptable because LHRs are limited by the THERMAL POWER reduction and sufficient time is provided to perform the required evaluation.

A.3.5

Performance of SR 3.2.5.1 provides a determination of the power peaking factors using the incore detector system. Verification of the NUCLEAR HEAT FLUX HOT CHANNEL FACTOR ($F_Q(Z)$) and NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR ($F^N_{\Delta H}$) from an incore power distribution map is necessary to ensure that excessive local LHRs will not occur due to CONTROL ROD misalignment. This is necessary because the assumption that all CONTROL RODS are aligned (used to determine the regulating rod insertion, AXIAL POWER IMBALANCE, and QPT limits), is not valid when the CONTROL RODS are not aligned. The required Completion Time of 73 hours is acceptable because LHRs are limited by the THERMAL POWER reduction and adequate time is allowed to obtain an incore power distribution map.

B.1

The plant must be placed in a MODE in which the LCO does not apply if the Required Actions and associated Completion Times for Condition A cannot be met. This is done by placing the plant in at least MODE 3 within 6 hours. The 6 hours allotted to reach MODE 3 is a reasonable time, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging plant systems.

C.1.1

More than one CONTROL ROD becoming inoperable or misaligned, or both inoperable but trippable and misaligned from its group average position, is not expected, and may violate the minimum SDM requirement. Therefore, SDM must be evaluated. Assuring the SDM meets the minimum requirement within 1 hour allows the operator adequate time to determine the SDM.

(continued)

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BASES (continued)

ACTIONS (continued)

C.1.2

Restoration of the required SDM requires increasing the RCS boron concentration to provide negative reactivity. RCS boration must occur as described in Section B 3.1.1. The required Completion Time of 1 hour to initiate boration is reasonable based on the time required for potential xenon redistribution, the low probability of an accident occurring, and the steps required to complete the action. This allows the operator sufficient time to align the required valves and start the boric acid pumps. Boration will continue until the required SDM is restored.

C.2

If more than one CONTROL ROD is inoperable but trippable or misaligned, continued operation of the reactor may cause the misalignment to increase, as the regulating rods insert or withdraw to control reactivity. If the CONTROL ROD misalignment increases, local power peaking may also increase, and local LHRs will also increase if the reactor continues operation at THERMAL POWER. The SDM is decreased when one or more CONTROL RODS become inoperable at a given THERMAL POWER level, or if one or more CONTROL RODS become misaligned by insertion from the group average position.

Therefore it is prudent to place the reactor in MODE 3. LCO 3.1.4 does not apply in MODE 3 since excessive power peaking cannot occur and the minimum required SDM is assured. The Completion Time of 6 hours is consistent with Specification 3.0.3.

D.1.1 and D.1.2

When one or more rods are inoperable to the extent that they are immovable and untrippable, there is a possibility that the required SDM may be adversely affected. Under these conditions it is important to determine the SDM, and if it is less than the required value, initiate boration until the required SDM is recovered. The Completion Time of 1 hour is adequate to determine SDM and, if necessary, to initiate emergency boration and restore SDM.

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BASES (continued)

ACTIONS (continued)

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

D.2

If the inoperable rod(s) cannot be restored to OPERABLE status, the plant must be placed in a MODE or condition in which the LCO requirements are not applicable. This is done by placing the plant in at least MODE 3 within 6 hours.

The allowed Completion Time is reasonable based on operating experience to reach MODE 3 from full power operation in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.4.1

Verification that individual CONTROL ROD AXIAL POWER IMBALANCE positions are within 6.5% of their group average height limits at a 12-hour Frequency allows the operator to detect a rod beginning to deviate from its expected position. If the asymmetric CONTROL ROD alarm is inoperable, a 4-hour Frequency is reasonable to prevent large deviations in CONTROL ROD alignment from occurring without detection. The specified frequency takes into account other rod position information that is continuously available to the operator in the control room so that during actual rod motion, deviations can immediately be detected. For this facility, each ROD AXIAL POWER IMBALANCE is considered inoperable if it has [ ] individual reed switches inoperable.

SR 3.1.4.2

Exercising individual CONTROL RODS every 92 days allows the operator to determine that all rods continue to be OPERABLE, even if they are not regularly moved. Moving each CONTROL ROD by 3% will not cause radial or axial power tilts or oscillations to occur. The intent of this surveillance is to move the rods an amount necessary to detect rod movement, thus confirming their OPERABILITY but without exceeding the alignment limit when only one rod is being moved.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

The 92-day Frequency takes into consideration other information available to the operator in the control room and other surveillances being performed more frequently which add to the determination of OPERABILITY of the rods.

A Note is provided to indicate that CONTROL ROD movement in accordance with this SR does not violate the regulating rod insertion limits or the safety rod insertion limits.

SR 3.1.4.3

Verification of rod drop time allows the operator to determine that the maximum rod drop time permitted is consistant with the assumed rod drop time used in the safety analysis. The rod drop time given in the safety analysis is 1.4 seconds to $\frac{2}{3}$ insertion. Using the identical rod drop curve gives a value of 1.66 seconds to $\frac{3}{4}$ insertion. The latter value is used in the surveillance because the zone reference lights are located at 25% insertion intervals. The zone reference lights will activate at $\frac{3}{4}$ insertion to give an indication of the rod drop time and rod location. Measuring rod drop times prior to reactor criticality after reactor vessel head removal and after CONTROL ROD drive system maintenance or modification assures that the reactor internals and CONTROL ROD drive mechanism will not interfere with CONTROL ROD motion or rod drop time. Individual rods whose drop times are greater than safety analysis assumptions are not OPERABLE. The 18-month Frequency was developed because it was considered prudent that this surveillance only be performed during a plant outage. This is due to the plant conditions needed to perform the SR and the potential for an unplanned plant transient if the surveillance is performed with the reactor at power. Operating experience has shown that these components usually pass this surveillance when performed on the 18-month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

This testing is normally performed with all reactor coolant pumps operating and average moderator temperature $\geq$ 525°F to simulate a reactor trip under actual conditions. However, if the rod drop times are determined with less than four reactor coolant pumps operating, a Note allows power operation to continue provided operation is restricted to the pump combination utilized during the rod drop time determination.

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BASES (continued)

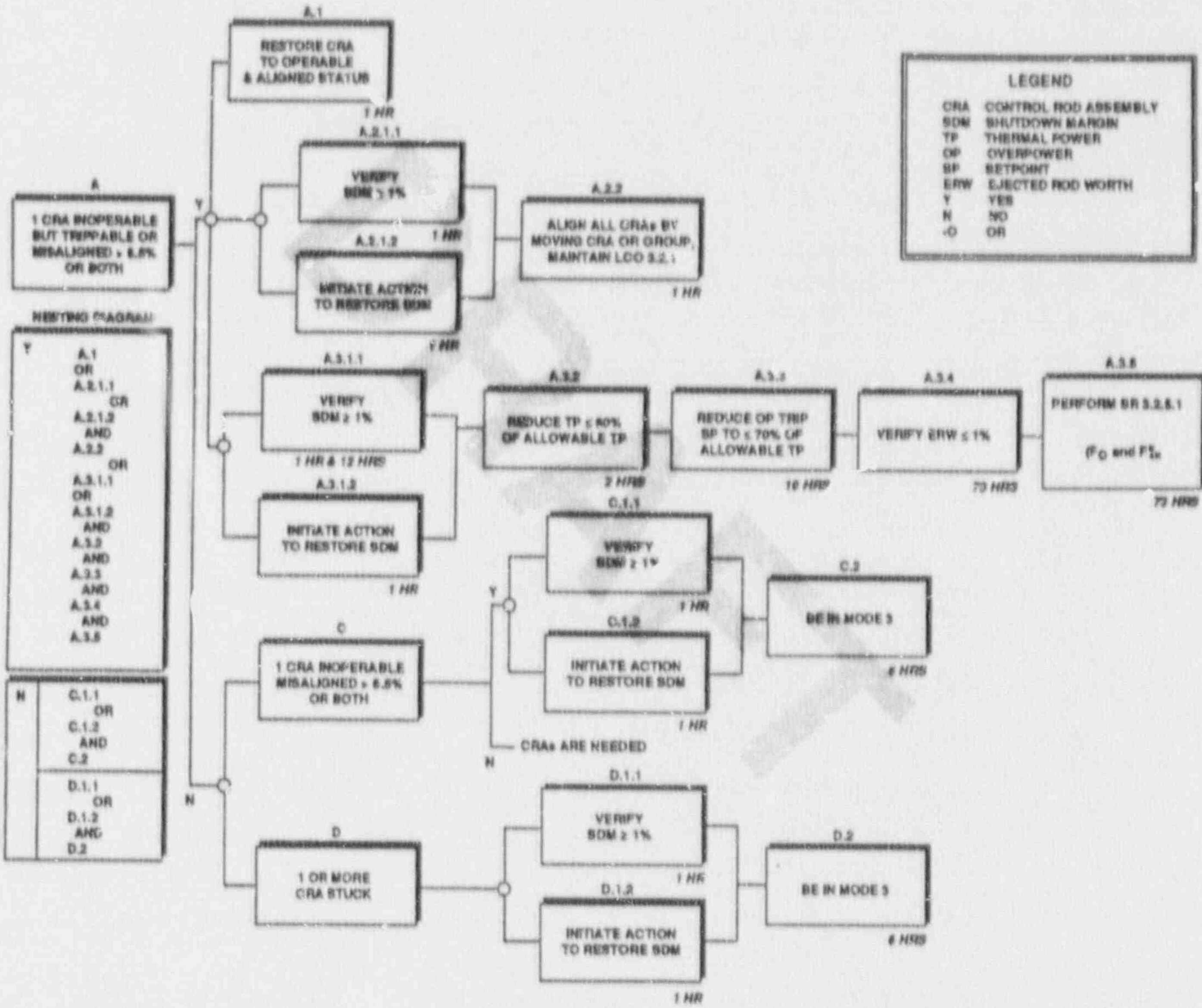

Figure B.3.1.4-1

CONTROL ROD Alignment Action Flowchart

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BASES (continued)

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix A, General Design Criterion 10, "Reactor Design," General Design Criterion 26, "Reactivity Limits."

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

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

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

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

6. [Unit Name] Core Operating Limits Report, "[Title]."

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

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

9. Draft NUREG-1366, "Improvements to Technical Specifications."

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.5 Safety Rod Insertion Limit

BASES

BACKGROUND

The insertion limits of the safety and regulating rods are initial assumptions in all safety analyses which assume rod insertion upon reactor trip. The insertion limits directly affect core power distributions and assumptions of available SHUTDOWN MARGIN (SDM), ejected rod worth, and initial reactivity insertion rate.

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

Limits on safety rod insertion have been established, and all rod positions are monitored and controlled during power operation to ensure that the reactivity limits, ejected rod worth, and SDM limits are preserved.

The regulating banks are used for precise reactivity control of the reactor. The positions of the regulating banks are normally automatically controlled by the Rod Control System, but they can also be manually controlled. They are capable of adding negative reactivity very quickly (compared to borating). The regulating banks must be maintained above designed insertion limits and are typically near the fully withdrawn position during normal operations. Hence, they are not capable of adding a large amount of positive reactivity. Boration or dilution of the Reactor Coolant System (RCS) compensates for the reactivity changes associated with large changes in RCS temperature.

The safety banks are used primarily to help ensure that the required SDM is maintained. The safety banks are controlled manually by the control room operator. During normal full power operation, the safety banks are fully withdrawn. The safety banks must be completely withdrawn from the core prior to withdrawing any regulating banks during an approach to criticality. The safety banks are then left in this position until the reactor is shut down.

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BASES (continued)

BACKGROUND (continued)

They add negative reactivity to shut down the reactor upon receipt of a reactor trip signal.

APPLICABLE SAFETY ANALYSES

On a reactor trip, all rods (safety banks and regulating banks), except the most reactive rod, are assumed to insert into the core. The safety banks shall be at their fully withdrawn limits and available to insert the maximum amount of negative reactivity on a reactor trip signal. The regulating banks may be partially inserted in the core as allowed by LCO 3.2.1, "Regulating Rod Insertion Limits." The safety bank 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 banks and safety banks (less the most reactive rod which is assumed to be fully withdrawn) is sufficient to take the reactor from full power conditions at rated temperature to zero power and to maintain the required SDM at rated no-load temperature (Ref. 3). The safety bank insertion limit also limits the reactivity worth of an ejected safety rod.

The acceptance criteria for addressing safety and regulating rod bank insertion limits and inoperability or misalignment are that:

a. There be no violations of:

    1. specified acceptable fuel design limits,

    2. centerline fuel temperature,

    3. RCS pressure boundary damage; and

b. The core must remain subcritical after accident transients.

As such, the safety rod insertion limits affect safety analysis involving core reactivity and SDM (Ref. 3).

The safety rod insertion limit preserves an initial condition assumed in the safety analyses and, as such, satisfies Criterion 2 of the NRC Interim Policy Statement.

(continued)

BASES (continued)

LCO

The safety banks must be fully withdrawn 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.

[For this facility an OPERABLE safety rod is verified as follows:]

[For this facility, the following support systems are required OPERABLE to ensure safety rod insertion limits are met: [List]]

[For this facility, the required support systems which, upon their failure, do not result in safety rods not meeting their insertion limits or in rod inoperability, and their justification are as follows:]

APPLICABILITY

The safety banks must be within their insertion limits with the reactor in MODES 1 and 2. The applicability in MODE 2 begins within 15 minutes prior to initial regulating bank withdrawal during an approach to criticality and continues throughout MODE 2 until all regulating rods are again fully inserted by scram or during shutdown. This ensures that a sufficient amount of negative reactivity is available to shutdown the reactor and maintain the required SDM following a reactor trip. The reactor is not critical or approaching criticality in MODE 3, 4, 5, or 6, and, therefore, the safety banks must be fully inserted.

This LCO has been modified by a Note that suspends the LCO requirement during SR 3.1.4.2 which assures the freedom of the rods to move. This SR requires the safety bank to move below the LCO limits, which would normally violate the LCO.

ACTIONS

A.1 and A.2

When one or more safety rod(s) is not fully withdrawn, 1 hour is allowed to restore the safety rod(s) to within the insertion limit. This is necessary because the

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BASES (continued)

ACTIONS (continued)

available SDM may be significantly reduced with one or more of the safety rods not within their insertion limits.

Also, initiation of boration within 15 minutes is required since the SDM in MODES 1 and 2 is normally ensured by adhering to the regulating and safety rod insertion limits (see LCO 3.1.1).

In the event that the safety rod's position indication system is found to be inoperable, the safety rod is considered to be not within limits and Required Action A.2 and LCO 3.1.4 apply.

The allowed Completion Time of 1 hour provides an acceptable time for evaluating and repairing minor problems without allowing the plant to remain in an unacceptable condition for an extended period of time.

B.1

If the safety banks cannot be restored to within their insertion limits within 1 hour, the only other acceptable action is to place the unit in a MODE where the LCO is not applicable. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach the required MODE in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.5.1

Verification that each safety rod is fully withdrawn assures the rods are available to provide reactor shutdown capability after criticality. Performing the surveillance 15 minutes prior to withdrawing the first regulating rod group during an approach to criticality assures the safety rods are withdrawn before they may be required for shutdown. This also allows the operator adequate time to halt the approach to criticality should a safety rod not be fully withdrawn. Since the safety rods must be fully withdrawn when MODE 2 is entered during a startup, it may be necessary to perform this surveillance in MODE 3.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

Verification that individual safety rod positions are fully withdrawn at a 12-hour Frequency allows the operator to detect a rod beginning to deviate from its expected position. Also, the 12-hour Frequency takes into account other information available to the operator in the control room that monitor the status of the safety rods.

SR 3.1.5.1 is modified by a Note which allows exemption to SR 3.0.4. SR 3.0.4 is not applicable before entering the Applicability Condition of "within 15 minutes prior to initial control bank withdrawal," because the surveillance is specifically selected to be concurrent with the Applicability.

[For this facility, an OPERABLE safety rod within limits is verified as follows:]

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix A, "Nuclear Power Plants," General Design Criterion 10, "Reactor Design," General Design Criterion 26, "Reactivity Limits."

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

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

4. [Unit Name] Core Operating Limits Report, "[Title]."

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.6 AXIAL POWER SHAPING ROD (APSR) Alignment Limits

BASES

BACKGROUND

The OPERABILITY of the APSRs and rod misalignment are initial assumptions in the safety analysis which directly affects core power distributions. The applicable criteria for these power distribution design requirements are 10 CFR 50, Appendix A, "General Design Criteria for Nuclear Power Plants," GDC 10, "Reactor Design" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 2).

Mechanical or electrical failures may cause an APSR to become inoperable or to become misaligned from its group. APSR inoperability or misalignment may cause increased power peaking due to the asymmetric reactivity distribution. Therefore, APSR alignment and OPERABILITY are related to core operation in design power peaking limits.

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

CONTROL RODS are moved by their CONTROL ROD drive mechanisms (CRDMs). Each CRDM moves its rod $\frac{3}{4}$ inch for one revolution of the leadscrew at varying rates depending on the signal output from the Rod Control System.

The APSRs are arranged into rod groups that are radially symmetric. Therefore, movement of the APSRs does not introduce radial asymmetries in the core power distribution. The APSRs which control the axial power distribution are positioned manually and do not trip.

The axial position of the APSRs is indicated by two separate and independent systems, which are the relative position indicators transducers, and the absolute position indicator transducers (see LCO 3.1.7).

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BASES (continued)

BACKGROUND (continued)

The relative position indicator transducer is a potentiometer that is driven by electrical pulses from the Rod Control System that moves the rods. There is one counter for each group of rods. Individual rods in a group all receive the same signal to move and should, therefore, all be at the same position indicated by the group counter for that group. The Relative Position Indicator System is considered highly accurate (one rotation of the leadscrew is $\frac{3}{4}$ inch). If a rod does not move for each demand pulse, the counter will still count the pulse and incorrectly reflect the position of the rod.

The Absolute Position Indicator System provides a highly accurate indication of actual rod position, but at a lower precision than the step counters. This system is based on inductive analog signals from a series of reed switches spaced along a tube with a center-to-center distance of [3.75 inches], which is [6 steps].

APPLICABLE SAFETY ANALYSES

APSR misalignment and inoperability are analyzed in the safety analysis (Ref. 3). The acceptance criteria for addressing APSR inoperability or misalignment is that there be no violations of:

a. Specified acceptable fuel design limits;

b. Centerline fuel temperature; and

c. Reactor Coolant System (RCS) pressure boundary damage.

Two types of misalignment or inoperability are distinguished. During movement of an APSR group, one rod may stop moving while the other rods in the group continue. This condition may cause excessive power peaking. The second type of misalignment occurs when one rod drops partially or fully into the reactor core. This event causes an initial power reduction, followed by a return towards the original power due to positive reactivity feedback from the negative moderator temperature coefficient. Increased peaking during the power increase may result in excessive local linear heat rates (LHRs).

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

The accident analysis and reload safety evaluations define APSR insertion limits that ensure that if an APSR is stuck in or dropped in, the increase in local LHR is within the design limits. The Required Action statement in the LCO provides a conservative approach to ensure that continued operation remains within the bounds of the safety analysis (Ref. 4).

Continued operation of the reactor with a misaligned APSR is allowed if AXIAL POWER IMBALANCE limits are preserved (Ref. 5).

The APSR group alignment limits are directly related to power peaking. Power peaking is a process variable that satisfies Criterion 2 of the NRC Interim Policy Statement, since it represents initial condition input to the plant safety analysis.

LCO

The limits on CONTROL ROD group alignment, safety rod insertion, and APSR alignment, together with the limits on regulating rod insertion, APSR insertion, AXIAL POWER IMBALANCE, and QUADRANT POWER TILT ensure the reactor will operate within the fuel design criteria. The Required Action in this LCO ensures deviations from the alignment limits will be adjusted so that excessive local LHRs will not occur.

The limit for individual APSR misalignment is 6.5% (9 inches) deviation from the group average position. This value is established based on the distance between reed switches, with additional allowances for uncertainty in the absolute position indicator amplifiers, group maximum or minimum synthesizer, and asymmetric alarm or fault detector outputs. The position of an inoperable rod is not included in the calculation of the rod group's average position.

APSRs are OPERABLE when they can meet the SR of this LCO and can be inserted and withdrawn to meet the alignment limits, sequence and overlap withdrawal requirements, and position indication requirements.

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BASES (continued)

LCO (continued)

[For this facility, an OPERABLE APSR is verified as follows:]

[For this facility, the following support systems are required OPERABLE to ensure APSR alignment limits are met: [List]]

[For this facility, the required support systems which, upon their failure, do not result in the APSR not meeting their alignment limits or in rod inoperability, and their justification are as follows:]

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

APPLICABILITY

The requirements on APSR 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 and alignment of rods has the potential to affect the safety of the plant. In MODES 3, 4, 5, and 6, the alignment limits do not apply because the CONTROL RODS are bottomed and the reactor is shut down and not producing fission power.

ACTIONS

The actions described below are required if one APSR is inoperable. The plant is not allowed to operate with more than one inoperable APSR. This would require the reactor to be shut down in accordance with LCO 3.0.3.

A.1

If an APSR is inoperable or misaligned, or both inoperable and misaligned beyond the specified alignment limit, the first preference is usually to restore it to OPERABLE status within the alignment requirements. When a misaligned APSR occurs, it can usually be moved. If the rod can be realigned within the requirement within 2 hours, local xenon

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BASES (continued)

ACTIONS (continued)

redistribution during this short interval will not be significant, and operation may proceed without further restriction (Ref. 6).

A.2

An alternate to realigning a single misaligned APSR to the group average position is to align the remainder of the APSR group to the position of the misaligned or inoperable APSR while maintaining APSR insertion in accordance with the limits in the CORE OPERATING LIMITS REPORT (COLR) (Ref. 7). This restores the alignment requirements. Deviations up to 2 hours will not cause significant xenon redistribution to occur. Required Action A.2 assumes the APSR group movement does not cause the limits of LCO 3.2.2 ("APSR Insertion Limits") to be exceeded. For this reason, Action A.2 is only practical for instances where small movements of the APSR group are sufficient to re-establish APSR alignment.

A.3

The reactor may continue in operation with the APSR misaligned if the limits of LCO 3.2.3 ("AXIAL POWER IMBALANCE Operating Limits"), and LCO 3.2.2 ("APSR Insertion Limits") are not exceeded. Further movement of the APSR group is prohibited so that the misalignment does not increase and cause the limits on AXIAL POWER IMBALANCE to be exceeded. The required Completion Time of up to 2 hours will not cause significant xenon redistribution to occur.

B.1

The plant must be in a MODE in which the LCO does not apply if the Required Actions and associated Completion Times cannot be met. This is done by placing the plant in at least MODE 3 within 6 hours. The 6 hours allotted to reach MODE 3 is a reasonable time, based on operating experience, to reach MODE 3 from RATED THERMAL POWER in an orderly manner and without challenging plant systems. In MODE 3, APSR group alignment limits are not required, because the reactor is not generating THERMAL POWER and excessive local LHRs cannot occur from APSR misalignment.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1

Verification that individual APSR positions are within 6.5% of the group average height limits at a 12-hour Frequency allows the operator to detect an APSR beginning to deviate from its expected position. If the asymmetric CONTROL ROD alarm is inoperable, a 4-hour Frequency is reasonable to prevent large deviations in APSR alignment from occurring without detection. In addition, APSR position is continuously available to the operator in the control room so that during actual rod motion, deviations can immediately be detected.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix A, General Design Criterion 10, "Reactor Design," and General Design Criterion 26, "Reactivity Limits."
2. Title 10, Code of Federal Regulations, Part 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors."
3. [Unit Name] FSAR, Section [ ], "[Title]."
4. [Unit Name] FSAR, Section [ ], "[Title]."
5. [Unit Name] FSAR, Section [ ], "[Title]."
6. [Unit Name] FSAR, Section [ ], "[Title]."
7. [Unit Name] Core Operating Limits Report, "[Title]."

B 3.1 REACTIVITY CONTROL

B 3.1.7 Position Indicator Channels

BASES

BACKGROUND

Per GDC 13 (Ref. 1), instrumentation to monitor variables and systems over their operating ranges during normal operation, anticipated operational occurrences, and accident conditions must be OPERABLE. LCO 3.1.7 is required to ensure OPERABILITY of the CONTROL ROD and AXIAL POWER SHAPING ROD (APSR) position indicators to determine CONTROL ROD and APSR positions, and thereby ensure compliance with the CONTROL ROD and APSR alignment and insertion limits.

The OPERABILITY, including position indication, of the safety and regulating rods are initial assumptions in all safety analyses that assume rod insertion upon reactor trip. Maximum rod misalignment for the safety, regulating, and APSR are initially assumed in the safety analysis which directly affect core power distributions and assumptions of available SHUTDOWN MARGIN (SDM).

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

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

Two methods of CONTROL ROD and APSR position indication are provided in the CONTROL ROD Drive Control System. The two

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BASES (continued)

BACKGROUND (continued)

means are by absolute position indicator and relative position indicator transducers. The absolute position indicator transducer consists of a series of magnetically operated reed switches
mounted in a tube parallel to the CONTROL ROD Drive Mechanism (CRDM) motor tube extension. Switch contacts close when a permanent magnet mounted on the upper end of the CONTROL ROD assembly (CRA) leadscrew extension comes near. As the leadscrew and CRA move, the switches operate sequentially, producing an analog voltage proportional to position. Other reed switches included in the same tube with the position indicator matrix provide full-in and full-out limit indications, and absolute position indications at 0%, 25%, 50%, 75%, and 100% travel (called zone reference indicators). The relative position indicator transducer is a potentiometer driven by a step motor that produces a signal proportional to CONTROL ROD position based on the electrical pulse steps that drive the CRDM.

Two absolute position indicator channel designs may be used in the unit, type A absolute position indicators and type A-R4C absolute position indicators. The type A absolute position indicator transducer is a voltage divider circuit made up of 48 resistors of equal value connected in series. One end of 48 reed switches is connected at a junction between each of the resistors, so that as the magnet mounted on the leadscrew moves, either one or two reed switches are closed in the vicinity of the magnet. The type A-R4C (redundant four channel) absolute position indicator transducer has two parallel sets of voltage divider circuits made up of 36 resistors each, connected in series (channels A and B). One end of 36 reed switches is connected at a junction between each of the resistors of the two parallel circuits. The reed switches making up each circuit are offset such that the switches for channel A are staggered with the switches for channel B. The type A-R4C is designed such that either two or three reed switches are closed in the vicinity of the magnet. By its design, the type A-R4C absolute position indicator provides redundancy with the two-three sequence of pickup and drop out of reed switches to enable a continuity of position signal when a single reed switch fails to close.

CONTROL ROD position-indicating readout devices located in the control room consist of single CRA position meters on a

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BASES (continued)

BACKGROUND (continued)

wall-mounted position indication panel and four group average position meters on the console. A selector switch permits either relative or absolute position indication to be displayed on all of the single rod meters. Indicator lights are provided on the single CRA meter panel to indicate when each CRA is fully withdrawn, fully inserted, enabled, or transferred, and whether a CRA position asymmetry alarm condition is present. Indicators on the console show full insertion, full withdrawal, and enabled for motion for each CONTROL ROD group. Identical instrumentation and devices exist for the APSR group. The consequence of continued operation with an inoperable absolute position indicator or relative position indicator channel is a decreased reliability in determining CONTROL ROD position. Therefore, the potential for operation in violation of design peaking factors or SHUTDOWN MARGIN is increased.

APPLICABLE SAFETY ANALYSES

CONTROL ROD and APSR position accuracy is essential during power operation. Power peaking, ejected rod worth, or SDM limits may be violated in the event of a Design Basis Accident (DBA) (Ref. 2a-2f) with CONTROL RODS or APSRs operating outside their limits undetected. Regulating rod, safety rod, and APSR positions must be known in order to verify the core is operating within the group sequence, overlap, design peaking limits, ejected rod worth, and with minimum SDM (LCO 3.1.5, "Safety Rod Insertion Limits," LCO 3.2.1, "Regulating Rod Insertion Limits," and LCO 3.2.2, "APSR Insertion Limits"). The rod positions must also be known in order to verify the alignment limits are preserved (LCO 3.1.4, "CONTROL ROD Group Alignment Limits," and LCO 3.1.6, "APSR Alignment Limits"). CONTROL ROD and APSR positions are continuously monitored to provide operators with information that assures the plant is operating within the bounds of the accident analysis assumptions. The CONTROL ROD position indicator channels satisfy Criterion 2 of the NRC Interim Policy Statement. The CONTROL ROD position indicators monitor CONTROL ROD position, which is an accident initial condition.

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BASES (continued)

LCO

LCO 3.1.7 specifies that one absolute position indicator channel and one relative position indicator channel be OPERABLE for each CONTROL ROD and APSR. OPERABILITY for the CONTROL ROD position indicators has the following meanings:

a. The absolute position indicator channels (either type A or type A-R4C) have passed a CHANNEL FUNCTIONAL CHECK within the prescribed interval;

b. For the type A Absolute Position Indicator System, there are no failed reed switches. Specific surveillance of reed switches for OPERABILITY during operation is not required with the type A System. This is because of the following:

   1. a reed switch failed closed results in a large indication of asymmetry unless the CONTROL ROD is near the failed closed reed switch,

   2. a failed open reed switch results in a large indication of asymmetry when the CONTROL ROD is near the failed open reed switch;

c. For the type A-R4C Absolute Position Indicator System, either there are no failed reed switches in one of the two channels or, with both channels in operation, two failed reed switches are not in sequence. Specific surveillance of reed switches for OPERABILITY during operation is not required with the type A-R4C System. This is because of the following:

   1. a reed switch failed closed results in a large absolute position indicator indication of asymmetry unless the CONTROL ROD is near the failed closed reed switch,

   2. two failed open reed switches in sequence result in a large indication of asymmetry when the CONTROL ROD is near the two failed open reed switches,

   Alternating failed open reed switches or one of the two type A-R4C channels disconnected is assumed in the analysis; and

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BASES (continued)

LCO (continued)

d. The relative position indicator channels have been calibrated either in the fully inserted position or to the absolute position indicator channels. The agreement between the relative position indicator channel and the absolute position indicator channels is within the limit given in the CORE OPERATING LIMITS REPORT (COLR), indicating that relative position indicators are adequately calibrated for measurement of CONTROL ROD group position. A deviation of less than the allowable limit given in the COLR in position indication for a single CONTROL ROD or APSR ensures confidence that the position uncertainty of the corresponding CONTROL ROD group or APSR group is within the assumed values used in the analysis that specifies CONTROL ROD group and APSR insertion limits.

[For this facility, the following support systems are required to be operable to ensure the position indicator channels are operable: [list]]

[For this facility, the required support systems, which upon their failure do not result in the inoperability of the position indicator channels, and the justification are as follows:]

These requirements provide adequate assurance that CONTROL ROD position indication during power operation and PHYSICS TESTS is accurate, and that design assumptions are not challenged. OPERABILITY of the position indicator channels ensures that inoperable, misaligned, or mispositioned CONTROL RODS or APSRs can be detected. Therefore, power peaking and SDM can be controlled within acceptable limits. The action statements required when the limits of LCO 3.1.7 are not met are diagrammed in Figure B 3.1.7-1.

APPLICABILITY

In MODES 1 and 2, OPERABILITY of position indicator channels is required, since the reactor is, or is capable of, generating THERMAL POWER in these MODES. In MODES 3, 4, 5, and 6, Applicability is not required, because the reactor is shut down with the required minimum SDM and is not generating THERMAL POWER.

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BASES (continued)

ACTIONS

A.1

If one relative position indicator channel is inoperable for one or more rods, the position of the rod(s) is still monitored by the absolute position indicator channel for each affected rod. The absolute position indicator channel may be used if it is determined to be OPERABLE. The required Completion Time of 8 hours is reasonable to provide adequate time for the operator to determine position indicator channel status. Continuing the verification every 8 hours thereafter in the applicable condition is acceptable based on the fact that during normal power operation excessive movement of the banks is not required. Also, if the rod is out of position during this 8-hour period, the simultaneous occurrence of an event sensitive to the rod position has a small probability.

B.1.1

If one absolute position indicator channel is inoperable for one or more rods, the position of the rod(s) is monitored by the relative position indicator channel for each affected rod. However, the relative position indicator channel is not as reliable a method of monitoring rod position as the absolute position indicator because it counts electrical pulse steps driving the CRDM motor rather than actuating a switch located at a known elevation. Therefore, the affected rod's position can be determined with more certainty by actuating one of its zone reference indicator switches located at discrete elevations. The required Completion Time of 8 hours provides the operator adequate time to adjust the affected rod's position to an appropriate zone reference indicator location. And if the rod is out of position during this 8-hour period, the simultaneous occurrence of an event sensitive to the rod position has a small probability.

B.1.2

To allow continued operation, the rods with inoperable absolute position indicator channels are maintained at the zone reference indicator position. In addition, the affected rods are maintained within the limits of LCO 3.1.5 (when the affected rod is a safety rod), LCO 3.2.1

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BASES (continued)

Figure B 3.1.7-1
Position Indicator Channel Action Flow Chart

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BASES (continued)

ACTIONS (continued)

(when the affected rod is a regulating rod), or LCO 3.2.2 (when the affected rod is an APSR). This Required Action ensures safety rods remain fully withdrawn, and that regulating rods and APSRs remain aligned within their insertion limits. The required Completion Time of 8 hours is reasonable to allow the operator adequate time to determine the affected rods are in compliance with these LCOs. Continuing to verify the rod positions every 8 hours thereafter is reasonable for providing assurance that rod alignment and insertion are not changing, and provides the operator adequate time to correct any deviation that may occur. The required Completion Time of 8 hours is reasonable to provide adequate time for the operator to determine position indicator channel status. Continuing the verification every 8 hours thereafter in the applicable condition is acceptable based on the fact that during normal power operation excessive movement of the banks is not required. Also, if the rod is out of position during this 8-hour period, the simultaneous occurrence of an event sensitive to the rod position has a small probability.

B.2

If the absolute position indicator is inoperable for one or more rods, the position of the rod can be monitored by the zone reference indicator switches. If the zone reference indicator switches are also inoperable, the position of the rod is not known with certainty since the relative position indicator is a less reliable method of monitoring rod position. Therefore, the rod must be declared inoperable. The required Completion Time of 8 hours is reasonable to allow the operator adequate time to determine if the affected rods are in compliance with these LCOs.

C.1

If both the absolute position indicator channel and relative position indicator channel are inoperable for one or more rods, the position of the rod(s) is not known with certainty. Therefore, each affected rod must be declared inoperable, and the limits of LCO 3.1.4 or LCO 3.1.6 apply. The required Completion Time to declare the rod(s) inoperable is immediately. Therefore LCO 3.1.4 or 3.1.6 is entered immediately, and the required Completion Times for the appropriate Required Actions in those LCOs apply without delay.

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BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.1.7.1

Verification that the relative position indicator agree with the absolute position indicator within 2% steps over the full indicated range of the regulating rods and safety rods provides assurance that the relative position indicator is operating correctly.

The 18-month Frequency was developed considering it was prudent that many surveillances only be performed during a plant outage. This was due to the plant conditions needed to perform the SR and the potential for unnecessary plant transients if the SR is performed with the reactor at power. Operating experience has shown these components virtually always pass the SR when performed on the 18-month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.1.7.2

Performance of a CHANNEL FUNCTIONAL TEST to verify the absolute position indicator channels are OPERABLE provides assurance that each absolute position indicator channel is functioning properly. The required Frequency of 18 months is sufficient to detect problems with the instrumentation.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix A, General Design Criterion 13, "Instrumentation and Control."

2. [Unit Name], FSAR, Docket 50-302, Florida Power Corporation,

   a. Section [14.1.2.2], "[Startup Accident]."

   b. Section [14.1.2.3], "[Rod Withdrawal Accident at Rated Power Operation]."

   c. Section [14.1.2.6], "[Loss of Coolant Flow]."

   d. Section [14.1.2.7], "[Stuck-Out, Stuck-In, or Dropped Control Rod Accident]."

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BASES (continued)

REFERENCES (continued)

e. Section [14.2.2.4], "[Rod Ejection Accident]."

f. Section [14.2.2.5], "[Loss-of-Coolant Accident (LOCA)]."

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.8 MODE 1 PHYSICS TESTS Exceptions

BASES

BACKGROUND

The purpose of this MODE 1 LCO is to permit PHYSICS TESTS to be conducted by providing exemptions from the requirements of other LCOs. Establishment of a test program to verify that structures, systems, and components will perform satisfactorily in service is required by Section XI of 10 CFR 50, Appendix B (Ref. 1). Testing is required as an integral part of the design, fabrication, construction, and operation of the power plant. All functions necessary to ensure that specified design conditions are not violated during normal operation and anticipated operational occurrences must be tested. Requirements for notification of the NRC for the purpose of conducting tests and experiments are specified in 10 CFR 50.59 (Ref. 2).

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

a. Provide assurance that the facility has been adequately designed;

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

c. Verify the assumptions used to predict unit response;

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

e. Verify that the operating and emergency procedures are adequate.

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

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BASES (continued)

BACKGROUND (continued)

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

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

APPLICABLE SAFETY ANALYSES

It is acceptable to suspend certain LCOs for PHYSICS TESTS because reactor protection criteria are preserved by the LCOs still in effect and by the SRs. Even if an accident occurs during PHYSICS TESTS with one or more LCOs suspended, fuel damage criteria are preserved because the limits on nuclear hot channel factors and shutdown capability are maintained during the PHYSICS TESTS.

Reference 5 defines requirements for initial testing of the facility, including PHYSICS TESTS. Tables 13-3 and 13-4 (Ref. 6) summarize the zero, low power, and power tests. Requirements for reload fuel cycle PHYSICS TESTS are given in Table 1 ANSI/ANS-19.6.1-1985 (Ref. 4). Although these PHYSICS TESTS are generally accomplished within the limits of all LCOs, one or more LCOs must sometimes be suspended to make completion of PHYSICS TESTS possible or practical.

This is acceptable as long as the fuel design criteria are not violated. When one or more of the limits specified in

LCO 3.1.4 "Control Rod Group Alignment Limits,"
LCO 3.1.5 "Safety Rod Insertion Limits,"
LCO 3.1.6 "AXIAL POWER SHAPING ROD (APSR) Alignment Limits,"
LCO 3.2.1 "Regulating Rod Insertion Limits,"
LCO 3.2.3 "AXIAL POWER IMBALANCE Operating Limits," or
LCO 3.2.4 "Quadrant Power Tilt (QPT)"

are suspended for PHYSICS TESTS, the fuel design criteria are preserved by maintaining the nuclear hot channel factors

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

(in MODE 1 PHYSICS TESTS) within their limits and by limiting maximum THERMAL POWER. Therefore, surveillance of the NUCLEAR HEAT FLUX HOT CHANNEL FACTOR ($F_Q(Z)$) and the NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR ($F^N_{\Delta H}$) is required to verify that their limits are not exceeded. The limits for the nuclear hot channel factors are specified in the CORE OPERATING LIMITS REPORT (COLR) (Ref. 7). Refer to the Basis for LCO 3.2.5 for a complete discussion of $F_Q(Z)$ and $F^N_{\Delta H}$. During PHYSICS TESTS, one or more of the LCOs that normally preserve the $F_Q(Z)$ and $F^N_{\Delta H}$ limits and shutdown capability may be suspended. However, the results of the safety analysis are not adversely impacted if verification that $F_Q(Z)$ and $F^N_{\Delta H}$ are within their limits is obtained while one or more of the LCOs is suspended. Therefore, limits are placed on $F_Q(Z)$ and $F^N_{\Delta H}$ during MODE 1 PHYSICS TESTS to verify that these factors remain within their limits. Periodic verification of these factors allows PHYSICS TESTS to be conducted while continuing to maintain the design criteria. Since the requirements to monitor $F_Q(Z)$ and $F^N_{\Delta H}$ are built into this specification, it is not necessary to enter Specification 3.2.5 to perform these surveillances. In addition, the actions required when $F_Q(Z)$ and $F^N_{\Delta H}$ exceed their limits are also built into the Required Actions of this specification. Therefore, the operator does not need to enter Specification 3.2.5 when MODE 1 PHYSICS TESTS are conducted.

PHYSICS TESTS include measurement of core nuclear parameters or exercise of control components that affect process variables. Among the process variables involved are API and QPT, which represent initial condition input (power peaking) for the accident analysis. Also involved are the movable control components, i.e., the regulating rod and the APSR, which affect power peaking and are required for shutdown of the reactor. The limits for these variables are specified for each fuel cycle in the COLR (Ref. 7).

PHYSICS TESTS meet the criteria for inclusion in Technical Specifications since the component and process variable LCOs suspended during PHYSICS TESTS meet criteria 1, 2, and 3 of the NRC Interim Policy Statement.

LCO

This LCO permits individual CONTROL RODS to be positioned outside of their specified group alignment and withdrawal

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BASES (continued)

LCO (continued)

limits and to be assigned to other than specified CONTROL ROD groups, and permits API and QPT limits to be exceeded during the performance of PHYSICS TESTS. In addition, this LCO permits verification of the fundamental core characteristics and nuclear instrumentation operation.

The requirements of LCO 3.1.4, LCO 3.1.5, LCO 3.1.6, LCO 3.2.1, LCO 3.2.3, and LCO 3.2.4 may be suspended during the performance of PHYSICS TESTS provided:

a. THERMAL POWER is maintained ≤ 85% RTP;

b. Nuclear overpower trip setpoint is ≤ 10% RTP higher than the THERMAL POWER at which the test is performed, with a maximum setting of 90% RTP; and

c. $F_Q(Z)$ and $F^N_{\Delta H}$ are maintained within limits specified in the COLR.

Operation with THERMAL POWER ≤ 85% of RTP during PHYSICS TESTS provides an acceptable thermal margin when one or more of the applicable LCOs is out of specification.

[For this facility, the following support systems are required to be OPERABLE to ensure that the LCO and SR conditions are met: [List]]

[For this facility, the required support systems, which upon their failure do not result in the Conditions of this LCO to not be met, and their justification are as follows: [List]]

APPLICABILITY

This LCO is applicable in MODE 1 when the reactor has completed low power testing and is in power ascension or during power operation with THERMAL POWER greater than 5% of RATED THERMAL POWER (RTP) but no greater than 85% of RTP. This LCO is applicable for power ascension testing as defined by Regulatory Guide 1.68 (Ref. 3). In MODE 2, Applicability of this LCO is not required because LCO 3.1.9 addresses PHYSICS TESTS exceptions in MODE 2. In MODES 3, 4, 5, and 6, Applicability is not required because physics testing is not performed in these MODES.

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BASES (continued)

ACTIONS

A.1

If THERMAL POWER exceeds 85% of RTP, then THERMAL POWER must be reduced to restore the additional thermal margin provided by the reduced power level. The required Completion Time of 1 hour gives the operator adequate time to restore THERMAL POWER to within specification. This required Completion Time is consistent with, or more conservative than, those specified for the individual LCOs addressed by PHYSICS TESTS Exceptions.

A.2

If THERMAL POWER exceeds 85% of RTP, then 1 hour is allowed for the operator to complete an orderly suspension of PHYSICS TESTS Exceptions. Suspension of PHYSICS TESTS Exceptions requires restoration of each of the applicable individual LCOs to within specification. This required Completion Time is consistent with, or more conservative than, those specified for the individual LCOs addressed by PHYSICS TESTS Exceptions.

B.1

The nuclear overpower trip setpoint must be no more than 10% of RTP higher than the PHYSICS TESTS power level to maintain both core protection and an operating margin at the reduced power level similar to that at RTP. The nuclear overpower trip setpoint must also be no more than 90% of RTP if PHYSICS TESTS are performed at a THERMAL POWER between 80% and 85% of RTP. If either of these conditions is not met, then 1 hour is allowed to restore the nuclear overpower trip setpoint to within limits. The required Completion Time of 1 hour to restore the nuclear overpower trip setpoint to within limits is reasonable, based on operating experience, for adjusting this setpoint and is consistent with that of Required Action A.1.

B.2

If the nuclear overpower trip setpoint is not within the specified limits, then 1 hour is allowed for the operator to complete an orderly suspension of PHYSICS TESTS exceptions. Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable individual LCOs to within

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BASES (continued)

ACTIONS (continued)

specification. This required Completion Time is consistent with, or more conservative than, those specified for the individual LCOs addressed by these PHYSICS TESTS exceptions.

SURVEILLANCE REQUIREMENTS

SR 3.1.8.1

Verification that THERMAL POWER is ≤ 85% of RTP ensures that the required additional thermal margin has been established prior to and during PHYSICS TESTS. The required Frequency of once per hour allows the operator adequate time to determine any degradation of the established thermal margin during PHYSICS TESTS.

SR 3.1.8.2

Verification that $F_Q(Z)$ and $F^N_{\Delta H}$ are within their limits ensures that core local linear heat rate and departure from nucleate boiling ratio will remain within their limits while one or more of the LCOs that normally control these design limits are out of specification. The required Frequency of 2 hours allows the operator adequate time to collect a flux map and to perform the hot channel factor verifications based on operating experience. This Frequency is more conservative that the Completion Time for restoration of the individual LCOs that preserve the $F_Q(Z)$ and $F^N_{\Delta H}$ limits.

SR 3.1.8.3

Verification that the nuclear overpower trip setpoint is within the limit specified for each Physics Test ensures that core protection at the reduced power level is established and will remain in place during the PHYSICS TESTS. Performing the verification once every 8 hours allows the operator adequate time to determine any degradation of the established trip setpoint margin before and during PHYSICS TESTS and to adjust the nuclear overpower trip setpoint.

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BASES (continued)

REFERENCES

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

2. Title 10, Code of Federal Regulations, Part 50.59, "Changes, Tests, and Experiments."

3. Regulatory Guide 1.68, Revision 2, "Initial Test Programs for Water-Cooled Nuclear Power Plants," U.S. Nuclear Regulatory Commission, August 1978.

4. ANSI/ANS-19.6.1-1985, "Reload Startup PHYSICS TESTS for Pressurized Water Reactors," American National Standards Institute, December 13, 1985.

5. [Unit Name] FSAR, Section [13.4.8], "[Post Criticality Testing]."

6. [Unit Name] FSAR, Section [13.4.8], [Tables 13-3 and 13-4, Am. 49, September 30, 1976].

7. [Unit Name] Core Operating Limits Report, "[Title]."

B 3.1 REACTIVITY CONTROL

B 3.1.9 MODE 2 PHYSICS TEST Exceptions

BASES

BACKGROUND

The purpose of this MODE 2 LCO is to permit PHYSICS TESTS to be conducted by providing exemptions from the requirements of other LCOs. Establishment of a test program to verify that structures, systems, and components will perform satisfactorily in service is required by 10 CFR 50, Appendix B (Ref. 1). Testing is required as an integral part of the design, fabrication, construction, and operation of the power plant. All functions necessary to ensure that specified design conditions are not violated during normal operation and anticipated operational occurrences (AOOs) must be tested. Requirements for notification of the NRC for the purpose of conducting tests and experiments are specified in 10 CFR 50.59 (Ref. 2).

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

a. Provide assurance that the facility has been adequately designed;

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

c. Verify the assumptions used to predict unit response;

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

e. Verify that the operating and emergency procedures are adequate.

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

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BASES (continued)

BACKGROUND (continued)

PHYSICS TEST procedures are written and approved in accordance with established guidelines. 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 approved prior to continued power escalation and long-term power operation.

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

APPLICABLE SAFETY ANALYSES

It is acceptable to suspend certain LCOs for PHYSICS TESTS because reactor protection criteria are preserved by the LCOs still maintained and by the SRs. Even if an accident occurs during PHYSICS TESTS with one or more LCOs suspended, fuel damage criteria are preserved because the limits on nuclear hot channel factors and shutdown capability are maintained during the PHYSICS TESTS.

Reference 5 defines requirements for initial testing of the facility, including PHYSICS TESTS. Tables 13-3 and 13-4 (Ref. 6) summarize the zero, low power, and power tests. Requirements for reload fuel cycle PHYSICS TESTS are given in Table 1 of ANSI/ANS - 19.6.1-1985 (Ref. 4). Although these PHYSICS TESTS are generally accomplished within the limits of all LCOs, conditions may occur when one or more of the LCOs must be suspended to make completions of PHYSICS TESTS possible or practical.

This is acceptable as long as the fuel design criteria are not violated. When one or more of the limits specified in

LCO 3.1.3, "Moderator Temperature Coefficient,"
LCO 3.1.4, "CONTROL ROD Group Alignment Limits,"
LCO 3.1.5, "Safety Rod Insertion Limits,"
LCO 3.1.6, "AXIAL POWER SHAPING ROD (APSR) Alignment Limits,"
LCO 3.2.1, "Regulating Rod Insertion Limits," and
LCO 3.4.2, "Minimum Temperature for Criticality (MTC)"

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

are suspended for PHYSICS TESTS, the fuel design criteria are preserved by maintaining the nuclear hot channel factors (in MODE 1 PHYSICS TESTS) within their limits and by limiting maximum THERMAL POWER. Therefore, surveillance of the NUCLEAR HEAT FLUX HOT CHANNEL FACTOR ($F_Q(Z)$) and the NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR ($F^N_{\Delta H}$) is required to verify that their limits are not exceeded. The limits for the nuclear hot channel factors are specified in the CORE OPERATING LIMITS REPORT (COLR) (Ref. 7). Refer to the Bases for LCO 3.2.5 for a complete discussion of $F_Q(Z)$ and $F^N_{\Delta H}$.

Shutdown capability is preserved by limiting maximum obtainable THERMAL POWER when in MODE 2 PHYSICS TESTS. In MODE 2, the Reactor Coolant System (RCS) temperature must be within the narrow-range instrumentation for plant control. The narrow-range temperature instrumentation goes on scale at 520°F. Therefore, it is considered safe to allow the minimum RCS temperature to decrease to 520°F during MODE 2 PHYSICS TESTS, based on the low probability of an accident occurring and on prior operating experience.

PHYSICS TESTS include measurement of core nuclear parameters or exercise of control components that affect process variables. Among the process variables involved are AXIAL POWER IMBALANCE (API) and QUADRANT POWER TILT (QPT), which represent initial condition input (power peaking) to the accident analysis. Also involved are the movable control components (i.e., the regulating rod and the AXIAL POWER SHAPING ROD (APSR), which affect power peaking and are required for shutdown of the reactor. The limits for these variables are specified for each fuel cycle in the COLR (Ref. 7).

PHYSICS TESTS meet the criteria for inclusion in Technical Specifications since the component and process variable LCOs suspended during PHYSICS TESTS meet criteria 1, 2, and 3 of the NRC Interim Policy Statement.

LCO

This LCO permits individual CONTROL RODS to be positioned outside of their specified group alignment and withdrawal limits and to be assigned to other than specified CONTROL

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BASES (continued)

LCO (continued)

ROD groups, and permits API and QPT limits to be exceeded during the performance of PHYSICS TESTS. In addition, this LCO permits verification of the fundamental core characteristics and nuclear instrumentation operation.

This LCO also allows suspension of LCO 3.1.3, LCO 3.1.4, LCO 3.1.5, LCO 3.1.6, LCO 3.2.1, and LCO 3.4.2 provided:

a. THERMAL POWER is ≤ 5% RATED THERMAL POWER (RTP);

b. Reactor trip setpoints on the OPERABLE nuclear overpower channels are set to ≤ 25% RTP; and

c. Nuclear instrumentation source range and intermediate range high startup rate CONTROL ROD withdrawal inhibit are OPERABLE.

The limits of LCO 3.2.3 and LCO 3.2.4 do not apply in MODE 2. Inhibiting CONTROL ROD withdrawal based on startup rate also limits local linear heat rate (LHR), departure from nucleate boiling ratio (DNBR), and peak RCS pressure during accidents initiated from low power.

[For this facility, the following support systems are required to be OPERABLE to ensure that the LCO and SR conditions are met: [List]]

[For this facility, the required support systems, which upon their failure do not result in the conditions of this LCO to not be met, and their justification are as follows: [list]]

APPLICABILITY

This LCO is applicable in MODE 2 when the reactor is either not critical or when THERMAL POWER is ≤ 5% RTP. This LCO is applicable for initial criticality or low power testing as defined by Regulatory Guide 1.68 (Ref. 3). In MODE 1, Applicability of this LCO is not required because LCO 3.1.8 addresses PHYSICS TEST Exceptions in MODE 1. In MODES 3, 4, 5, and 6, Applicability is not required because physics testing is not performed in these MODES.

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BASES (continued)

ACTIONS

A.1

If THERMAL POWER exceeds 5% RTP, a positive reactivity addition could be occurring and a nuclear excursion could result. To ensure that local LHR, DNBR, and RCS pressure limits are not violated, the reactor is tripped. The necessary prompt action requires manual operator action without attempts to reduce THERMAL POWER by actuating the control system (i.e., CONTROL ROD insertion or RCS boration).

B.1

If the nuclear overpower trip setpoint is greater than 25% RTP, a reactivity addition could result in a greater DNBR decrease or RCS pressure increase than would be allowed under normal conditions. The required Completion Time of 1 hour to restore the nuclear overpower trip setpoint to within the specified limits is reasonable based on operating experience, the low probability of an accident occurring, and the time required to complete the action without challenging safety systems.

B.2

If the nuclear overpower trip setpoint is greater than 25% RTP, then 1 hour is allowed for the operator to complete an orderly suspension of PHYSICS TEST Exceptions. Suspension of PHYSICS TEST Exceptions requires restoration of each of the applicable individual LCOs to within specification in order to ensure that continuity of reactor operation is within initial condition limits. This required Completion Time is consistent with, or more conservative than, those specified for the individual LCOs addressed by PHYSICS TEST Exceptions.

C.1

The nuclear instrumentation source and intermediate-range, high startup rate, CONTROL ROD withdrawal inhibit functions prevent an uncontrolled CONTROL ROD withdrawal (such as during a startup accident) from contributing to an undesirable reactivity addition. If these functions are

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BASES (continued)

ACTIONS (continued)

inoperable, restoring these functions to OPERABLE status within 1 hour is a reasonable amount of time for the operator to resolve any system problems that have rendered them inoperable.

C.2

If the nuclear instrumentation source and intermediate-range, high startup rate, CONTROL ROD withdrawal inhibit functions are inoperable, then 1 hour is allowed for the operator to complete an orderly suspension of PHYSICS TEST Exceptions. Suspension of PHYSICS TEST Exceptions requires restoration of each of the applicable individual LCOs to within specification. This required Completion Time is consistent with, or more conservative than, those specified for the individual LCOs addressed by PHYSICS TEST Exceptions.

SURVEILLANCE REQUIREMENTS

SR 3.1.9.1

Performing a CHANNEL FUNCTIONAL TEST on each nuclear instrumentation source and intermediate-range and nuclear overpower channel ensures that the instrumentation required to detect a channel ensures that the instrumentation required to detect deviation from THERMAL POWER or to detect a high startup rate is OPERABLE. Performing the test once within 24 hours prior to initiating PHYSICS TESTS ensures that the instrumentation is OPERABLE shortly before PHYSICS TESTS begin and allows the operator to correct any instrumentation problems.

SR 3.1.9.2

Verification that THERMAL POWER is $\leq$ 5% RTP ensures that an adequate margin is maintained between the THERMAL POWER level and the nuclear overpower trip setpoint. Hourly verification is adequate for the operator to determine any change in core conditions, such as xenon redistribution occurring after a THERMAL POWER reduction, that could cause THERMAL POWER to exceed the specified limit.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.9.3

Verification that the nuclear overpower trip setpoint is within the limit specified for PHYSICS TESTS ensures that core protection at the reduced power level is established and will remain in place during PHYSICS TESTS. Performing the verification once per 8 hours allows the operator adequate time to determine any degradation of the established trip setpoint margin before and during PHYSICS TESTS and to adjust the nuclear overpower trip setpoint.

REFERENCES

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

2. Title 10, Code of Federal Regulations, Part 50.59. "Changes, Tests, and Experiments."

3. Regulatory Guide 1.68, Revision 2, "Initial Test Programs for Water-Cooled Nuclear Power Plants," U.S. Nuclear Regulatory Commission, August 1978.

4. ANSI/ANS-19.6.1-1985, "Reload Startup Physics Tests for Pressurized Water Reactors," American National Standards Institute, December 13, 1985.

5. [Unit Name] FSAR, Section [13.4.8], "[Post Criticality Testing]."

6. [Unit Name] FSAR, Section [13.4.8], [Table 13-3, "Title," and Table 13-4, "Title]."

7. [Unit Name] Core Operating Limits Report, "[Title]."

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1 Regulating Rod Insertion Limits

BASES

BACKGROUND

The insertion limits of the regulating rods are initial assumptions used in all safety analyses which assume rod insertion upon reactor trip. The insertion limits directly affect the core power distributions, the assumptions of available SHUTDOWN MARGIN (SDM), and the initial reactivity insertion rate.

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

The limits on regulating rod insertion have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are not violated.

The regulating rod groups operate with a predetermined amount of position overlap in order to approximate a linear relation between rod worth and rod position (integral rod worth). To achieve this linear relationship, the regulating rod groups are withdrawn and operated in a predetermined sequence. The automatic control system controls reactivity by moving the regulating rod groups in sequence within analyzed ranges. The group sequence and overlap limits are specified in the CORE OPERATING LIMITS REPORT (COLR).

The regulating rods are used for precise reactivity control of the reactor. The positions of the regulating rods are normally controlled automatically by the automatic Control System but can also be manually controlled. They are capable of adding reactivity very quickly compared to borating or diluting the Reactor Coolant System (RCS).

The power density at any point in the core must be limited to maintain specified acceptable fuel design limits, including limits which assure that the criteria specified in

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BASES (continued)

BACKGROUND (continued)

10 CFR 50.46 (Ref. 2) are not violated. Together, LCO 3.2.1, LCO 3.2.2, LCO 3.2.3, and LCO 3.2.4 provide limits on control component operation and on monitored process variables to ensure that the core operates within the limits for the NUCLEAR HEAT FLUX HOT CHANNEL FACTOR ($F_Q(Z)$) and the NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR ($F^N_{\Delta H}$) limits in the COLR, Reference 3. Operation within the $F_Q(Z)$ limits given in the COLR prevents power peaks that would exceed the loss- of-coolant accident (LOCA) limits derived from the analysis of the Emergency Core Cooling System (ECCS). Operation within the $F^N_{\Delta H}$ limits, given in the COLR, prevents Departure from Nucleate Boiling (DNB) during a loss-of- forced-reactor-coolant-flow accident. In addition to the $F_Q(Z)$ and $F^N_{\Delta H}$ limits, certain reactivity limits are met by regulating rod insertion limits. The regulating rod insertion limits also restrict the ejected CONTROL ROD worth to the values assumed in the safety analysis and maintain the minimum required SDM in MODES 1 and 2.

The regulating rod insertion and alignment limits, AXIAL POWER IMBALANCE, QUADRANT POWER TILT (QPT), and AXIAL POWER SHAPING ROD (APSR) position are process variables which together characterize and control the three-dimensional power distribution of the reactor core. Additionally, the safety and regulating bank insertion limits control the reactivity which could be added in the event of a rod ejection accident, and the shutdown and regulating bank insertion limits assure that the required SDM is maintained.

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

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation (Condition 1) and anticipated operational occurrences (Condition 2). The LCO governing regulating rod insertion, APSR position, AXIAL POWER IMBALANCE, and QPT

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

preclude core power distributions from occurring which 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 (Ref. 2).

b. During a loss-of-forced-reactor-coolant-flow accident, there must be at least a 95% probability at a 95% confidence level that the hot fuel rod in the core does not experience a DNB condition (Ref. 1). This is referred to hereafter as the 95/95 DNB criterion.

c. During an ejected rod accident, the fission energy input to the fuel must not exceed 280 cal/gm (Ref. 4b).

d. The CONTROL RODS must be capable of shutting down the reactor with a minimum required SDM with the highest worth CONTROL ROD stuck fully withdrawn (Ref. 1).

Fuel cladding damage does not occur when the core is operated outside these LCOs during normal operation. However, fuel-cladding damage could result should an accident occur with the simultaneous violation of one or more of the LCOs limiting the regulating rod position, the APSR position, the AXIAL POWER IMBALANCE and the QPT. This potential for fuel-cladding damage exists because changes in the power distribution can cause increased power peaking and corresponding increased local linear heat rates.

The SDM requirement is assured by limiting the regulating and safety rod insertion limits so that the allowable inserted worth of the rods is such that sufficient reactivity is available in the rods to shut down the reactor to hot zero power with a reactivity margin which assumes that the maximum worth rod remains fully withdrawn upon trip (Ref. 5).

Operation at the regulating rod insertion limits and/or the AXIAL POWER IMBALANCE limits may cause the core power to approach the maximum linear heat generation rate or peaking factor with the allowed QPT present. Operation at the regulating rod insertion limit may also indicate that the

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

maximum ejected rod worth could be equal to the limiting value in fuel cycles which have sufficiently high ejected rod worths.

The regulating and safety rod insertion limits ensure that the safety analyses assumptions for SDM, ejected rod worth, and power distribution peaking factors remain valid. (Ref. 4).

The regulating rod insertion limits satisfy Criterion 2 of the NRC Interim Policy Statement, in that they are initial conditions assumed in the safety analysis.

LCO

The limits on the safety and regulating rod sequences, including their overlap and insertion positions as defined in the COLR (Ref. 3), must be maintained because they ensure that the resulting power distribution will be within the range of analyzed power distributions, ensure that the SDM is maintained, ensure that the ejected rod worth is maintained, and ensures that there will be adequate negative reactivity inserted by a reactor 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 rod motion.

Error-adjusted maximum allowable setpoints for regulating rod insertion and APSR insertion positions are provided in the COLR. The setpoints are derived by an adjustment of the measurement system-independent limits given in the COLR to allow for THERMAL POWER level uncertainty and rod position errors.

Actual alarm setpoints implemented in the unit may be more restrictive than the maximum allowable setpoint values to provide additional conservatism between the actual alarm setpoint and the measurement system-independent limit.

[For this facility, an OPERABLE regulating rod insertion alarm and an OPERABLE regulating rod group constitute the following:]

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BASES (continued)

LCO (continued)

[For this facility, the following support systems are required to be OPERABLE to ensure regulating rod insertion alarm and regulating rod group OPERABILITY:]

[For this facility, the required support systems, which upon their failure do not result in the inoperability of the regulating rod insertion alarm and the regulating rod group, and the justifications are as follows:]

APPLICABILITY

The regulating rod sequence, overlap and physical insertion limits shall be maintained with the reactor in MODES 1 and 2. These limits must be maintained since they maintain the validity of the assumed power distribution, ejected rod worth, SDM, and reactivity rate insertion assumptions used in the safety analyses. Applicability in MODES 3, 4, and 5 is not required, since neither the power distribution nor ejected rod worth assumptions would be exceeded in these MODES.

This LCO has been modified by a Note which states that this LCO is not applicable during the performance of SR 3.1.4.2. This surveillance requires that the rods be moved at least every 92 days to verify their OPERABILITY. The individual rods are moved at least 3% of the core height and then returned to their original position.

ACTIONS

General

The regulating rod insertion alarm setpoints provided in the COLR (Ref. 3) are based on both the initial conditions assumed in the accident analyses and on the SDM. Specifically, separate insertion limits are specified to determine whether the unit is operating in violation of the initial conditions (e.g., the range of power distributions) assumed in the accident analyses or whether the unit is in violation of the SDM. Separate insertion limits are provided because different Required Actions and Completion Times apply, depending on which insertion limit has been violated. The area between the boundaries of acceptable operation and unacceptable operation, illustrated on the regulating rod insertion limit figures in the COLR, is

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BASES (continued)

ACTIONS (continued)

called the restricted region. The actions required when operation in the restricted region occurs are described under Condition A. The actions required when operation in the unacceptable region occurs are described under Condition C.

A.1

Operation with the regulating rods in the restricted region shown on the regulating rod insertion figures in the COLR or with any group sequence or overlap outside the limits of this LCO potentially violates the LOCA linear heat rate limits ($F_Q(Z)$ limits), loss-of-flow accident DNB peaking limits ($F^N_{\Delta H}$ limits), or ejected rod worth limits. The design calculations assume no deviation in nominal overlap between regulating rod banks. However, deviations of 5% of the core height above or below the nominal overlap may be typical and will not cause significant differences in core reactivity, in power distribution, or in rod worth, relative to the design calculations. The group sequence must be maintained because design calculations assume the regulating rods withdraw and insert in a pre-determined order.

To verify that $F_Q(Z)$ and $F^N_{\Delta H}$ are within their limits, SR 3.2.5.1 is performed using the Incore Detector System to obtain a three-dimensional power distribution map. Verification that $F_Q(Z)$ and $F^N_{\Delta H}$ are within their limits ensures that operation with the regulating rods inserted into the restricted region does not violate the ECCS or DNB criteria (Ref. 6). The required Completion Time of 2 hours is acceptable in that it allows the operator sufficient time to obtain a power distribution map and to verify the power peaking factors. Repeating SR 3.2.5.1 every 2 hours is acceptable because it ensures that continued verification of the power peaking factors is performed as core conditions (primarily regulating rod insertion and induced xenon redistribution) change.

Monitoring the power peaking factors $F_Q(Z)$ and $F^N_{\Delta H}$ does not provide verification that the reactivity insertion rate on the rod trip or the ejected rod worth limit is maintained, since worth is a reactivity parameter rather than a power peaking parameter. However, if the COLR figures do not show a rod insertion limit labelled as ejected rod worth limited, then the ejected rod worth is no more limiting than the SDM

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BASES (continued)

ACTIONS (continued)

based rod insertion limit in the core design (Ref. 7). Ejected rod worth limits are independently maintained by the Required Actions of Condition C.

[For this facility, the regulating rods can be restored to within acceptable operating limits by the following actions:]

A.2

Indefinite operation with the regulating rods inserted in the restricted region, or in violation of the group sequence or overlap limits is not prudent because [ ]. Even if power peaking monitoring per Required Action A.1 is continued, reactivity limits may not be met and the abnormal regulating rod insertion or group configuration may cause an adverse xenon redistribution, may cause the limits on AXIAL POWER IMBALANCE to be exceeded, or may adversely affect the long-term fuel depletion pattern. Therefore, power peaking monitoring is allowed for up to 4 hours after discovery of failure to meet the requirements of this LCO. This required Completion Time is reasonable based on the low probability of an event occurring simultaneously with the limit out of specification in this relatively short time period. In addition, it precludes long-term depletion with abnormal group insertions or configurations, thereby limiting the potential for an adverse xenon redistribution.

B.1

If the regulating rods cannot be restored within the acceptable operating limits shown on the figures in the COLR within the required Completion Time (i.e., Required Action A.2 not met), then the limits can be restored by reducing the THERMAL POWER to a value allowed by the regulating rod insertion limits in the COLR. The required Completion Time of 2 hours is sufficient to allow the operator to complete the power reduction in an orderly manner and without challenging the plant systems. Operation for up to 2 more hours in the restricted region shown in the COLR is acceptable based on the low probability of an event occurring simultaneously with the limit out of specification in this relatively short time period. In addition, it precludes long-term depletion with abnormal group insertions

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BASES (continued)

ACTIONS (continued)

or configurations, and limits the potential for an adverse xenon redistribution.

C.1

Operation in the unacceptable region shown on the figures in the COLR corresponds to power operation with an SDM less than the minimum required value. Restoration of the required SDM requires increasing the RCS boron concentration because the regulating rods may be inserted too far to provide sufficient negative reactivity insertion following a scram. The RCS boration must occur as described in Section B 3.1.1. The required Completion Time of 15 minutes to initiate boration is reasonable based on limiting the potential xenon redistribution, the low probability of an accident occurring in this relatively short time period and the number of steps required to complete this ACTION. This allows the operator sufficient time to align the required valves and to start the boric acid pumps. Boration will continue until the regulating rod group positions are restored to at least within the restricted operational region, which restores the minimum SDM capability.

C.2

The required Completion Time of 2 hours from initial discovery of a regulating rod group in the unacceptable region until its restoration to within the restricted operating region shown on the figures in the COLR allows sufficient time for borated water to enter the RCS from the chemical addition and makeup systems, thereby allowing the regulating rods to be withdrawn to the restricted region. Operation in the restricted region for up to an additional 2 hours is reasonable based on limiting the potential for an adverse xenon redistribution, the low probability of an accident occurring in this relatively short time period, and the number of steps required to complete this ACTION.

In the event that the regulating rods position indication system is found to be inoperable the regulating rod is considered to be not within limits and Required Action C.2 and LCO 3.1.4 apply.

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BASES (continued)

ACTIONS (continued)

D.1

If the regulating rods cannot be restored to within the acceptable operating limits for the original THERMAL POWER, or if the power reduction cannot be completed within the required Completion Time, then the reactor is placed in MODE 3, in which this LCO does not apply. This ACTION ensures that the reactor does not continue operating in violation of the peaking limits, the ejected rod worth, the reactivity insertion rate assumed as initial conditions in the accident analyses, or the required minimum SDM assumed in the accident analyses. The required Completion Time of 6 hours is reasonable based on operating experience regarding the amount of time required to reach MODE 3 from RATED THERMAL POWER without challenging the plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1

When the regulating rod banks are maintained within their insertion limits as verified by SR 3.2.1.2 as specified in the COLR, it is unlikely that their sequence and overlap will not be in accordance with their required limits in the COLR. Nevertheless, this surveillance provides the necessary assurance that the sequence and overlap limits are not violated. A surveillance Frequency of 12 hours or 4 hours depending on whether the CONTROL ROD drive sequence alarm is OPERABLE or not, is acceptable because [ ]. Also, the frequency takes into account other information available to the operator in the control room that monitors the status of the regulating rods.

This surveillance is modified by a Note which states that SR 3.0.4 is not applicable since the unit must be in the applicable MODES in order to perform surveillances which demonstrate that the limits of this LCO are met.

SR 3.2.1.2

With an OPERABLE regulating rod insertion limit alarm, verification of the regulating rod insertion limits as specified in the COLR at a Frequency of 12 hours is sufficient to ensure the OPERABILITY of the regulating rod insertion limit alarm and to detect regulating rod banks

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

that may be approaching the bank insertion limits since very little rod motion occurs in 12 hours due to fuel burnup. If the insertion limit alarm becomes inoperable, verification of the regulating rod bank position at a Frequency of 4 hours is sufficient to detect whether the regulating rod banks may be approaching or exceeding the regulating rod banks insertion limit because [ ]. Also, the frequency takes into account other information available to the operator in the control room that monitor the status of the regulating rods.

SR 3.2.1.2 is modified by a Note which states that SR 3.0.4 is not applicable since the unit must be in the applicable MODES in order to perform surveillances which demonstrate the LCO limits are met.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," GDC 10, "Reactor Design," and GDC 26, "Reactivity Control System Redundancy and Capability."

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

3. [Unit Name] Core Operating Limits Report, "[Title]."

4. [Unit Name] FSAR,

   a. Section [ ], "Rod Ejection Accident, Accident Bases."

   b. Section [ ], "Rod Ejection Accident, Fuel Rod Damage."

   c. Section [ ], "Thermal and Hydraulic Limits."

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

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

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BASES (continued)

REFERENCES (continued)	7. [Unit Name] FSAR, Section [ ], "[Title]."

B 3.2 POWER DISTRIBUTION LIMIT

B 3.2.2 AXIAL POWER SHAPING ROD (APSR) Insertion Limits

BASES

BACKGROUND

The insertion limits of the APSRs are initial assumptions in all safety analyses which are affected by core power distributions. The applicable criteria for these power distribution design requirements are 10 CFR 50, Appendix A, GDC 10 "Reactor Design" (Ref. 1), and 10 CFR Part 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plants" (Ref. 2).

Limits on the APSRs rod insertion have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution defined by the design power peaking limits are maintained.

The power density at any point in the core must be limited to maintain specified acceptable fuel design limits, including limits that meet the criteria specified in 10 CFR 50.46 (Ref. 2). Together, LCO 3.2.1, LCO 3.2.2, LCO 3.2.3, and LCO 3.2.4 provide limits on control component operation and on monitored process variables to ensure that the core operates within the Nuclear Heat Flux Hot Channel Factor ($F_Q(Z)$) and the Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$) limits in the CORE OPERATING LIMITS REPORT (COLR) (Ref. 3). Operation within the $F_Q(Z)$ limits given in the COLR prevents power peaks that would exceed the loss-of-coolant accident (LOCA) limits derived from the analysis of the EMERGENCY CORE COOLING SYSTEM (ECCS). Operation within the $F^N_{\Delta H}$ limits given in the COLR prevents Departure from Nucleate Boiling (DNB) during a loss-of-forced-reactor-coolant-flow accident. In addition to the $F_Q(Z)$ and $F^N_{\Delta H}$ limits, certain reactivity limits are maintained by regulating rod insertion limits. The regulating rod insertion limits also restrict the ejected rod worth to the values assumed in the safety analysis and preserve the minimum required SHUTDOWN MARGIN (SDM) in MODES 1 and 2. The APSRs are not required for reactivity insertion rate on trip or SDM and, therefore, they do not trip on scram.

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BASES (continued)

BACKGROUND (continued)

The regulating rod insertion and alignment limits, AXIAL POWER IMBALANCE, QUADRANT POWER TILT (QPT), and APSR position are process variables that together characterize and control the three-dimensional power distribution of the reactor core. Additionally, the safety and regulating rod insertion limits control the reactivity that could be added in the event of a rod ejection accident, and assure that 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 rod, or other postulated accident requiring termination by a Reactor Trip System (RTS) trip function.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation (Condition 1) and anticipated operational occurrences (AOOs) (Condition 2). The acceptance criteria for the safety and regulating rod insertion, APSR position, AXIAL POWER IMBALANCE, and QPT LCOs preclude core power distributions from occurring which 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 (10 CFR 50.46, Ref. 2);

b. During a loss-of-forced-reactor-coolant-flow accident, there must be at least a 95% probability at a 95% confidence level that the hot fuel rod in the core does not experience a DNB condition. This is referred to hereafter as the 95/95 DNB criterion;

c. During an ejected rod accident, the fission energy input to the fuel must not exceed 280 cal/gm (Ref. 4b); 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 (GDC 26, Ref. 1).

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

Fuel-cladding damage does not occur when the core is operated outside these LCOs during normal operation. However, fuel-cladding damage could result should an accident occur with simultaneous violation of one or more of these LCOs. This potential for fuel-cladding damage exists because changes in the power distribution can cause increased power peaking and corresponding increased local linear heat rates.

Operation at the APSR insertion limits or AXIAL POWER IMBALANCE limits may approach the maximum allowable linear heat generation rate or peaking factor with the allowed QPT present. Operation at the insertion limit may also indicate that the maximum ejected rod worth could be equal to the limiting value in fuel cycles that have sufficiently high ejected rod worths.

The APSR insertion limits satisfy Criterion 2 of the NRC Interim Policy Statement, in that they are initial conditions assumed in the safety analysis.

LCO

The limits on APSR physical insertion as defined in the COLR (Ref. 3) must be maintained because they serve the function of maintaining the power distribution within an acceptable range.

Error-adjusted maximum allowable setpoints for APSR insertion are provided in the COLR. The setpoints are derived by adjustment of the measurement system-independent limits given in the COLR to allow for THERMAL POWER level uncertainty and rod position errors.

Actual alarm setpoints implemented in the unit may be more restrictive than the maximum allowable setpoint values to allow for additional conservatism between the actual alarm setpoint and the measurement system-independent limit.

[For this facility, an OPERABLE APSR insertion alarm and an OPERABLE APSR constitute the following:]

[For this facility, the following support systems are required to be OPERABLE to ensure APSR insertion alarm and APSR OPERABILITY:]

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BASES (continued)

LCO (continued)

[For this facility, the required support systems, which upon their failure do not result in the inoperability of the APSR insertion alarm and APSR inoperability, and the justifications are as follows:]

APPLICABILITY

The APSR physical insertion limits shall be maintained with the reactor in MODES 1 and 2. These limits must be maintained since they maintain the power distribution within the range assumed in the accident analyses. In MODE 1, the limits on APSR insertion specified by this LCO must be maintained to maintain the axial fuel burnup design conditions assumed in the reload safety evaluation analysis. The fuel cycle design assumes APSR withdrawal at the effective full power day (EFPD) burnup window specified in the COLR. Prior to this window, the APSRs cannot be maintained fully withdrawn in steady-state operation. After this window, the APSRs are not allowed to be reinserted for the remainder of the fuel cycle. In MODE 2, applicability is required since the reactor can be in a slightly subcritical condition (i.e., $K_{eff} > 0.99$). Applicability in MODES 3, 4, and 5 is not required, since neither the power distribution nor ejected rod worth assumptions in the accident analyses would be exceeded in these MODES.

ACTIONS

A.1

To verify that both the core parameters $F_Q(Z)$ and $F^N_{\Delta H}$ are within their limits, SR 3.2.5.1 is performed using the Incore Detector System to obtain a three-dimensional power distribution map. Successful verification that $F_Q(Z)$ and $F^N_{\Delta H}$ are within their limits ensures that operation with the APSRs inserted or withdrawn in violation of the times specified in the COLR do not violate either the ECCS or DNB criteria (Ref. 5). The required Completion Time of 2 hours is reasonable to allow the operator to obtain a power distribution map and to verify the power peaking factors. Repeating SR 3.2.5.1 every 2 hours is reasonable to ensure that continued verification of the power peaking factors is obtained as core conditions (primarily the

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BASES (continued)

ACTIONS (continued)

regulating rod insertion and induced xenon redistribution) change.

In the event that the APSR position indication system is found to be inoperable the APSR is considered to be not within limits and Required Actions A.1, A.2 and LCO 3.1.4 apply.

A.2

Indefinite operation with the APSRs inserted or withdrawn in violation of the times specified in the COLR is not prudent. Even if power peaking monitoring per required Action A.1 is continued, the abnormal APSR insertion or withdrawal may cause an adverse xenon redistribution, or may cause the limits on AXIAL POWER IMBALANCE to be exceeded, or may affect the long-term fuel-depletion pattern. Therefore, power peaking monitoring is allowed for up to 4 hours. This required Completion Time is reasonable based on the low probability of an event occurring simultaneously with the APSR limit out of specification. In addition, it precludes long-term depletion with the APSRs in positions which have not been analyzed thereby limiting the potential for an adverse xenon redistribution. This time limit also ensures that a significant unintended burnup distribution does not develop, and allows the operator sufficient time to reposition the APSRs to correct their position.

Since the APSRs are not operated by the automatic control system, manual action by the operator is required to restore the APSRs to the position specified in the COLR.

B.1

If the APSRs cannot be restored to their intended position within the required Completion Time of 4 hours, then the reactor must be placed in MODE 3, in which this LCO does not apply. This ACTION ensures that the fuel does not continue to be depleted in an unintended burnup distribution. The required Completion Time of 6 hours is reasonable based on operating experience regarding the time required to reach MODE 3 from RATED THERMAL POWER in an orderly manner and without challenging plant systems.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1

When the plant computer is OPERABLE, the operator will receive a computer alarm if the APSRs insert after that time in core life when the APSR withdrawal occurs. Verification that the APSRs are within their insertion limits at a 12-hour Frequency is sufficient to ensure that the APSRs insertion limits are preserved and the computer alarm remains OPERABLE because [ ]. Also, the frequency takes into account other information available to the operator in the control room that monitors the status of the regulating rods.

This surveillance is modified by a Note which states that SR 3.0.4 is not applicable since the unit must be in the applicable MODES in order to perform surveillances which demonstrate that the limits of this LCO are met.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," GDC 10, "Reactor Design," GDC 26, "Reactivity Control System Redundancy and Capability."
2. Title 10, Code of Federal Regulations, Part 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors."
3. [Unit Name] Core Operating Limits Report, [Title].
4. [Unit Name] FSAR,
   a. Section [ ], "[Rod Ejection Accident, Accident Bases]."
   b. Section [ ], "[Rod Ejection Accident, Fuel Rod Damage]."
   c. Section [ ], "[Thermal and Hydraulic Limits]."
5. [Unit Name] FSAR, Section [ ], "[Title]."

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3 AXIAL POWER IMBALANCE Operating Limits

BASES

BACKGROUND

The purpose of these MODE 1 power distribution limits is to establish any LCO which limits the core power distribution based on accident initial condition criteria.

The power density at any point in the core must be limited to maintain specified acceptable fuel design limits, including limits that satisfy the criteria specified in 10 CFR 50.46 (Ref. 1). This LCO provides limits on control component operation and on monitored process variables to ensure the core operates within the NUCLEAR HEAT FLUX HOT CHANNEL FACTOR ($F_Q(Z)$) and the NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR ($F^N_{\Delta H}$) limits in the CORE OPERATING LIMITS REPORT (COLR) (Ref. 2). Operation within the $F_Q(Z)$ limits given in the COLR prevents power peaks that would exceed the loss-of-coolant accident (LOCA) limits derived from the analysis of the EMERGENCY CORE COOLING SYSTEM (ECCS). Operation within the $F^N_{\Delta H}$ limits given in the COLR prevents Departure from Nucleate Boiling (DNB) during a loss-of-forced-reactor-coolant-flow accident. In addition to the $F_Q(Z)$ and $F^N_{\Delta H}$ limits, certain reactivity limits are preserved by regulating rod insertion limits. The regulating rod insertion limits also restrict the ejected CONTROL ROD worth to the values assumed in the safety analysis and preserve the minimum required SHUTDOWN MARGIN (SDM) in MODES 1 and 2.

This LCO is required to limit fuel-cladding failures that would breach the primary fission product barrier and release fission products into the reactor coolant in the event of a LOCA, loss-of-forced-reactor-coolant-flow, ejected control rod, or other postulated accident requiring termination by a Reactor Protection Systems (RPS) trip function. This LCO limits the amount of damage to the fuel cladding during an accident by maintaining the validity of the assumptions in the safety analyses related to the initial power distribution and reactivity.

Fuel-cladding failure during a postulated LOCA is limited by restricting the maximum linear heat generation rate so that the peak cladding temperature does not exceed 2200°F

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BASES (continued)

BACKGROUND (continued)

(Ref. 1). Peak cladding temperatures greater than 2200°F would cause severe cladding failure by oxidation due to a Zircaloy-water reaction.

Proximity to the DNB condition is expressed by the Departure from Nucleate Boiling Ratio (DNBR), defined as the ratio of the cladding surface heat flux required to cause DNB to the actual cladding surface heat flux. The minimum DNBR value during both normal operation and anticipated transients is limited to the DNBR correlation limit for the particular fuel design in use and is accepted as an appropriate margin to DNB. The DNB correlation limit ensures that there is a 95% probability at a 95% confidence level that the hot fuel rod in the core does not experience DNB.

Fuel failure during a postulated ejected control rod accident is limited by restricting the maximum possible worth of a potential ejected rod to 0.65% $\Delta k/k$ at hot full power and 1.00% $\Delta k/k$ at hot zero power (Ref. 3a). If the ejected rod worth is limited to these values, the fission energy input during the accident will not exceed 280 cal/gm (Ref. 3b). Below 280 cal/gm, the fuel rods can be expected to remain intact; above this value, fuel fragmentation may occur.

The measurement system-independent limits on AXIAL POWER IMBALANCE are determined directly by the reload safety evaluation analysis without any adjustment for measurement system error and uncertainty. Operation beyond these limits could invalidate the assumptions used in the accident analyses regarding the core power distribution. The error-adjusted maximum allowable alarm setpoints (measurement system-dependent limits) for the AXIAL POWER IMBALANCE are specified in the COLR.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation (Condition 1) and anticipated operational occurrences (AOOs) (Condition 2). This LCO precludes core power distributions from occurring which would violate the following fuel design criteria:

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

a. During a large-break LOCA, the peak cladding temperature must not exceed a limit of 2200°F (Ref. 1);

b. During a loss-of-forced-reactor-coolant-flow accident, there must be at least a 95% probability at a 95% confidence level that the hot fuel rod in the core does not experience a DNB condition. This is referred to hereafter as the 95/95 DNB criterion;

c. During an ejected-rod accident, the fission-energy input to the fuel must not exceed 280 cal/gm (Ref. 3b); and

d. The CONTROL RODS must be capable of shutting down the reactor with a minimum required SDM with the highest worth CONTROL ROD stuck fully withdrawn (Ref. 4).

The regulating rod positions, the AXIAL POWER SHAPING ROD (APSR) positions, the AXIAL POWER IMBALANCE, and the QUADRANT POWER TILT (QPT) are process variables which characterize and control the three-dimensional power distribution of the reactor core.

Fuel-cladding damage does not occur when the core is operated outside this LCO during normal operation. However, fuel-cladding damage could result should an accident occur with simultaneous violation of one or more of the LCOs governing the four process variables cited above. This potential for fuel-cladding damage exists because changes in the power distribution can cause increased power peaking and corresponding increased local linear heat rates (LHRs).

The regulating rod insertion, the APSR positions, the AXIAL POWER IMBALANCE, and the QPT are monitored and controlled during power operation to ensure that the power distribution is within the bounds set by the safety analyses. The axial power distribution is maintained primarily by the AXIAL POWER IMBALANCE and the APSR position limits, and the radial power distribution is maintained primarily by the QPT limits. The regulating rod insertion limits affect both the radial and axial power distribution.

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

The dependence of the core power distribution on burnup, regulating rod insertion, APSR position, and spatial xenon distribution are taken into account when performing the reload safety evaluation analysis. APSR position limits are not imposed for operation with gray (Inconel) APSRs, except for compliance with APSR withdrawals that may designed into the fuel cycle.

Operation at the AXIAL POWER IMBALANCE or the rod insertion limits must be interpreted as operating the core at the maximum allowable $F_Q(Z)$ or $F^N_{\Delta H}$ peaking factors assumed as initial conditions for the accident analyses with the allowed QPT present.

The limits on the AXIAL POWER IMBALANCE satisfy Criterion 2 of the NRC Interim Policy Statement because they reflect changes in power peaking, which is input as an initial condition in the ECCS analysis and the plant safety analysis. The limits for each cycle are contained in the COLR.

LCO

The power distribution LCO limits have been established based on correlations between power peaking and easily measured process variables: the regulating rod position, the APSR positions, the AXIAL POWER INBALANCE and the QPT. The regulating rod insertion limits and the AXIAL POWER IMBALANCE, boundaries contained in the COLR represent the measurement system-independent limits at which the core power distribution would either exceed the LOCA LHR limits or cause a reduction in the DNBR below the Safety Limit during the loss-of-flow accident with the allowable QPT present and with the APSR positions consistent with the limitations on APSR withdrawal determined by the fuel cycle design and specified by LCO 3.2.2.

Operation beyond the power distribution-based LCO limits for the corresponding allowable THERMAL POWER and simultaneous occurrence of either the LOCA, loss-of-forced-reactor-coolant-flow accident, or ejected rod accident has an acceptably low probability. Therefore, if the LCO limits are violated, a short time is allowed for corrective action before a significant power reduction is required.

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BASES (continued)

LCO (continued)

The AXIAL POWER IMBALANCE maximum allowable setpoints (measurement system-dependent limits) applicable for the full incore detector system, the minimum incore detector system, and the excore detector system are provided in the COLR.

Actual alarm setpoints implemented in the unit may be more restrictive than the maximum allowable setpoint values to provide additional conservatism between the actual alarm setpoints and the measurement system-independent limit.

[For this facility, OPERABLE incore and excore detector systems constitute the following:]

[For this facility, the following support systems are required to be OPERABLE to ensure OPERABILITY of the incore and excore detector systems, and to ensure that AXIAL POWER IMBALANCE operating limits are met:]

[For this facility, the required support systems which, upon their failure, do not result in the incore or excore detector systems being inoperable, or in the AXIAL POWER IMBALANCE operating limits not being met, and the justifications are as follows:]

APPLICABILITY

In MODE 1, the limits on AXIAL POWER IMBALANCE must be maintained when THERMAL POWER is > 40% of RATED THERMAL POWER (RTP) in order to preclude the core power distribution from exceeding the LOCA and loss-of-flow assumptions used in the accident analyses. Applicability of these limits below 40% of RTP in MODE 1 is not required. This is acceptable because the combination of the AXIAL POWER IMBALANCE with the maximum ALLOWABLE THERMAL POWER level will not result in LHRs sufficiently large so as to violate the fuel design limits. In MODES 2, 3, 4, 5, and 6, this LCO is not applicable because the reactor is not generating sufficient THERMAL POWER which could lead to fuel damage.

In MODE 1 it may be necessary to suspend the AXIAL POWER IMBALANCE limits during PHYSICS TESTS per LCO 3.1.8. Suspension of these limits is permissible because the reactor protection criteria are maintained by the

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BASES (continued)

APPLICABILITY (continued)

remaining LCOs governing the three-dimensional power distribution and by the surveillances required by LCO 3.1.8 and LCO 3.1.9.

ACTIONS

A.1

The AXIAL POWER IMBALANCE operating limits that maintain the validity of the assumptions regarding the power distributions in the accident analyses of the LOCA and the loss-of-flow accident are provided in the COLR. Operation within the AXIAL POWER IMBALANCE limits given in the COLR is labeled as the acceptable region of operation. Operation in violation of the AXIAL POWER IMBALANCE limits given in the COLR is labeled as the restricted region of operation.

Operation with AXIAL POWER IMBALANCE in the restricted region shown on the AXIAL POWER IMBALANCE figures in the COLR potentially violates the LOCA LHR limits ($F_Q(Z)$ limits) or the loss-of-flow accident DNB peaking limits ($F^N_{\Delta H}$), or both. To verify that $F_Q(Z)$ and $F^N_{\Delta H}$ are within their specified limits, SR 3.2.5.1 is performed using the Incore Detector System to obtain a three-dimensional power distribution map. Verification that $F_Q(Z)$ and $F^N_{\Delta H}$ are within their specified limits ensures that operation with the AXIAL POWER IMBALANCE in the restricted region does not violate the ECCS or DNB criteria. The required Completion Time of 2 hours is a reasonable amount of time to allow the operator to obtain a power distribution map and to determine and verify that the power peaking factors are within their specified limits. The 2-hour Frequency is a reasonable amount of time to ensure that continued verification of the power peaking factors is obtained as core conditions (primarily regulating rod insertion and induced xenon redistribution) change because [ ].

A.2

Indefinite operation with the AXIAL POWER IMBALANCE in the restricted region is not prudent. Even if power peaking monitoring per required Action A.1 is continued, excessive AXIAL POWER IMBALANCE over an extended period of time may cause an adverse xenon redistribution to occur. Therefore,

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BASES (continued)

ACTIONS (continued)

power peaking monitoring is only allowed for a maximum of 4 hours. This required Completion Time is reasonable based on the low probability of a limiting event occurring simultaneously with the AXIAL POWER IMBALANCE outside the limits of this LCO. In addition, this limited Completion Time precludes long-term depletion of the reactor fuel with excessive AXIAL POWER IMBALANCE, thereby limiting the potential for an adverse xenon redistribution. [For this facility, AXIAL POWER IMBALANCE is restored to within limits by the following actions:]

The 4-hour Completion Time for restoring AXIAL POWER IMBALANCE within acceptable limits gives the operator sufficient time to reposition the APSRs or regulating rods to the reduce AXIAL POWER IMBALANCE because [ ].

B.1

If the Required Actions and their associated Completion Times of Condition A cannot be met, or if AXIAL POWER IMBALANCE cannot be determined due to support system (i.e., Incore or Excore Detector System) inoperability, the AXIAL POWER IMBALANCE may exceed its specified limits and the reactor may be operating with a global axial power distribution mismatch. Continued operation in this configuration may induce an axial xenon oscillation and may result in increased linear heat generation when the xenon redistributes. Reducing THERMAL POWER to be at most 40% of RTP will reduce the maximum LHR to a value which will not exceed the $F_Q(Z)$ and $F^N_{\Delta H}$ initial condition limits assumed in the accident analyses. The required Completion Time of 2 hours is reasonable based on limiting a potentially adverse xenon redistribution, the low probability of an accident occurring in this relatively short time period, and the number of steps required to complete this ACTION.

SURVEILLANCE REQUIREMENTS

General

The AXIAL POWER IMBALANCE can be monitored by both the Incore and Excore Detector Systems. The AXIAL POWER IMBALANCE maximum allowable setpoints are derived from their corresponding measurement system-independent limits by adjusting for both the system observability errors and

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

instrumentation errors. Although they may be based on the same measurement system-independent limits, the setpoints for the different systems are not identical because of differences in the errors applicable for each of these systems. The uncertainty analysis which defines the required error adjustment to convert the measurement system-independent limits to alarm setpoints assumes that 75% of the detectors in each quadrant are OPERABLE. Detectors located on the core major axes are assumed to contribute one-half of their output to each quadrant, and detectors in the center assembly are assumed to contribute one-quarter of their output to each quadrant. For AXIAL POWER IMBALANCE measuremonts using the Incore Detector System, the Minimum Incore Detector System consists of OPERABLE detectors configured as follows:

a. Nine detectors shall be arranged such that there are three detectors in each of three strings and there are three detectors lying in the same axial plane, with one plane at the core midplane and one plane in each axial core half;

b. The axial planes in each core half shall be symmetrical about the core midplane; and

c. The detector strings shall not have radial symmetry.

Figure B 3.2.3-1 depicts an example of this configuration (Minimum Incore System for AXIAL POWER IMBALANCE Measurement). This arrangement is chosen to reduce the uncertainty in the measurement of the AXIAL POWER IMBALANCE with the Minimum Incore Detector System. For example, the requirement for placing one detector of each of the three strings at the core midplane puts three detectors in the central region of the core where the neutron flux tends to be higher. It also helps prevent measuring axial offsets that are excessively large when the reactor is operating at low THERMAL POWER levels. The third requirement for placement of detectors (i.e., radial asymmetry) reduces uncertainty by measuring the neutron flux at core locations which are not radially symmetric.

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BASES (continued)

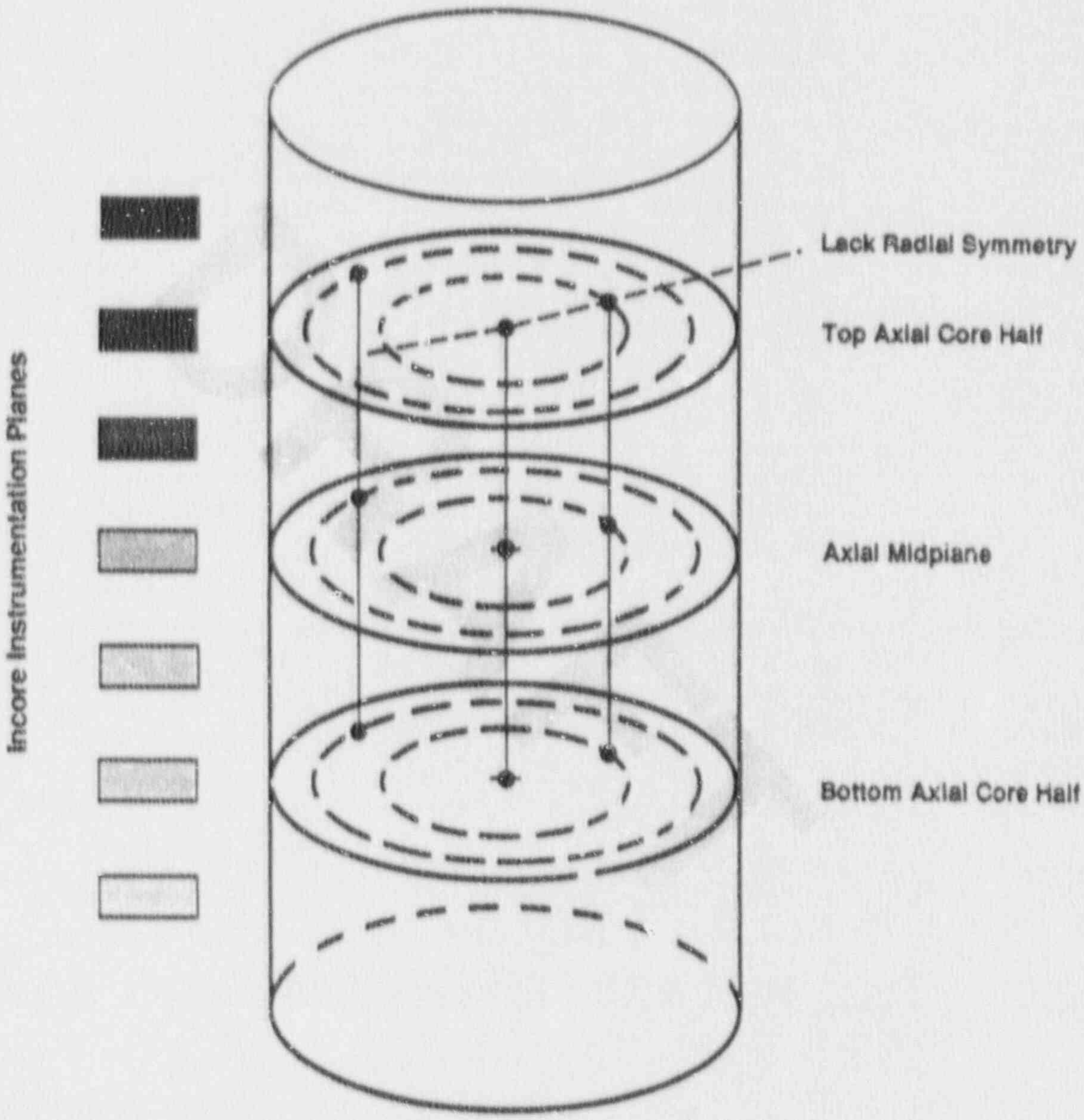

Figure B 3.2.3-1 (Page 1 of 1)

Minimum Incore System for AXIAL POWER IMBALANCE Measurement

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.2.3.1

Should the plant computer become inoperable, then the Excore System or Minimum Incore Detector System may be used to monitor the AXIAL POWER IMBALANCE. Since these systems do not provide a direct calculation and display of the AXIAL POWER IMBALANCE, performing the calculations at a 1-hour Frequency is a reasonable amount of time between calculations to detect any trends in the AXIAL POWER IMBALANCE which may exceed its alarm setpoint, and to provide the operator with sufficient time to undertake corrective action(s).

When the Full Incore Detector System is OPERABLE, the operator will receive an alarm if the AXIAL POWER IMBALANCE increases to its alarm setpoint. When the AXIAL POWER IMBALANCE is less then the alarm setpoint, verification of the AXIAL POWER IMBALANCE indication every 12 hours ensures that the AXIAL POWER IMBALANCE limits are not violated and verifies that the alarm system is OPERABLE. This surveillance Frequency is acceptable because the mechanisms that can cause AXIAL POWER IMBALANCE, such as xenon redistribution or CONTROL ROD drive mechanism malfunctions which cause slow AXIAL POWER IMBALANCE increases, will be discovered by the operator before the specified limits are violated.

REFERENCES

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

2. [Unit Name] Core Operating Limits Report, "[Title]."

3. [Unit Name] FSAR,

   a. Section [ ], "Rod Ejection Accident, Accident Bases."

   b. Section [ ], "Rod Ejection Accident, Fuel Rod Damage."

   c. Section [ ], "Thermal and Hydraulic Limits."

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BASES (continued)

REFERENCES (continued)	4.	Title 10, Code of Federal Regulations, Part 50, Appendix A, GDC 26, "Reactivity Control System Redundancy and Capability."

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 QUADRANT POWER TILT (QPT)

BASES

BACKGROUND

This LCO is required to limit the core power distribution based on accident initial condition criteria.

The power density at any point in the core must be limited to maintain specified acceptable fuel design limits, including limits that preserve the criteria specified in 10 CFR 50.46 (Ref. 1a). Together, LCO 3.2.1, LCO 3.2.2, LCO 3.2.3, LCO 3.2.4, and this LCO provide a limit on control component operation and on monitored process variables to ensure the core operates within the NUCLEAR HEAT FLUX HOT CHANNEL FACTOR ($F_Q(Z)$) and the NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR ($F^N_{\Delta H}$) limits in the CORE OPERATING LIMITS REPORT (COLR) (Ref. 2). Operation within the $F_Q(Z)$ limits given in the COLR prevents power peaks that would exceed the loss-of-coolant accident (LOCA) limits derived by Emergency Core Cooling Systems (ECCS) analysis. Operation within the $F^N_{\Delta H}$ limits given in the COLR prevents Departure from Nucleate Boiling (DNB) during a loss-of-forced-reactor- coolant-flow accident. In addition to the $F_Q(Z)$ and $F^N_{\Delta H}$ limits, certain reactivity limits are preserved by regulating rod insertion limits. The regulating rod insertion limits also restrict the ejected CONTROL ROD worth to the values assumed in the safety analysis and preserve the minimum required SHUTDOWN MARGIN (SDM) in MODES 1 and 2.

The regulating rod groups operate with a predetermined amount of position overlap in order to approximate a linear relation between rod worth and rod position (integral rod worth). The regulating rod groups are withdrawn and operate in a predetermined sequence. The automatic control system controls reactivity by moving the regulating rod groups in sequence within analyzed ranges. The group sequence and overlap limits are specified in the COLR.

This QPT LCO is required to limit 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-forced-reactor-coolant-flow, ejected CONTROL ROD, or other accident requiring termination by a

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BASES (continued)

BACKGROUND (continued)

Reactor Protection System (RPS) trip function. This LCO limits the amount of damage to the fuel cladding during an accident by maintaining the validity of the assumptions used in the safety analysis related to the initial power distribution and reactivity. These are called accident initial condition assumptions.

Fuel-cladding failure during a postulated LOCA is limited by restricting the maximum LINEAR HEAT GENERATION RATE (LHGR) so that the peak cladding temperature does not exceed 2200°F (Ref. 1a). Peak cladding temperatures greater than 2200°F would cause severe cladding failure by oxidation due to a Zircaloy-water reaction.

Proximity to the DNB condition is expressed by the departure from nucleate boiling ratio (DNBR), defined as the ratio of the cladding surface heat flux required to cause DNB to the actual cladding surface heat flux. The minimum DNBR value during both normal operation and anticipated transients is limited to the DNBR correlation limit for the particular fuel design in use, and is accepted as an appropriate margin to DNB. The DNB correlation limit ensures that there is a 95% probability at a 95% confidence level that the hot fuel rod in the core does not experience DNB.

Fuel failure during a postulated ejected CONTROL ROD accident is limited by restricting the maximum possible worth of a potential ejected rod to 0.65% Δk/k at hot full power (1.00% Δk/k at hot zero power) (Ref. 3a). If the ejected rod worth is limited to this [illegible] the fission energy input during the accident will [illegible]xceed 280 cal/gm (Ref. 3b). Below 280 cal/gm, the fue[illegible]s can be expected to remain intact; above this value, fue[illegible] fragmentation may occur.

The SDM requirement is assured by limiting the allowable inserted worth of the regulating CONTROL RODS so that sufficient reactivity is available in the rods to shut down the reactor to hot zero power with the minimum required reactivity margin with the maximum worth CONTROL ROD assembly stuck fully withdrawn.

The measurement system-independent limits on QPT are determined directly by the reload safety evaluation analysis without adjustment for measurement system error and

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BASES (continued)

BACKGROUND (continued)

uncertainty. Operation beyond these limits could violate the core power distribution during an accident. The error adjusted maximum allowable alarm setpoints (measurement system-dependent limits) for the QPT are specified in the COLR.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation (Condition 1) and anticipated operational occurrences (Condition 2). This LCO precludes 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 (10 CFR 50.46, Ref. 1a).

b. During a loss-of-forced-reactor-coolant-flow accident, there must be at least a 95% probability at a 95% confidence level that the hot fuel rod in the core does not experience a DNB condition. This is referred to hereafter as the 95/95 DNB criterion.

c. During an ejected rod accident, the fission energy input to the fuel must not exceed 280 cal/gm (Ref. 3b).

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 (GDC 26, Ref. 1b).

The regulating rod position, AXIAL POWER SHAPING ROD (APSR) position, AXIAL POWER IMBALANCE, and QPT are process variables that together characterize and control the three-dimensional power distribution of the reactor core.

Fuel-cladding damage does not occur when the core is operated outside this LCO during normal operation. However, fuel-cladding damage could result should an accident occur with simultaneous violation of one or more of the LCOs

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

governing the four process variables cited above. Changes in the power distribution can cause increased power peaking and corresponding increased local linear heat rates (LHRs).

Regulating rod insertion, APSR position, AXIAL POWER IMBALANCE, and QPT are monitored and controlled during power operation to ensure that the power distribution and reactivity remain within the bounds set by the safety analysis. The axial power distribution is maintained primarily by the AXIAL POWER IMBALANCE and the APSR position limits, and the radial power distribution is maintained primarily by the QPT limits. The regulating rod insertion limits affect both the radial and axial power distribution.

The dependence of the core power distribution on burnup, regulating rod insertion, APSR position, and spatial xenon distribution are taken into account during the reload safety evaluation analysis. An allowance for QPT is accommodated in the analysis and resultant LCO limits. The increase in peaking taken for QPT is developed from a database of full-core power distribution calculations (Ref. 4). The calculations consist of simulations of many power distributions with tilt-causing mechanisms (e.g., dropped CONTROL RODS, misaligned CONTROL RODS, broken APSR fingers fully inserted, misloaded assemblies, and burnup gradients). An increase of less than 2% peak power per 1% QPT is supported by the analysis, therefore a value of 2% peak power increase per 1% QPT is used to bound peak power increases due to QPT.

Operation at the AXIAL POWER IMBALANCE or rod insertion limits must be interpreted as operating the core at the maximum allowable $F_Q(Z)$ or $F^N_{\Delta H}$ peaking factor for accident initial conditions with the allowed QPT present. Operation at the rod insertion limit may also indicate the maximum ejected rod worth could be equal to the limiting value, in fuel cycles which have sufficiently high ejected rod worths. Operation with the regulating rod position at the SDM insertion limit indicates the SDM is equal to the minimum required value, and is assumed to decrease below the minimum required value if the regulating rods insert further.

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

The limits on QPT satisfy the requirements of Criterion 2 the NRC Interim Policy Statement because they reflect changes in power peaking, which are initial condition inputs to the ECCS analysis and plant safety analysis. The limits for each cycle are contained in the COLR.

LCO

The power distribution LCO limits have been established based on correlations between power peaking and easily measured process variables: regulating rod position, APSR position, AXIAL POWER IMBALANCE, and QPT. The regulating rod insertion limits and the AXIAL POWER IMBALANCE boundaries contained in the COLR represent the measurement system-independent limits at which the core power distribution would either exceed the LOCA LHR limits or cause a reduction in DNBR below the safety limit during the loss-of-flow accident with the allowable QPT present and with an APSR position consistent with the limitations on APSR withdrawal determined by the fuel cycle design and specified by LCO 3.2.2.

Operation beyond the power distribution-based LCO limits for the corresponding allowable THERMAL POWER and simultaneous occurrence of either the LOCA, loss-of-forced-reactor-coolant-flow accident, or ejected rod accident has an acceptably low probability. Therefore, if these LCO limits are violated, a short time is allowed for corrective action before a significant power reduction is required.

The maximum allowable setpoints for steady state, transient, and maximum limits for QPT applicable for the full symmetrical incore detector system, minimum incore detector system, and excore detector system are provided; the setpoints are given in the COLR. The setpoints for the three systems are derived by adjustment of the measurement system-independent QPT limits given in the COLR to allow for system observability and instrumentation errors. Figure B 3.2.4-1 on page B 3.2-36 diagrams the action statements required when QPT exceeds the allowable setpoints.

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BASES (continued)

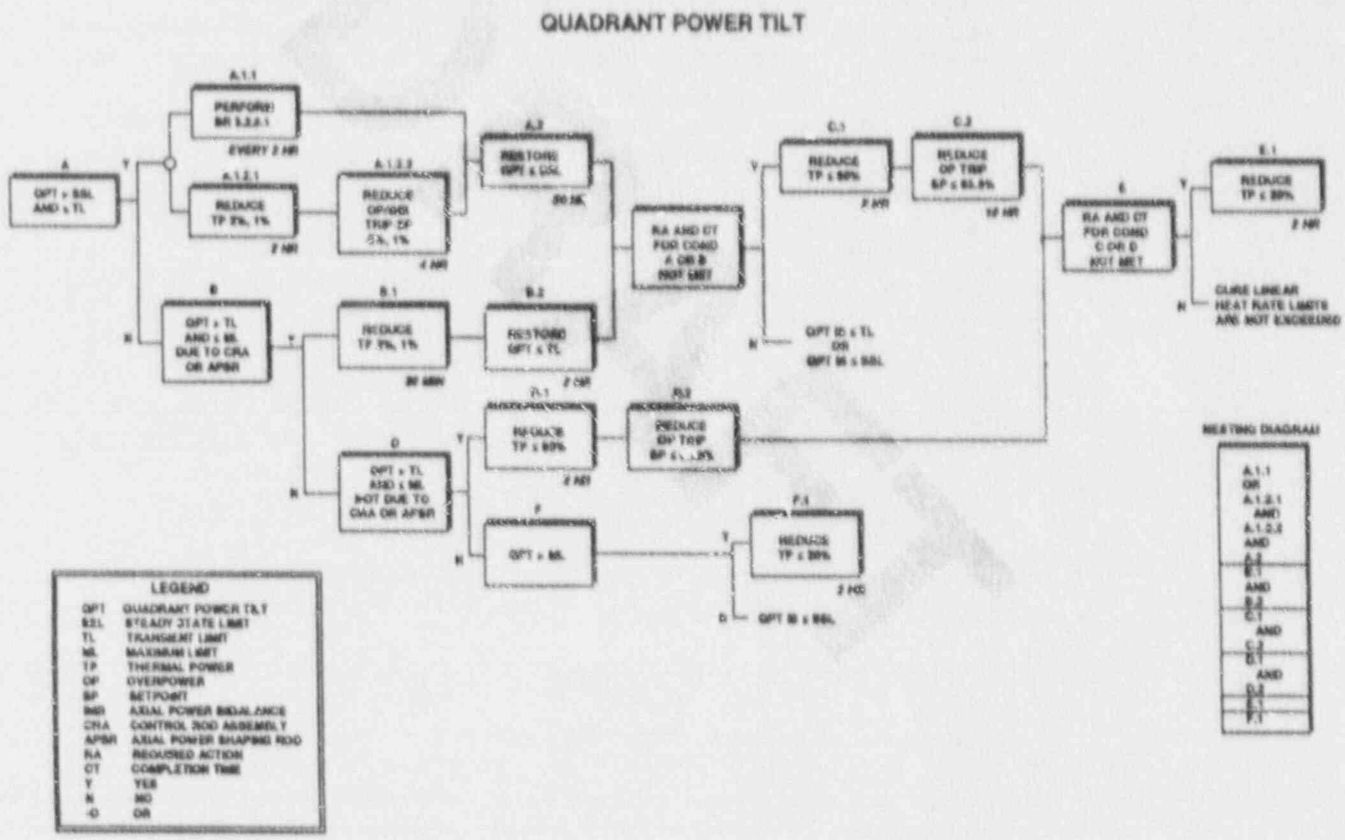

Figure B 3.2.4-1
QUADRANT POWER TILT Action Flow Chart

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BASES (continued)

LCO (continued)

Actual alarm setpoints implemented in the plant may be more restrictive than the maximum allowable setpoint values to allow for additional conservatism between the actual alarm setpoint and the measurement system-independent limit.

It is desirable to retain the ability to operate the reactor when a QPT exists. In certain instances, operation of the reactor with a QPT may be helpful or necessary to discover the cause of the QPT. The combination of power level restriction with QPT in each Required Action statement restricts the local LHR to a safe level. This allows movement through the specified applicability conditions in the exception to Specification 3.0.3.

[For this facility, OPERABLE Incore and Excore Detector Systems constitute the following:]

[For this facility, the following support systems are required to be OPERABLE to ensure OPERABILITY of the Incore and Excore Detector Systems, and to ensure that QPT limits are met:]

[For this facility, the required support systems which, upon their failure, do not result in the Incore or Excore Detector Systems being inoperable, or in the QPT limits not being met, and the justifications are as follows:]

APPLICABILITY

In MODE 1, the limits on QPT must be maintained when THERMAL POWER is greater than 20% of RATED THERMAL POWER (RTP) in order to preclude the core power distribution from exceeding the design limits. The minimum power level of 20% of RTP is large enough to obtain meaningful QPT indications without compromising safety. Operation at or below 20% of RTP with QPT up to 20% is acceptable because the resulting maximum LHR is not high enough to cause violation of the LOCA LHR limit ($F_Q(Z)$ limit) or the initial condition DNB allowable peaking limit ($F^N_{\Delta H}$ limit) during accidents initiated from this power level. In MODE 2, the combination of QPT with maximum ALLOWABLE THERMAL POWER level will not result in

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BASES (continued)

APPLICABILITY (continued)

LHRs sufficiently large to violate the fuel design limits, and therefore, applicability in this mode is not required. Although not specifically addressed, QPTs greater that 20% in MODE 1 with THERMAL POWER less than 20% of RTP are allowed for the same reason.

In MODES 2, 3, 4, 5, and 6, this LCO is not Applicable, since the reactor is not generating THERMAL POWER and QPT is indeterminate.

In MODE 1, it may be necessary to suspend the QPT limits during PHYSICS TESTS per LCO 3.1.8. Suspension of these limits is permissible because the reactor protection criteria are maintained by the remaining LCOs governing the three-dimensional power distribution and by the surveillances required by LCO 3.1.8 and LCO 3.1.9.

ACTIONS

A.1

The steady-state limits specified in the COLR provide an allowance for QPT that may be expected to occur during normal operation. A peaking increase to accommodate QPTs up to the steady-state limit is allowed by the regulating rod insertion limits of LCO 3.2.1 and the AXIAL POWER IMBALANCE limits of LCO 3.2.3.

Operation with a QPT greater than the steady-state limit specified in the COLR potentially violates the LOCA LHR limits ($F_Q(Z)$ limits), or loss-of-flow accident DNB peaking limits ($F^N_{\Delta H}$ limits), or both. To verify that $F_Q(Z)$ and $F^N_{\Delta H}$ are within their specified limits, SR 3.1.5.2 is performed using the Incore Detector System to obtain a three-dimensional power distribution map. Verification that $F_Q(Z)$ and $F^N_{\Delta H}$ are within their limits ensures that operation with QPT in the restricted region does not violate the ECCS or DNB criteria. The required Completion Time of 2 hours is a reasonable amount of time to allow the operator to obtain a power distribution map and to verify the power peaking factors. Repeating SR 3.2.5.1 every 2 hours is reasonable frequency with which to ensure that continued verification of the power peaking factors is obtained as core conditions that influence QPT change.

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BASES (continued)

ACTIONS (continued)

A.2

Indefinite operation with QPT in the restricted region is not prudent. Even if power peaking monitoring per required Action A.1 is continued, excessive QPT may cause radial xenon redistribution to occur. Therefore, power peaking monitoring is allowed for up to 4 hours. This required Completion Time is reasonable based on the low probability of an infrequent or limiting event occurring simultaneously with the limit on QPT out of specification. In addition, it precludes long-term depletion with excessive QPT, and limits the potential for radial xenon redistribution. [For this facility, QPT is restored to within limits by the following actions:]

The 4-hour Completion Time for restoring QPT within acceptable limits gives operator sufficient time to correct the QPT mechanism, including repositioning a dropped or misaligned CONTROL ROD.

A.3.1

The safety analysis has shown that a conservative corrective action is to reduce THERMAL POWER by 2% or more of RTP from the ALLOWABLE THERMAL POWER for each 1% of QPT in excess of the steady-state limit. This ACTION limits the local LHGR to a value corresponding to steady-state operation, thereby reducing it to a value within the assumed accident initial condition limits. The required Completion Time of 2 hours is reasonable based on limiting the potential for xenon redistribution, the low probability of an accident occurring, and the steps required to complete the action.

If QPT can be reduced to less than or equal to the steady-state limit in less than 2 hours, the reactor may return to normal operation without undergoing a power reduction. Significant radial xenon redistribution would not occur within this amount of time.

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ACTIONS (continued)

A.3.2

Power operation is allowed to continue if THERMAL POWER is reduced in accordance with Required Action A.3.1. The same reduction (i.e., 2% or more of RTP) is also applicable to the nuclear overpower trip setpoint and the nuclear overpower based on Reactor Coolant System (RCS) flow and AXIAL POWER IMBALANCE trip setpoint, for each 1% of QPT in excess of the steady-state limit. This reduction maintains both core protection and an OPERABILITY margin at the reduced THERMAL POWER level similar to that at RTP. The required Completion Time of 8 hours is reasonable based on limiting the potential xenon redistribution, the low probability of an accident occurring while operating out of specification, and the number of steps required to complete the action.

A.3.3.1

A.3.3.2.1

A.3.3.2.2

Missing -24 hours to restore QPT with a reduction in power is acceptable [reason].

B.1

If QPT exceeds the transient limit but is equal to or less than the maximum limit due to a misaligned CONTROL ROD or APSR, then power operation is allowed to continue if the THERMAL POWER is reduced 2% or more of RTP from the ALLOWABLE THERMAL POWER for each 1% of QPT in excess of the steady-state limit. Thus, the transient limit is the upper bound within which the 2% for 1% power reduction rule may be applied, but only for QPTs caused by CONTROL ROD or APSR misalignment. The required Completion Time of 30 minutes ensures that the operator will complete the THERMAL POWER reduction before significant xenon redistribution occurs.

B.2

When a misaligned CONTROL ROD or APSR occurs, a local xenon redistribution may occur. The required Completion Time of

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BASES (continued)

ACTIONS (continued)

2 hours allows the operator sufficient time to relatch and/or realign a CONTROL ROD or APSR, but is short enough to limit xenon redistribution so that large increases in the local LHR will not occur due to a xenon redistribution resulting from the power tilt.

C.1

If the QUADRANT POWER TILT cannot be reduced below the required limit within 2 hours (i.e., Required Action A.2 or B.2 not met), a further power reduction is required. Power reduction to < 60% of RTP provides conservative protection from increased peaking due to xenon redistribution. The required Completion Time of 4 hours is reasonable to allow the operator to reduce THERMAL POWER to < 60% of ALLOWABLE THERMAL POWER without challenging plant systems, after the Completion Time for Required Action A.2 or B.2 expires.

C.2

Reduction of the nuclear overpower trip setpoint to 65.5% or less of ALLOWABLE THERMAL POWER after THERMAL POWER has been reduced to < 60% of ALLOWABLE THERMAL POWER maintains both core protection and an OPERABILITY margin at reduced power similar to that at full power. The required Completion Time of 8 hours allows the operator sufficient time to reset the trip setpoint and is reasonable based on operating experience. This Completion Time does not start after that for Required Action C.1 expires.

D.1

Power reduction to 60% of the ALLOWABLE THERMAL POWER is a conservative method of limiting the maximum core LHGR for QPTs up to 20%. Although the power reduction is based on the correlation used in Required Actions A.1 and B.1, the database for a power peaking increase as a function of QPT is less extensive for tilt mechanisms other than misaligned CONTROL RODS and APSRS. Since greater uncertainty in the potential power peaking increase exists with the less extensive database, a more conservative action is taken when the tilt is caused by a mechanism other than a misaligned

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BASES (continued)

ACTIONS (continued)

CONTROL ROD or APSR. The required Completion Time of 2 hours allows the operator to reduce THERMAL POWER to less than 60% of the ALLOWABLE THERMAL POWER without challenging plant systems.

D.2

Reduction of the nuclear overpower trip setpoint to 65.5% or less of the ALLOWABLE THERMAL POWER after THERMAL POWER has been reduced to less than 60% of the ALLOWABLE THERMAL POWER maintains both core protection and an OPERABILITY margin at reduced power similar to that at full power. The required Completion Time of 6 hours allows the operator sufficient time to reset the trip setpoint and is reasonable based on operating experience. The start of this Completion Time is simultaneous with that of Required Action D.1.

E.1

If the Required Actions for Condition C or D cannot be met within the required Completion Time, then the reactor will continue in power operation with a significant QPT. Either the power level has not been reduced to comply with the Required Action or the nuclear overpower trip setpoint has not been reduced within the required Completion Time. To preclude risk of fuel damage in any of these conditions, THERMAL POWER is reduced further. Specification 3.0.3 would normally require a shutdown to MODE 3. However, operation at 20% of RTP would allow the operator to investigate the cause of the QPT and to correct it. Local LHRs with a large QPT will not violate the fuel design limits at or below 20% of RTP. The required Completion Time of 2 hours is acceptable based on limiting the potential increase in local LHRs that could occur due to xenon redistribution with the QPT out of specification.

If the power reduction to $\leq$ 20% of RTP is required due to a delay in resetting the nuclear overpower trip setpoint, then THERMAL POWER may be increased to 60% of RTP after the trip setpoint has been adjusted. [This increase in THERMAL POWER is NOT in LCO! Revise either Bases or LCO.]

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BASES (continued)

ACTIONS (continued)

F.1

The maximum limit of 20% QPT is set as the upper bound within which power reduction to 60% of ALLOWABLE THERMAL POWER or the 2% for 1% power reduction (misaligned CONTROL RODS only) applies [Ref. 4].

The maximum limit of 20% QPT is consistent with allowing power operation up to 60% of ALLOWABLE THERMAL POWER when QPT setpoints are exceeded. QPT that cannot be determined due to support system (i.e., incore or excore detector) inoperability or are in excess of the maximum limit can be an indication of a severe power distribution anomaly, and a power reduction to at most 20% of RTP ensures local LHGRs will not exceed allowable limits while the cause is determined and corrected.

The required Completion Time of 2 hours is acceptable because [ ].

SURVEILLANCE REQUIREMENTS

General

The QPT can be monitored by both the Incore and Excore Detector Systems. The QPT setpoints are derived from their corresponding measurement system-independent limits by adjustment for system observability errors and instrumentation errors. Although they may be based on the same measurement system-independent limit, the setpoints for the different systems are not identical because of differences in the errors applicable for these systems. For the QPT measurements using the Incore Detector System, the Minimum Incore Detector System consists of OPERABLE detectors configured as follows:

a. Two sets of four detectors shall lie in each core half. Each set of detectors shall lie in the same axial plane. The two sets in the same core half may lie in the same axial plane.

b. Detectors in the same plane shall have quarter-core radial symmetry.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

Figure B 3.2.4-2 on page B 3.2-45 depicts an example of this configuration (Minimum Incore System for QPT Measurement). The symmetric incore system for QPT uses the Incore Detector System as described above and is configured such that at least 75% of the detectors in each core quadrant are OPERABLE.

SR 3.2.4.1

Should the plant computer become inoperable, then the Excore System or Minimum Incore Detector System may be used to monitor the QPT. Since these systems do not provide a direct calculation and display of the QPT, performing the calculations at a 12-hour Frequency is sufficient to follow any changes in the QPT that may approach the setpoint because [ ]. This Frequency will also provide the operator with sufficient time to undertake corrective action(s) should the QPT be found to approach the setpoints.

When the full symmetrical Incore Detector System is in use, the operator will receive an alarm, should the QPT increase to the alarm setpoint. When the QPT is less than the alarm setpoint, checking the QPT indication every 7 days ensures that the operator will determine whether the plant computer software and Incore Detector System inputs for monitoring QPT are functioning properly, and that the monitoring and alarm system remains OPERABLE. This allows the QPT mechanisms such as xenon redistribution, burnup gradients, or CONTROL ROD drive mechanism malfunctions that can cause slow development of a QPT, to be detected. Operating experience has confirmed the acceptability of a surveillance Frequency of 7 days.

Following restoration of the QPT to within the steady-state limit, operation at or above 95% of RTP may proceed provided the QPT is determined to remain within the steady-state limit at the increased THERMAL POWER level. In the case where QPT has exceeded the steady-state limit for more than 24 hours or has exceeded the transient limit (Conditions A, B, or D), the potential for xenon redistribution is greater.

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BASES (continued)

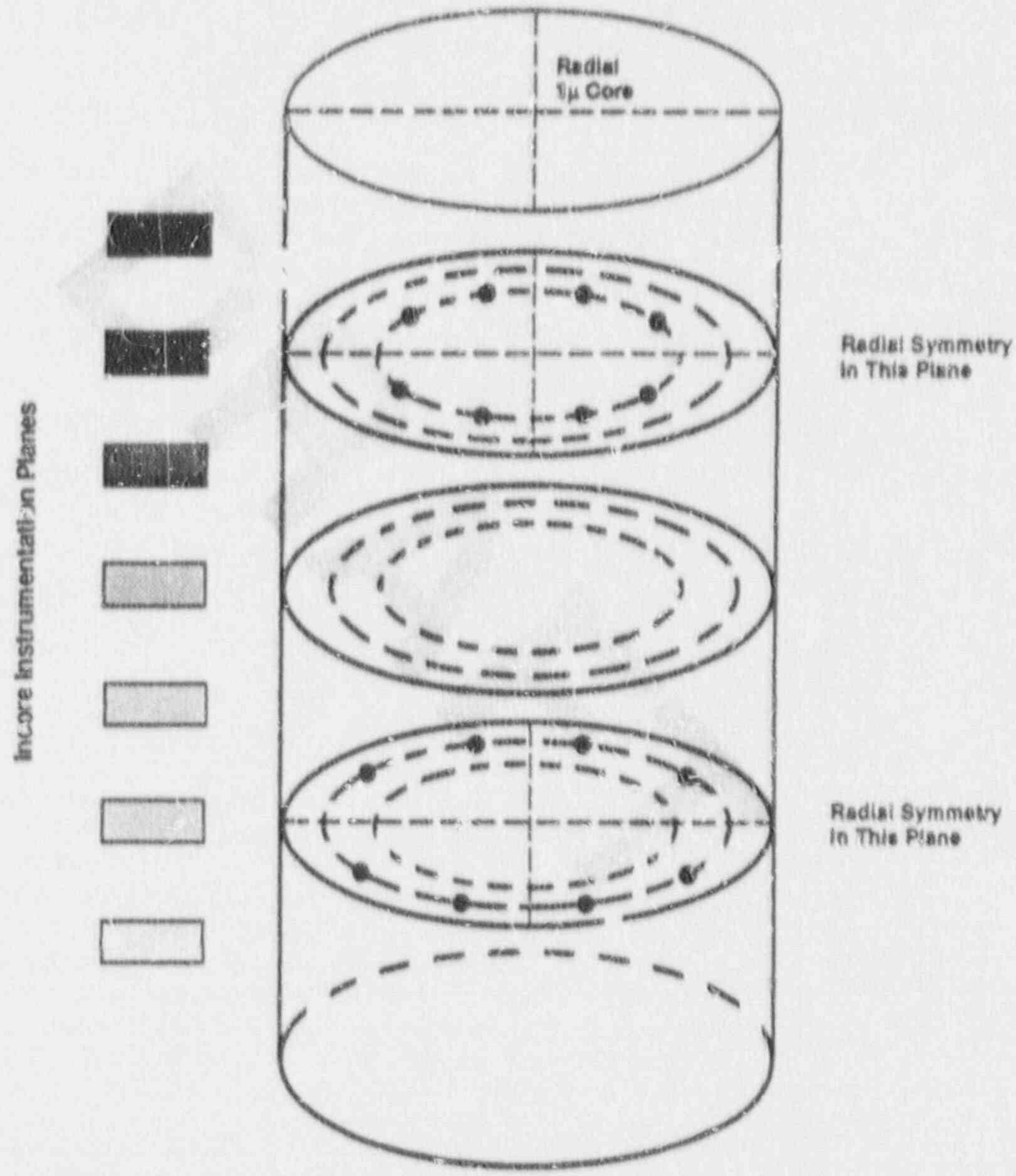

Figure B 3.2.4-2 (Page 1 of 1)
Minimum Incore System for QUADRANT POWER TILT Measurement

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

Therefore, the QPT is monitored for 12 consecutive hourly intervals to determine whether the period of any oscillation due to xenon redistribution will cause the QPT to exceed the steady-state limit again.

REFERENCES

1a. Title 10, Code of Federal Regulations, Part 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors."

1b. Title 10, Code of Federal Regulations, Part 50, Appendix A, GDC 26, "Reactivity Control System Redundancy and Capability."

2. Core Operating Limits Report, [Unit Name].

3. [Unit Name] FSAR,

   a. Section [ ], "Rod Ejection Accident, Accident Bases."

   b. Section [ ], "Rod Ejection Accident, Fuel Rod Damage."

   c. Section [ ], "Thermal and Hydraulic Limits."

4a. Title 10, Code of Federal Regulations, Part 50, Appendix A, GDC 10, "Reactor Design."

4b. Title 10, Code of Federal Regulations, Part 50, Appendix A, GDC 28, "Reactivity Limits."

5. ANSI N18.2-1973, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," American National Standards Institute, August 6, 1973.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.5 Power Peaking Factors

BASES

BACKGROUND

The purpose of this MODE 1 LCO is to establish limits which will constrain the core power distribution within design limits during normal operation (Condition 1) and anticipated operational occurrences (AOOs) (Condition 2) such that accident initial condition protection criteria are preserved. The accident initial condition criteria are preserved by bounding operation at THERMAL POWER within specified acceptable fuel design limits.

The NUCLEAR HEAT FLUX HOT CHANNEL FACTOR, $F_Q(Z)$, is a specified acceptable fuel design limit that preserves the initial conditions for the Emergency Core Cooling System (ECCS) analysis. $F_Q(Z)$ is defined as the maximum local fuel rod linear power density divided by the average fuel rod linear power density, assuming nominal fuel pellet and rod dimensions. Since $F_Q(Z)$ is a ratio of local power densities, it is related to the total local power density in a fuel rod. Operation within the $F_Q(Z)$ limits given in the CORE OPERATING LIMITS REPORT (COLR) (Ref. 1), prevents power peaking that would exceed the loss-of-coolant accident (LOCA) linear heat rate (LHR) limits derived from the analysis of the ECCS.

The NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR ($F^N_{\Delta H}$) limit is a specified acceptable fuel design limit that preserves the initial conditions for the limiting loss-of-flow transient. $F^N_{\Delta H}$ is defined as the ratio of the integral of linear power along the fuel rod on which the minimum Departure from Nucleate Boiling Ratio (DNBR) occurs to the average integrated rod power. Since $F^N_{\Delta H}$ is a ratio of integrated powers, it is related to the radial power density in a fuel rod. Operation within the $F^N_{\Delta H}$ limits given in the COLR prevents Departure from Nucleate Boiling (DNB) during a postulated loss-of-forced-reactor-coolant-flow accident.

Measurement of the core power peaking factors using the Incore Detector System to obtain a three-dimensional power distribution map provides direct confirmation that $F_Q(Z)$ and

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BASES (continued)

BACKGROUND (continued)	$F^N_{\Delta H}$ are within their limits, and may be used to verify that the power-peaking factors remain bounded when one or more of the normal operating parameters exceeds its limit.

APPLICABLE SAFETY ANALYSES

The limits on $F_Q(Z)$ are determined by the ECCS analysis in order to limit peak cladding temperatures to 2200°F during a LOCA. The maximum acceptable cladding temperature is specified by 10 CFR 50.46 (Ref. 2). Higher cladding temperatures could cause severe cladding failure by oxidation due to a Zircaloy-water reaction. $F_Q(Z)$ limits assumed in the LOCA analyses are limiting (i.e., lower) relative to the $F_Q(Z)$ assumed in the safety analyses for other postulated accidents, including loss-of-flow accident (Ref. 3). Therefore, this LCO provides conservative limits for other postulated accidents.

The limits on $F^N_{\Delta H}$ provide protection from DNB during a limiting loss-of-flow transient. Proximity to the DNB condition is expressed by the DNBR, defined as the ratio of the cladding surface heat flux required to cause DNB to the actual cladding surface heat flux. The minimum DNBR value during both normal operation and anticipated transients is limited to the DNBR correlation limit for the particular fuel design in use, and is accepted as an appropriate margin to DNB. The DNB correlation limit ensures there is a 95% probability at a 95% confidence level that the hot fuel rod in the core does not experience DNB.

This LCO precludes 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 (Ref. 2).

b. During a loss-of-forced-reactor-coolant-flow accident, there must be at least a 95% probability, at a 95% confidence level, that the hot fuel rod in the core does not experience a DNB condition.

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

c. During an ejected rod accident, the fission energy input to the fuel must not exceed 280 cal/gm (Ref.4).

d. The CONTROL RODS must be capable of shutting down the reactor with a minimum required SHUTDOWN MARGIN with the highest worth CONTROL ROD stuck fully withdrawn (GDC 26, Ref. 4).

The reload safety evaluation analysis determines limits on global core parameters that characterize the core power distribution. The primary parameters used to monitor and control the core power distribution are the regulating rod position, the AXIAL POWER SHAPING ROD (APSR) position, the AXIAL POWER IMBALANCE, and the QUADRANT POWER TILT (QPT). These parameters are normally used to monitor and control the core power distribution because their measurements are continuously observable. Limits are placed on these parameters to ensure that the core power peaking factors remain bounded during operation in MODE 1. Nuclear design model calculational uncertainty, manufacturing tolerances (e.g., the engineering hot channel factor), effects of fuel densification and rod bow, and modeling simplifications (such as treatment of the spacer grid effects) are accommodated through the use of peaking augmentation factors in the reload safety evaluation analysis.

$F_Q(Z)$ and $F^N_{\Delta H}$ satisfy Criterion 2 of the NRC Interim Policy Statement because they are initial condition inputs to the ECCS analyses and plant safety analyses. The limits for each are contained in the COLR.

LCO

This LCO for the power peaking factors $F_Q(Z)$ and $F^N_{\Delta H}$ ensures that the core operates within the bounds assumed for the ECCS and thermal-hydraulic analyses. Verification that $F_Q(Z)$ and $F^N_{\Delta H}$ are within the limits of this LCO specified in the COLR allows continued operation at THERMAL POWER when the Required Actions of LCO 3.1.4, Control Rod Group Alignment Limits, LCO 3.2.1, Regulating Rod Insertion Limits, LCO 3.2.2, AXIAL POWER SHAPING ROD Insertion Limits, LCO 3.2.3, AXIAL POWER IMBALANCE Operating Limits, and LCO 3.2.4, QUADRANT POWER TILT, are entered. Conservative

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BASES (continued)

LCO (continued)

THERMAL POWER reductions are required if the limits on $F_Q(Z)$ and $F^N_{\Delta H}$ are exceeded.

Measurement uncertainties are applied when $F_Q(Z)$ and $F^N_{\Delta H}$ are determined using the Incore Detector System. The measurement uncertainties applied to the measured values of $F_Q(Z)$ and $F^N_{\Delta H}$ account for uncertainties in observability and instrument string signal processing.

[For this facility, an OPERABLE Incore Detector System constitutes the following:]

[For this facility, the following support systems are required to be OPERABLE to ensure OPERABILITY of the Incore Detector System, and to ensure that power peaking factor limits are met:]

[For this facility, the required support systems which, upon their failure, do not result in the Incore Detector System being declared inoperable, or in the power peaking factor limits not being met, and the justifications are as follows:]

APPLICABILITY

In MODE 1, the limits on $F_Q(Z)$ and $F^N_{\Delta H}$ must be maintained in order to preclude the core power distribution from exceeding the limits assumed in the analyses of the LOCA and loss-of-flow accidents. In MODES 2, 3, 4, 5, and 6, this LCO is not applicable because the reactor has insufficient stored energy in the fuel or energy being transferred to the coolant to require a limit on the distribution of core power. The values of $F_Q(Z)$ and $F^N_{\Delta H}$ in these MODES is, therefore, not important.

The applicability of this LCO is modified by a Note which states that the Completion Times are on a Condition basis.

ACTIONS

General

Care must be taken in interpreting the relationship of the power peaking factors $F_Q(Z)$ and $F^N_{\Delta H}$ to their limits. The limiting values of $F_Q(Z)$ and $F^N_{\Delta H}$ in the COLR may be expressed

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BASES (continued)

ACTIONS (continued)

in either LHR units or in peaking units. Since $F_Q(Z)$ and $F^N_{\Delta H}$ are power peaking factors, a constant LHR is maintained as THERMAL POWER is reduced thereby allowing power peaking to be increased in inverse proportion to THERMAL POWER.

Therefore, the $F_Q(Z)$ and $F^N_{\Delta H}$ limits increase as THERMAL POWER decreases (assuming $F_Q(Z)$ and $F^N_{\Delta H}$ are expressed in peaking units) so that a constant LHR limit is maintained.

A.1

When $F_Q(Z)$ is determined not to be within its specified limit as determined by a three-dimensional power distribution map, a THERMAL POWER reduction is taken to reduce the maximum LHR in the core. Design calculations have verified that a conservative THERMAL POWER reduction is 1% or more of RATED THERMAL POWER (RTP) for each 1% by which $F_Q(Z)$ exceeds its limit (Ref. [ ]). The required Completion Time of 15 minutes is reasonable to take the actions necessary to reduce unit power demand, based on [ ].

A.2

Power operation is allowed to continue by Required Action A.1 if THERMAL POWER is reduced by 1% or more of RTP from the ALLOWABLE THERMAL POWER for each 1% by which $F_Q(Z)$ exceeds its limit. The same reduction in the nuclear overpower trip setpoint and the nuclear overpower based on the Reactor Coolant System (RCS) flow and the AXIAL POWER IMBALANCE trip setpoint for each 1% by which $F_Q(Z)$ is in excess of its limit. These reductions maintain both core protection and an OPERABILITY margin at the reduced THERMAL POWER. The required Completion Time of 8 hours is reasonable based on the low probability of an accident occurring in this relatively short time period (Ref. [ ]) and the number of steps required to complete the action.

A.3

Continued operation with $F_Q(Z)$ exceeding its limit is not permitted, because the initial conditions assumed in the accident analyses are no longer valid. The required Completion Time of 24 hours to restore $F_Q(Z)$ within its limits at the reduced THERMAL POWER level is reasonable based on the low probability of a limiting event occurring

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BASES (continued)

ACTIONS (continued)

simultaneously with $F_Q(Z)$ exceeding its limit (Ref. [ ]). In addition, it precludes long-term depletion with local LHRs higher than the limiting values, and limits the potential for inducing an adverse perturbation in the axial xenon distribution. [For this facility, $F_Q(Z)$ is restored to within its limit by the following actions:]

B.1

When $F^N_{\Delta H}$ is determined not to be within its acceptable limit as determined by a three-dimensional power distribution map, a THERMAL POWER reduction is taken to reduce the maximum LHR in the core. Design calculations have verified that a conservative THERMAL POWER reduction is 3.3% or more of RTP for each 1% by which $F^N_{\Delta H}$ exceeds its limit. The required Completion Time of 15 minutes is reasonable to take the ACTIONS necessary to reduce the unit power.

B.2

Power operation is allowed to continue by Required Action B.1 if THERMAL POWER is reduced by 3.3% or more of RTP for each 1% by which $F^N_{\Delta H}$ exceeds its limit. The same reduction in the nuclear overpower trip setpoint and the nuclear overpower based on RCS flow and the AXIAL POWER IMBALANCE trip setpoint for each 1% by which $F^N_{\Delta H}$ is in excess of its limit, maintains both core protection and an OPERABILITY margin at the reduced THERMAL POWER. The required Completion Time of 8 hours is reasonable based on the low probability of an accident occurring in this relatively short time period and the number of steps required to complete this ACTION.

B.3

Continued operation with $F^N_{\Delta H}$ exceeding its limit is not permitted, because the initial conditions assumed in the accident analyses are no longer valid. The required Completion Time of 24 hours to restore $F^N_{\Delta H}$ within its limit at the reduced THERMAL POWER level is reasonable based on the low probability of a limiting event occurring simultaneously with $F^N_{\Delta H}$ exceeding its limit. In addition, this Completion Time precludes long-term depletion with an

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BASES (continued)

ACTIONS (continued)

unacceptably high local power and limits the potential for inducing an adverse perturbation in the radial xenon distribution. [For this facility, $F^N_{\Delta H}$ is restored to within its limit by the following actions:]

C.1

If a THERMAL POWER reduction is not sufficient to restore $F_Q(Z)$ or $F^N_{\Delta H}$ within its limit (i.e., the Required Actions and associated Completion Times for either Condition A or B not met) or if the power peaking factors ($F_Q(Z)$ and $F^N_{\Delta H}$) cannot be determined (i.e., due to Incore Detector System inoperability), then THERMAL POWER operation should cease. The reactor is placed in MODE 2 where this LCO does not apply. The required Completion Time of 2 hours is a reasonable amount of time for the operator to reduce THERMAL POWER in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.5.1

Since the limits on $F_Q(Z)$ and $F^N_{\Delta H}$ are preserved when the parameters specified by LCO 3.1.4, LCO 3.2.1, LCO 3.2.2, LCO 3.2.3, and LCO 3.2.4 are within their limits, a Note is provided to indicate that monitoring of the power-peaking factors is required only when complying with the Required Actions of these LCOs.

Frequencies for monitoring of the power peaking factors are specified in the Action Statements of the individual LCOs cited above. These Frequencies are reasonable based on the low probability of a limiting event occurring simultaneously with either $F_Q(Z)$ or $F^N_{\Delta H}$ exceeding its limit (Ref. [ ]), and provide sufficient time for the operator to obtain a power distribution map from the Incore Detector System. Indefinite THERMAL POWER operation in a Required Action of LCO 3.1.4, LCO 3.2.1, LCO 3.2.2, LCO 3.2.3, or LCO 3.2.4 is not permitted, in order to limit the potential for exceeding both the power peaking factors assumed in the accident analyses due to operation with unanalyzed core power distributions and spatial xenon distributions beyond their analyzed ranges.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

[Core monitoring is performed using the Incore Detector System to obtain a three-dimensional power distribution map. Maximum values of $F_Q(Z)$ and $F^N_{\Delta H}$ obtained from this map may then be compared to the $F_Q(Z)$ and $F^N_{\Delta H}$ limits in the COLR to verify that the limits have not been exceeded. Measurement of the core power peaking factors in this manner may be used to verify that the measured values of $F_Q(Z)$ and $F^N_{\Delta H}$ remain with their specified limits when one or more of the limits specified by LCO 3.1.4 (Control Rod Group Alignment limits), LCO 3.2.1 (Regulating Rod Insertion Limits), LCO 3.2.2 (AXIAL POWER SHAPING ROD Insertion Limits), LCO 3.2.3 (AXIAL POWER IMBALANCE Operating Limits), or LCO 3.2.4 (QUADRANT POWER TILT) is exceeded. If $F_Q(Z)$ and $F^N_{\Delta H}$ remain within their limits when one or more of these parameters exceeds its limit, operation at THERMAL POWER may continue because the true initial conditions (the power peaking factors) remain within their specified limits.]

REFERENCES

1. [Unit Name] FSAR, Core Operating Limits Report, "[Title]."

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

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

4. Title 10, Code of Federal Regulations, Part 50, Appendix A, GDC 26, "Reactivity Control System Redundancy and Capability."8

B 3.3 INSTRUMENTATION

B 3.3.1 Reactor Protection System (RPS) Instrumentation

BASES

BACKGROUND

The RPS initiates a reactor trip to protect against violating the core fuel design limits and the Reactor Coolant System (RCS) pressure boundary during anticipated operational occurrences (AOOs), and to assist the Engineered Safety Feature (ESF) Systems in mitigating accidents.

The protection and monitoring systems have been designed to assure safe operation of the reactor. This is achieved by specifying limiting safety system settings (LSSS) in terms of parameters directly monitored by the RPS, as well as the 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's life, the acceptable limit is:

1. The departure from nucleate boiling ratio (DNBR) shall be maintained above the Safety Limit (SL) value;
2. Fuel centerline melt shall not occur; and
3. The RCS pressure SL of [2750] psia shall not be exceeded.

Maintaining the SLs within the above values assures that the offsite dose will be within the 10 CFR 50 and 10 CFR 100 criteria during AOOs.

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

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BASES (continued)

BACKGROUND (continued)

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

RPS Overview

The RPS consists of four separate redundant protection channels that receive inputs of neutron flux, RCS pressure, RCS flow, RCS temperature, pressurizer level, RCS pump status, reactor building (RB) pressure, main feedwater, pump status, and turbine status instrumentation channels.

Figure B 3.3.1-1 shows the arrangement of a typical RPS protection channel. A protection channel is composed of measurement channels, a manual trip channel, a reactor trip module (RTM), and CONTROL ROD drive (CRD) trip devices. LCO 3.3.1, "Reactor Protection System (RPS) Instrumentation," provides requirements for the individual measurement channels. These channels encompass all equipment and electronics from the point at which the measured parameter is sensed through the bistable relay contacts in the trip string. LCO 3.3.2, "Reactor Protection System (RPS) Manual Reactor Trip," LCO 3.3.3, "Reactor Protection System (RPS) Reactor Trip Module (RTM)," and LCO 3.3.4, "CONTROL ROD Drive (CRD) Trip Devices," discuss the remaining RPS elements.

The RPS instrumentation measures critical plant parameters and compares these to predetermined setpoints. [For this facility, RPS channels are maintained independent from control system functions as follows:] If the setpoint is exceeded, a channel trip signal is generated. The generation of any two trip signals in any of the four RPS channels will result in the trip of the reactor.

Figure B 3.3.1-2 shows a simplified diagram of the RPS. The RPS contains multiple CRD trip devices, two AC trip breakers, and two DC trip breaker pairs that provide a path for power to the CRD System. Additionally, the power for most of the CRDs passes through silicon controlled rectifier (SCR) relays. The system has two separate paths (or channels), with each path having either two breakers or a

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BASES (continued)

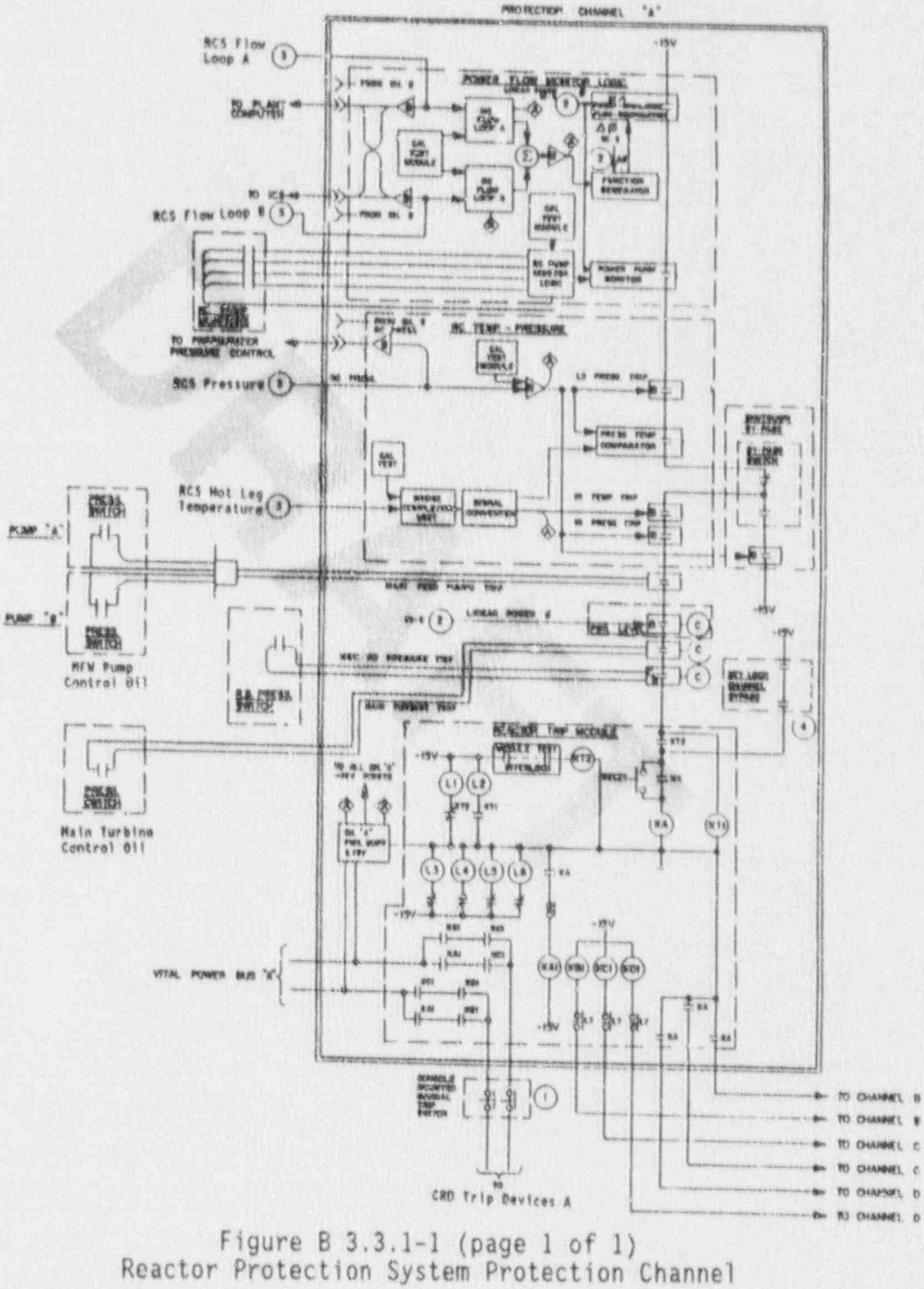

Figure B 3.3.1-1 (page 1 of 1)
Reactor Protection System Protection Channel

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BASES (continued)

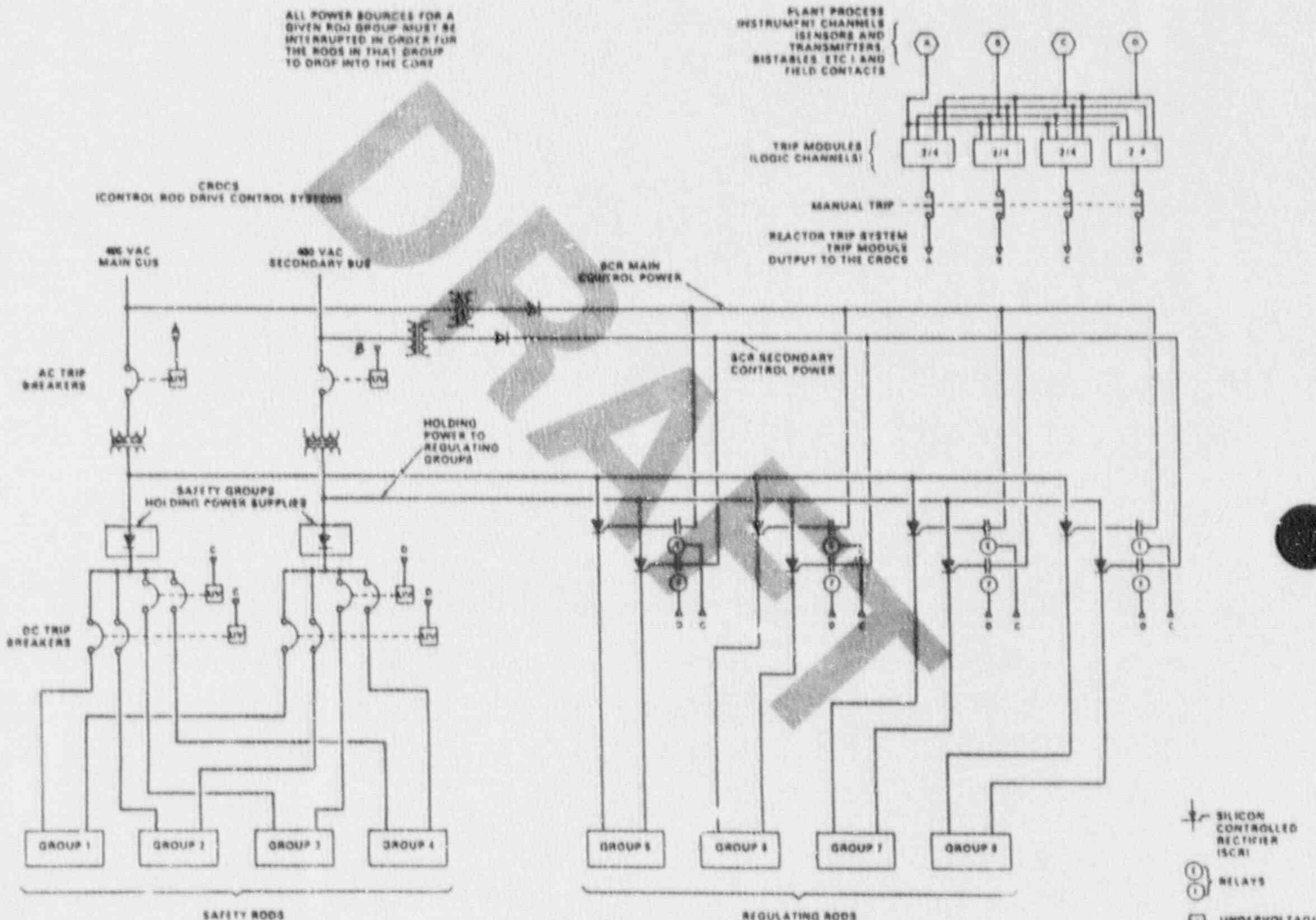

Figure B 3.3.1-2 (page 1 of 1)
Simplified Reactor Protection System Diagram

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BASES (continued)

BACKGROUND (continued)

breaker and an SCR relay in series. Each path provides independent power to the CRDs. Either path can provide sufficient power to operate all CRDs.

To trip the reactor, AC power to the CRDs must be removed. This requires opening a trip device in each path. Loss of power causes the CRD mechanisms to release the CONTROL RODS, which then fall by gravity into the core.

Two separate power paths to the CRDs ensure that a single failure that opens one path will not cause an unwanted reactor trip.

The RPS consists of four independent protection channels, each containing an RTM. The RTM receives signals from its own measurement channels that indicate a protection channel trip is required. The RTM transmits this signal to its own two-out-of-four trip logic and to the two-out-of-four logic of the RTMs in the other three RPS channels. Whenever any two RPS channels transmit channel trip signals, the RTM logic in each channel actuates to remove 120 Vac power from its associated CRD trip breaker.

The reactor is tripped by opening circuit breakers that interrupt the power supply to the CRDs. Six breakers are installed to increase reliability and allow testing of the trip system. A one-out-of-two taken twice logic is used to interrupt power to the rods.

The RPS has two bypasses: a shutdown bypass and a channel bypass. Shutdown bypass allows the withdrawal of safety rods for SHUTDOWN MARGIN availability during plant cooldowns or heatups. Channel bypass is used for maintenance and testing. Test circuits in the trip strings allow complete testing of all RPS trip functions.

All four protection channels are necessary to meet the redundancy and testability of the GDC in 10 CFR 50, Appendix A (Ref. 1). Bypass of a single protection channel is allowed for a short period. [For this facility, the basis for allowing bypass of a single RPS channel is as follows:]

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BASES (continued)

BACKGROUND (continued)

RPS Instrumentation

The RPS operates from the instrumentation channels discussed below. The specific relationship between measurement channels and protection channels differs from parameter to parameter. Three basic configurations are used:

1. Four completely redundant measurements (e.g., reactor coolant flow) with one channel input to each protection channel;

2. Four channels that provide similar, but not identical, measurements (e.g., power range nuclear instrumentation where each RPS channel monitors of a different quantrant), with one channel input to each protection channel; and

3. Redundant measurements with combinational trip logic outside of the protection channels and the combined output provided to each protection channel (e.g., main turbine trip instrumentation).

These arrangements and the relationship of instrumentation channels to trip functions are discussed below to assist in understanding the overall effect of instrumentation channel failure.

A. Power Range Nuclear Instrumentation

Power range nuclear instrumentation channels provide inputs to the following trip functions:

1.a. Nuclear Overpower--High Setpoint;

1.b. Nuclear Overpower--Low Setpoint;

7. Reactor Coolant Pump-to-Power;

8. Nuclear Overpower Reactor Coolant System Flow and Measured Axial Power Imbalance. (Power-Imbalance-Flow);

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BASES (continued)

BACKGROUND (continued)

9. Main Turbine Trip (control oil pressure); and

10. Loss of Main Feedwater Pumps (control oil pressure).

The power range instrumentation has four linear level channels, one for each core quadrant. Each channel feeds one RPS protection channel. Each channel originates in a detector assembly containing two uncompensated ion chambers. The ion chambers are positioned to represent the top half and bottom half of the core. The individual currents from the chambers are fed to individual linear amplifiers. The summation of the top and bottom is the total reactor power. The difference of the top minus the bottom neutron signal is the measured Axial Power Imbalance of the reactor core.

B. Reactor Coolant System Outlet Temperature

The RCS Outlet Temperature provides input to the following functions:

2. RCS High Outlet Temperature; and

5. RCS Variable Low Pressure.

The RCS Outlet Temperature is measured by two resistance elements in each loop, for a total of four. One temperature detector is associated with each protection channel.

C. Reactor Coolant System Pressure

The RCS Pressure provides input to the following functions:

3. RCS High Pressure;

4. RCS Low Pressure;

5. RCS Variable Low Pressure; and

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BASES (continued)

BACKGROUND (continued)

11. Shutdown Bypass RCS High Pressure.

The RPS inputs of reactor coolant pressure are provided by two pressure transmitters in each loop, for a total of four. One sensor is associated with each protection channel.

D. Reactor Building Pressure

The RB Pressure measurements provide input only to the Reactor Building High Pressure trip, Function 6. There are four RB high pressure sensors, one associated with each protection channel.

E. Reactor Coolant Pump Power Monitoring

Reactor coolant pump (RCP) power monitors are inputs to the Reactor Coolant Pump-to-Power trip, Function 7. Each RCP, operating current and voltage is measured by four current transformers and four potential transformers driving four overpower and four underpower relays. Each power monitoring channel consists of an overpower relay and an underpower relay. One channel for each pump is associated with each protection channel.

F. Reactor Coolant System Flow

The RCS Flow measurements are an input to the Nuclear Overpower RCS Flow and Measured Axial Power Imbalance trip, Function 8. The reactor coolant flow inputs to the RPS are provided by eight high-accuracy differential pressure transmitters, four on each loop, which measure flow through calibrated flow tubes. One flow input in each loop is associated with each protection channel.

G. Main Turbine Automatic Stop Oil Pressure

Main Turbine Automatic Stop Oil Pressure is an input to the Main Turbine Trip reactor trip, Function 9. Each of the four protection channels receives turbine status information from the same four pressure switches monitoring main turbine automatic stop oil

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BASES (continued)

BACKGROUND (continued)

pressure. An open indication will be provided to the RPS on a turbine trip. Contact buffers in each protection channel continuously monitor the status of the contact inputs and initiate an RPS trip when a turbine trip is indicated.

H. Feedwater Pump Control Oil Pressure

Feedwater Pump Control Oil Pressure is an input to the Loss of Main Feedwater Pumps trip, Function 10. Control oil pressure is measured by four switches on each feedwater pump. One switch on each pump is associated with each protection channel.

RPS Bypasses

The RPS is designed with two types of bypasses: channel bypass and shutdown bypass.

Channel bypass provides a method of placing all functions in one RPS protection channel in a bypassed condition, and shutdown bypass provides a method of leaving the safety rods withdrawn during cooldown and depressurization of the RCS. Each bypass is discussed below.

Channel Bypass

A channel bypass provision is provided to allow for maintenance and testing of the RPS. The use of Channel Bypass keeps the protection channel trip relay energized regardless of the status of the instrumentation channel of the bistable relay contacts. To place a protection channel in channel bypass, the other three channels must not be in channel bypass. This is ensured by contacts from the other channels being in series with the channel bypass relay. If any contact is open, then the second channel cannot be bypassed. The second condition is the closing of the key switch. When the bypass relay is energized, the bypass contact closes, maintaining the channel trip relay in an energized condition. All RPS trips are reduced to a two-out-of-three logic in channel bypass.

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BASES (continued)

BACKGROUND (continued)

Shutdown Bypass

During plant cooldown, it is desirable to leave the safety rods withdrawn to provide shutdown capabilities in the event of unusual positive reactivity additions (moderator dilution, etc.).

However, the plant is also depressurized as coolant temperature is decreased. If the safety rods are withdrawn and coolant pressure is decreased, an RCS Low Pressure trip will occur at [1800] psig and the rods will fall into the core. To avoid this, the protection system allows the operator to bypass the low pressure trip and maintain shutdown capabilities. During the cooldown and depressurization, the safety rods are inserted prior to the low pressure trip [1800] psig. The RCS pressure is decreased to less than [1720] psig, then each RPS channel is placed in shutdown bypass.

In shutdown bypass, a normally closed contact opens and the operator closes the shutdown bypass key switch. This action bypasses the RCS low pressure trip, Nuclear Overpower RCS Flow and measured Axial Power Imbalance trip, Reactor Coolant Pump-to-power trip, and the RCS Variable Low Pressure trip and inserts a new High Pressure RCS, [1850] psig trip. The operator can now withdraw the safety rods for additional SHUTDOWN MARGIN.

The insertion of the new high pressure trip performs two functions. First, with a trip setpoint of [1850] psig, the bistable prevents operation at normal system pressure, [ ] psig, with a portion of the RPS bypassed. The second function is to ensure that the bypass is removed prior to normal operation. When the RCS pressure is increased during a plant heatup, the safety rods are inserted prior to reaching [ ] psig. The shutdown bypass is removed, which returns the RPS to normal, and system pressure is increased to greater than [ ] psig. The safety rods are then withdrawn and remain at the full out condition for the rest of the heatup.

In addition to the Shutdown Bypass RCS High Pressure trip, the high flux trip setpoint is administratively reduced to 5% RATED THERMAL POWER (RTP) while the RPS is in shutdown bypass. This provides a backup to the Shutdown Bypass

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BASES (continued)

BACKGROUND (continued)

RCS High Pressure trip, and allows low temperature physics testing while preventing the generation of any significant amount of power.

Module Interlock and Test Trip Relay

Each channel and each trip module is capable of being individually tested. When a module is placed into the test mode, it causes the test trip relay to open and to indicate an RPS channel trip. Under normal conditions, the channel to be tested is placed in bypass before a module is tested.

Trip Setpoints/ALLOWABLE VALUE

The trip setpoints are the normal values at which the bistables are set. Any bistable is considered to be properly adjusted when the "as left" value is within the band for CHANNEL CALIBRATION accuracy, i.e., (± rack calibration + comparator setting accuracy).

The trip setpoints used in the bistables are based on the analytical limits stated in Reference 2. 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, instrument drift, and severe environment errors for those RPS channels that must function in harsh environments as defined by 10 CFR 50.49 (Ref. 3), the ALLOWABLE VALUES specified in Table 3.3.1-1 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 actual, nominal trip setpoint entered into the bistable is more conservative than that specified by the ALLOWABLE VALUE, to account for changes in random measurement errors detectable by a CHANNEL FUNCTIONAL TEST. One example of such a change in measurement error is drift during the Surveillance Frequency. If the measured setpoint does not exceed the ALLOWABLE VALUE, the bistable is considered OPERABLE.

Setpoints in accordance with the ALLOWABLE VALUE ensure that the limits of Specification 2.0, "Safety Limits" are not violated during AOOs, and the consequences of DBAs will be

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BASES (continued)

BACKGROUND (continued)

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 LCO 3.3.1, "Reactor Protection System (RPS) Instrumentation," the ALLOWABLE VALUES of Table 3.3.1-1 are the LSSS. This ALLOWABLE VALUE is established to prevent violation of the SLs during normal plant operation and AOOs.

Each channel can be tested online to verify that the signal and setpoint accuracy is within the specified allowance requirements of Reference 5. Once a designated channel is taken out of service for testing, a simulated signal is injected in place of the field instrument signal. The process equipment for the channel in test is then tested, verified, and calibrated. Surveillances for the channels are specified in the SR section.

The ALLOWABLE VALUES listed in Table 3.3.1-1 are based on the methodology described in Reference 4, which incorporates all of the known uncertainties applicable for each channel. The magnitudes of these uncertainties are factored into the determination of each trip setpoint. All field sensors and signal processing equipment for these channels are assumed to operate within the allowances of these uncertainty magnitudes.

APPLICABLE SAFETY ANALYSES

Each 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. Functions not specifically credited in the accident analysis were qualitatively credited in the safety analysis and the Nuclear Regulatory Commission (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. These functions also serve as backups to functions that were credited in the safety analysis.

The safety analyses applicable to each RPS function are discussed below.

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

1.a. Nuclear Overpower--High Setpoint

The Nuclear Overpower--High trip provides protection for the design thermal overpower condition based on the measured out-of-core fast neutron leakage flux. The trip initiates a reactor trip when the neutron power reaches a pre-defined setpoint at the design overpower limit. Because thermal power lags the neutron power, tripping when the neutron power reaches the design overpower will limit thermal power to a maximum value of the design overpower. Thus, the Nuclear Overpower--High Trip protects against violation of the DNBR and centerline fuel melt SLs. However, the RCS Variable Low Pressure and Nuclear Overpower RCS Flow and measured Axial Power Imbalance provide more direct protection. The role of the Nuclear Overpower--High trip is to limit reactor thermal power below the highest power at which the other two trips are known to provide protection.

The Nuclear Overpower--High trip also provides transient protection, for positive reactivity excursions during power operations. These events include the rod withdrawal accident, the rod ejection accident (REA), and the steam line break (SLB) accident. By providing a trip during these events, the Nuclear Overpower--High trip protects the plant from excessive power levels and also serves to reduce reactor power to prevent violation of the RCS pressure SL.

Rod withdrawal accident analyses cover a large spectrum of reactivity insertion rates (rod worths) which exhibit slow and rapid rates of power increases. At high reactivity insertion rates, the Nuclear Overpower-High trip provides the primary protection. At low reactivity insertion rates, the high pressure trip provides primary protection.

1.b. Nuclear Overpower--Low Setpoint

While in shutdown bypass, with the Shutdown Bypass RCS High Pressure trip OPERABLE, the Nuclear Overpower--Low Setpoint trip must be reduced to < 5% RTP.

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

The low-power setpoint, in conjunction with the lower Shutdown Bypass RCS High Pressure setpoint, ensure that plant is protected from excessive power conditions when other RPS trips are bypassed.

2. RCS High Outlet Temperature

The RCS High Outlet Temperature trip, in conjunction with the RCS Low Pressure and RCS Variable Low Pressure trips, provide protection for the DNBR SL. A trip is initiated whenever the reactor vessel outlet temperature approaches the conditions necessary for departure from nucleate boiling (DNB). Each RCS High Outlet Temperature channel encompasses the modules from the resistance temperature detectors (RTDs) through the bistable contact. Portions of each RCS High Outlet Temperature trip channel are common with the RCS Variable High Pressure trip. The RCS High Outlet Temperature trip provides steady-state protection for the DNBR SL.

The RCS High Outlet Temperature trip limits the maximum RCS temperature to below the highest value for which DNB protection by the Variable Low Pressure trip is ensured. Above the high temperature trip, the variable low pressure trip need not provide protection, because the plant would have tripped already.

3. RCS High Pressure

The RCS High Pressure trip works in conjunction with the pressurizer and main steam safety valves (MSSVs) to prevent RCS overpressurization, thereby protecting the RCS high pressure SL.

The RCS High Pressure trip has been credited in the accident analysis calculations for rod withdrawal accidents, moderator dilution accidents, and loss of feedwater accidents. The rod withdrawal accidents cover a large spectrum of reactivity insertion rates and rod worths that exhibit slow and rapid rates of power increases. At high reactivity insertion rates, the Nuclear Overpower--High trip provides the primary

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

protection. At low reactivity insertion rates, the RCS High Pressure trip provides the primary protection.

4. RCS Low Pressure

The RCS Low Pressure trip, in conjunction with the RCS High Outlet Temperature and Variable Low Pressure trips, provides protection for the DNBR SL. A trip is initiated whenever the system pressure approaches the conditions necessary for DNB. The RCS Low Pressure trip provides DNB low pressure limit for the RCS Variable Low Pressure trip.

The RCS Low Pressure trip also provides transient protection for primary system depressurization events. The RCS Low Pressure trip was also credited in the safety analyses during small break loss-of-coolant accidents (LOCAs).

5. RCS Variable Low Pressure

The RCS Variable Low Pressure trip, in conjunction with the RCS High Outlet Temperature and Low Pressure trips provide protection for the DNBR SL. A trip is initiated whenever the system parameters of pressure and temperature approach the conditions necessary for DNB. The RCS Variable Low Pressure trip provides a floating low pressure trip based on the RCS High Outlet Temperature within the range specified by the High Temperature and Low Pressure trips.

6. Reactor Building High Pressure

The RB High Pressure trip provides an early indication of a high energy line break (HELB) inside the RB. By detecting changes in the RB pressure, the RPS can provide a reactor trip before the other system parameters have varied significantly. Thus, this trip acts to minimize accident consequences. It also provides a backup for RPS trip instruments exposed to an RB HELB environment.

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

7. Reactor Coolant Pump-to-Power

The Reactor Coolant Pump-to-Power trip provides protection for changes in the reactor coolant flow due to the loss of multiple RCPs. The trip senses a loss of an RCP by measuring the power supplied to the pump. Because the flow reduction lags loss of power indications due to the inertia of the RCPs, the trip initiates protective action earlier than a trip based on a measured flow signal.

The trip also prevents operation with both pumps in either coolant loop tripped. Under these conditions, core flow and core fluid mixing are insufficient for adequate heat transfer. Thus, the Reactor Coolant Pump-to-Power trip functions to protect the DNBR and fuel centerline melt SLs.

The Reactor Coolant Pump-to-Power trip has been credited in the accident analysis calculations for the loss of four RCPs. The trip also provides the primary protection for the loss of a pump or pumps, which would result in both pumps in a single SG loop being tripped.

8. Nuclear Overpower RCS Flow and Measured Axial Power Imbalance

The Nuclear Overpower RCS Flow and Measured Axial Power Imbalance trip provides steady-state protection for the power-imbalance SLs. A reactor trip is initiated when the core power, axial power peaking and reactor coolant flow conditions indicate an approach to DNB or fuel centerline melt limits. By measuring reactor coolant flow, and tripping only when conditions approach an SL, the plant can operate with the loss of one pump from a four pump initial condition.

This trip also provides transient protection, through the power-to-flow ratio, for loss of reactor coolant flow events. The power-to-flow ratio provides input for an acceptable protection from DNB for the loss of a single RCP and for locked RCP rotor accidents.

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

The imbalance portion of the trip is used for steady-state protection only.

The power-to-flow ratio of the Nuclear Overpower RCS Flow and Measured Axial Power Imbalance trip also provides steady-state protection to prevent reactor power from exceeding the allowable power for flow rates less than full four-pump flow. Thus, the power-to-flow ratio prevents overpower conditions similar to the Nuclear Overpower trip. This protection ensures that during reduced flow conditions, the core power is maintained below that required to begin DNB.

9. Main Turbine Trip (Control Oil Pressure)

The Main Turbine Trip function trips the reactor when the main turbine is lost at high power levels. The Main Turbine Trip function provides an early reactor trip in anticipation of the loss of heat sink associated with a turbine trip. The Main Turbine trip Function was added to the B&W designed plants in accordance with NUREG-0737 (Ref. 5) following the Three Mile Island Unit 2 accident. The trip lowers the probability of an RCS power operated relief valve (PORV) actuation for turbine trip cases. This trip is activated at higher power levels, thereby limiting the range through which the Integrated Control System must provide an automatic runback on a turbine trip.

10. Loss of Main Feedwater Pumps (Control Oil Pressure)

The Loss of Main Feedwater Pump 6 trip provides a reactor trip at high power levels when both main feedwater pumps are lost. The trip provides an early reactor trip in anticipation of the loss of heat sink associated with the loss of main feedwater. This trip was added in accordance with NUREG-0737 (Ref. 5) following the Three Mile Island Unit 2 accident. This trip provides a reactor trip at high power levels for a loss of main feedwater to minimize challenges to the PORVs.

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

11. Shutdown Bypass RCS High Pressure

The RPS Shutdown Bypass RCS High Pressure is provided to allow for withdrawing the CONTROL RODS prior to reaching the normal RCS Low Pressure trip setpoint. The shutdown bypass provides trip protection during deboration and RCS heatup by allowing the operator to withdraw the safety groups of CONTROL RODS. This makes their negative reactivity available to terminate inadvertent reactivity excursions. Use of the shutdown bypass trip requires that the neutron power trip setpoint be reduced to 5% of full power or less. This forces a trip to occur whenever the plant switches from power operation to shutdown bypass or vice versa. This ensures that the CONTROL RODS are all inserted and the flux distribution is known before power operation can begin. The operator is required to remove the shutdown bypass, reset the Nuclear Overpower--High Power trip setpoint, and again withdraw the safety rod groups before proceeding with startup.

Accidents analyzed in the FSAR (Ref. 2) do not describe events that occur during shutdown bypass operation, because the consequences of these events are enveloped by the events presented in the FSAR.

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

LCO

The LCO requires all instrumentation performing an RPS function to be OPERABLE. Failure of any instrument 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.

The four channels of each function of the RPS shall be OPERABLE at all times the reactor is critical to ensure that a reactor trip will be actuated if needed. The trip

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BASES (continued)

LCO (continued)

function channels specified in Table 3.3.1-1 are considered OPERABLE when the applicable criteria below are met:

1. All channel components necessary to provide a reactor trip are functional and in service;
2. Channel measurement uncertainties and trip setpoints are known via test, analysis, or design information within the assumptions of the setpoint analysis;
3. Required surveillance testing is current and has demonstrated performance within each surveillance test's acceptance criteria; and
4. Associated operational bypasses are not enabled except under the conditions specified by the LCO applicability statement for the function.

The following Bases for each trip function identify which of the above items are applicable to the established OPERABILITY.

Required Actions allow maintenance (protection channel) bypass of individual channels, but the bypass activates interlocks that prevent operation with a second channel in the same function bypassed. Operation in this condition is restricted to 48 hours before restoring the function to four channel operation (two-out-of-four logic), or placing the channel in trip (one-out-of-three logic). Bypass effectively places the plant in a two-out-of-three logic configuration.

Only the ALLOWABLE VALUES are specified for each RPS trip function in the LCO. Nominal trip setpoints are specified in the plant-specific setpoint calculations. The nominal setpoints are selected to ensure that the setpoint measured by CHANNEL FUNCTIONAL TESTS does not exceed the ALLOWABLE VALUE if the bistable 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 is consistent with the assumptions of the plant-specific setpoint calculations. Each ALLOWABLE VALUE specified is more conservative than the analytical limit assumed in the safety analyses to account

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BASES (continued)

LCO (continued)

for instrument uncertainties appropriate to the trip function. These uncertainties are defined in the plant-specific setpoint analysis (Ref. 4).

For most RPS functions, the trip setpoint ALLOWABLE VALUE is to ensure that the DNB or RCS pressure SLs are not challenged. Figure B 3.3.1-3 illustrates the relationship between these RPS trips and the SLs. The Nuclear Overpower RCS flow and AXIAL POWER IMBALANCE trip setpoint ALLOWABLE VALUE are selected to protect the AXIAL POWER IMBALANCE SLs as illustrated by Figure B 3.3.1-4. The two figures are illustrative only. Cycle specific figures for use during operation are contained in the CORE OPERATING LIMITS REPORTS (COLR). Certain RPS trips function to indirectly protect the SLs by detecting specific conditions that do not immediately challenge SLs but will eventually lead to challenge if no action is taken. These trips function to minimize the plant transients caused by the specific conditions. The ALLOWABLE VALUE for these functions is selected at the minimum deviation from normal values that will indicate the condition, without risking spurious trips due to normal fluctuations in the measured parameter.

Bases for the individual function requirements are as follows:

1.a. <u>Nuclear Overpower--High Setpoint</u>

The Nuclear Overpower--High Setpoint trip provides protection for overpower conditions based on the measured, out-of-core, fast neutron leakage flux. Each channel instrumentation encompasses the module from the uncompensated ion chambers through the bistable contacts. Portions of each channel are common with the Reactor Coolant Pump-to-Power, Nuclear Overpower RCS Flow and Measured Axial Power Imbalance, Turbine Trip, and Loss of Main Feedwater Pumps trips. Nuclear Overpower--High Setpoint trip channels are OPERABLE when OPERABILITY criteria 1 through 3 above are met.

The ALLOWABLE VALUE specified is selected to ensure that a trip occurs before reactor power exceeds the highest point at which the Variable Low Pressure and the Nuclear Overpower RCS Flow and Measured Axial

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BASES (continued)

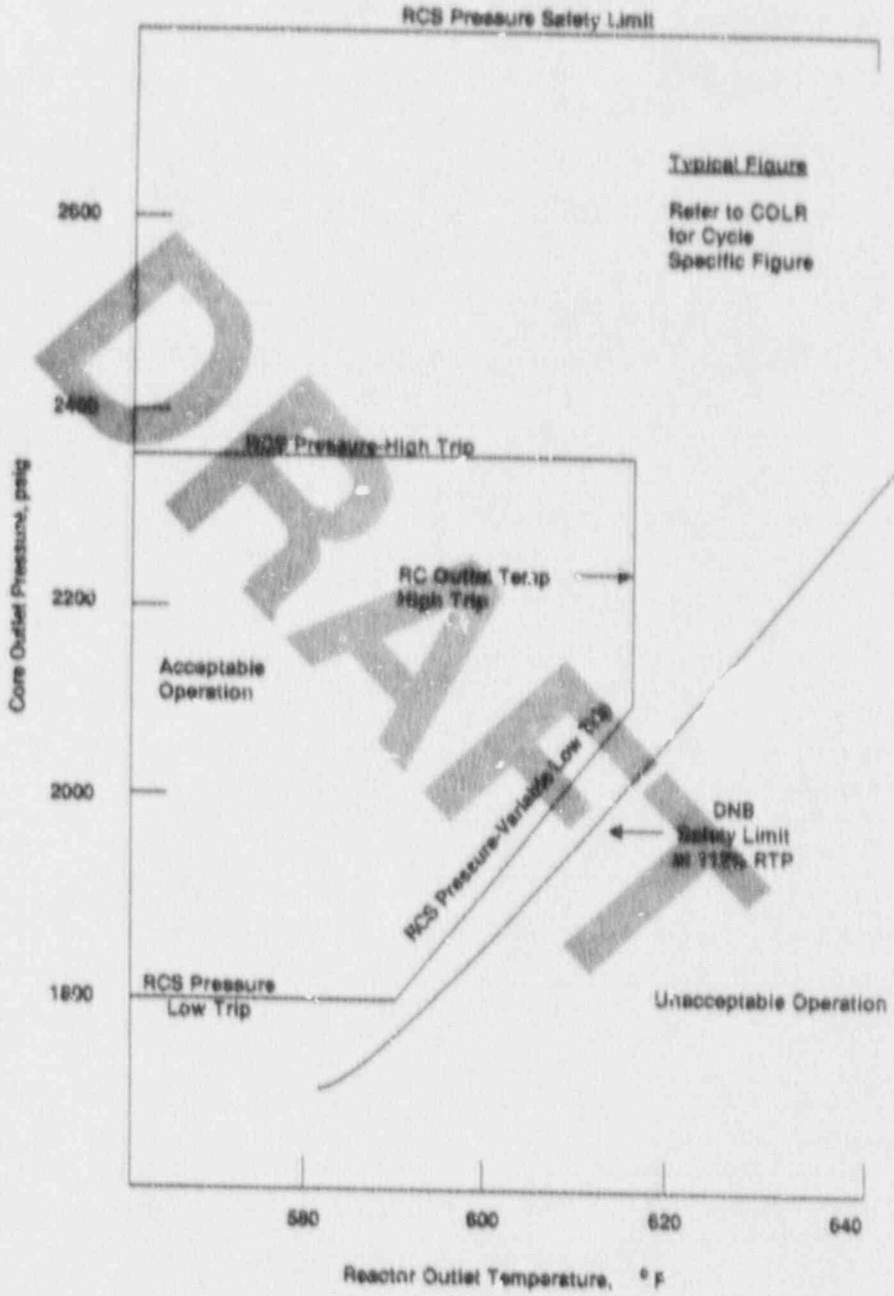

Figure B 3.3.1-3 (page 1 of 1)
Relationship of Reactor Protection System Trips to Departure from Nucleate Boiling and Reactor Coolant System Pressure Safety Limits

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BASES (continued)

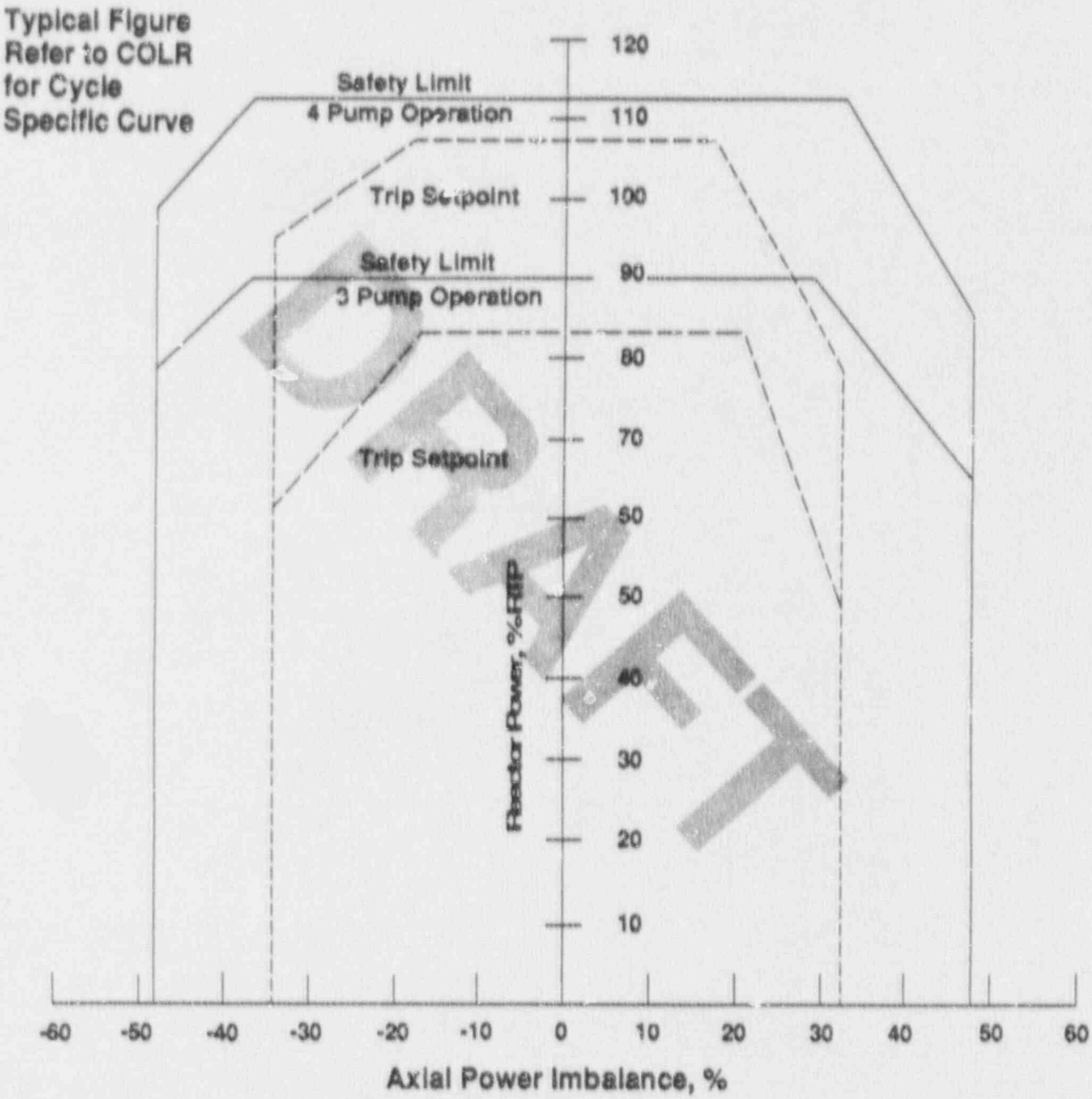

Figure B 3.3.1-4 (page 1 of 1)
Relationship of Nuclear Overpower Reactor Coolant System Flow and Measured Axial Power Imbalance Trip to AXIAL POWER IMBALANCE Safety Limit

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BASES (continued)

LCO (continued)

Power Imbalance trips are analyzed to provide protection against DNB and fuel centerline melt. The ALLOWABLE VALUE is also selected to ensure a trip occurs before an overpower condition causes RCS pressure to exceed the RCS pressure SL. The function's setpoint ALLOWABLE VALUE does not account for harsh environment, induced errors, because the trip will actuate prior to degraded environmental conditions being reached.

1.b. <u>Nuclear Overpower--Low Setpoint</u>

While in shutdown bypass with the Shutdown Bypass RCS High Pressure trip OPERABLE, the Nuclear Overpower trip setpoint must be reduced to less than 5% RTP. The Nuclear Overpower--Low Setpoint, in conjunction with the lower Shutdown Bypass RCS High Pressure setpoint, ensure that the plant is protected from excessive power conditions when other RPS trips are bypassed. The setpoint ALLOWABLE VALUE was chosen to be as low as practical and still lie within the range of the out-of-core instrumentation. Nuclear Overpower--Low setpoint trip channels are OPERABLE when OPERABILITY criteria 1 through 3 above are met.

2. <u>RCS High Outlet Temperature</u>

The RCS High Outlet Temperature trip, in conjunction with the Low Pressure and Variable Low Pressure trips, provides protection for the DNBR SL. A trip is initiated whenever the reactor vessel outlet temperature approaches the conditions necessary for DNB. Portions of each RCS High Outlet Temperature trip channel are common with the RCS Variable Low Pressure trip. Each RCS High Outlet Temperature trip channel encompasses the modules from the resistance temperature detectors (RTDs) through the bistable contact and the reactor trip modules. RCS High Outlet Temperature trip channels are OPERABLE when OPERABILITY criteria 1 through 3 above are met.

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BASES (continued)

LCO (continued)

The trip setpoint ALLOWABLE VALUE is selected to ensure that a trip occurs before hot leg temperatures reach the point beyond which the RCS Low Pressure and Variable Low Pressure trips are analyzed. The setpoint ALLOWABLE VALUE does not reflect errors induced by harsh environmental conditions the equipment is expected to experience because the trip is not required to mitigate accidents that create harsh conditions in the RB.

3. RCS High Pressure

The RCS High Pressure trip setpoint provides protection for the RCS high pressure SL. The high pressure trip works in conjunction with the pressurizer safety valves and main steam safety valves to prevent RCS overpressurization. Each RCS High Pressure trip channel encompasses equipment from the RCS pressure taps through the bistable contacts. Portions of each High Pressure trip channel are common with the RCS Low Pressure and Variable Low Pressure trips. RCS High Pressure trip channels are OPERABLE when OPERABILITY criteria 1 through 3 above are met.

The setpoint ALLOWABLE VALUE is selected to ensure that the RCS high pressure SL is not challenged during steady-state operation or slow power transients. The setpoint ALLOWABLE VALUE does not reflect errors induced by harsh environmental conditions, because the equipment is not required to mitigate accidents that create harsh conditions in the RB.

4. RCS Low Pressure

The RCS Low Pressure trip, in conjunction with the RC High Outlet Temperature and Variable Low Pressure trips, provide protection for the DNBR SL. Each low pressure instrumentation channel encompasses the modules from the pressure taps through the bistable contacts. Portions of the channels are common with the RCS High Pressure and Variable Low Pressure trips.

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BASES (continued)

LCO (continued)

The RCS Low Pressure trip channels are OPERABLE when OPERABILITY criteria 1 through 4 are met.

The RCS Low Pressure setpoint ALLOWABLE VALUE is selected to ensure that a reactor trip occurs before RCS pressure reduced below the lowest point at which the Variable Low Pressure trip is analyzed. The RCS Low Pressure trip has been credited in the accident analysis calculations for small break LOCAs. Consequently, harsh RB conditions created by small break LOCAs can affect performance of the RCS pressure sensors and transmitters. Therefore, degraded environmental conditions are considered in the ALLOWABLE VALUE determination.

5. <u>RCS Variable Low Pressure</u>

The RCS Variable Low Pressure trip, in conjunction with the RCS High Temperature and Low Pressure trips, provide protection for the DNBR SL. The RCS Variable Low Pressure trip provides a floating low pressure trip based on the RCS High Outlet Temperature within the range specified by the high temperature and low RCS pressure trips. Each RCS Variable Low Pressure trip channel encompasses equipment from the RCS pressure taps and the outlet temperature RTDs to the bistable contacts. Portions of these channels are common with the RCS High and Low Pressure functions and the RCS High Outlet Temperature function. RCS Variable Low Pressure trip channels are OPERABLE when OPERABILITY criteria 1 through 4 above are met.

The RCS Variable Low Pressure setpoint ALLOWABLE VALUE is selected to ensure that a trip occurs when temperature and pressure approach the conditions necessary for DNB while operating in a temperature-pressure region constrained by the low pressure and high temperature trips. The RCS Variable Low Pressure trip is not assumed for transient protection in the plant safety analysis; therefore, determination of the setpoint ALLOWABLE VALUE does not need to account for errors induced by a harsh RB environment.

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BASES (continued)

LCO (continued)

6. Reactor Building High Pressure

The RB High Pressure trip provides an early indication of a HELB inside the RB. Each trip channel encompasses the modules from RB pressure taps through the bistable contacts. RB High Pressure trip channels are considered OPERABLE when OPERABILITY criteria 1 through 3 above are met.

The ALLOWABLE VALUE for RB High Pressure trip is set at the lowest value consistent with avoiding spurious trips during normal operation. The electronic components of the RB High Pressure trip [are located in an area that is not exposed to high temperature steam environments during LOCA/HELB transients]. The components are exposed to high radiation conditions. Therefore, the determination of setpoint ALLOWABLE VALUE accounts for errors induced by the high radiation.

7. Reactor Coolant Pump-to-Power

The Reactor Coolant Pump-to-Power trip provides protection for changes in the RCS flow due to the loss of multiple RCS pumps. The trip also prevents operation during conditions in which core flow and core fluid mixing are insufficient for adequate heat transfer.

Each of the function's instrumentation channels encompasses equipment from the uncompensated ion chambers and the individual pump status indicators to the channel bistable contact. Portions of each channel are common with the Nuclear Overpower, Nuclear Overpower RCS Flow and Measured Axial Power Imbalance, Turbine Trip, and Loss of Main Feedwater Pump trips. Reactor Coolant Pump-to-Power trip channels are considered OPERABLE when OPERABILITY criteria 1 through 4 above are met.

The ALLOWABLE VALUE for the Reactor Coolant Pump-to-Power trip setpoint is selected to prevent normal power operation unless at least three RCPs are operating. RCP status is monitored by power relays on each pump. These relays trip on overpower with

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BASES (continued)

LCO (continued)

an ALLOWABLE VALUE of ≥ [14,400] kW and on underpower with an ALLOWABLE VALUE of ≤ [1,752] kW. The overpower ALLOWABLE VALUE is selected low enough to detect locked rotor conditions, but high enough to avoid a spurious trip on the in-rush current when the pumps start. The underpower ALLOWABLE VALUE is selected to reliably trip on loss of voltage to the RCPs. Neither the reactor power nor the pump power ALLOWABLE VALUE account for instrumentation errors caused by harsh environments because the trip function is not required to respond to events that could create harsh environments in the RB or any other plant location.

8. Nuclear Overpower RCS Flow and Measured Axial Power Imbalance

The Nuclear Overpower RCS Flow and Axial Power Imbalance trip provides steady state protection for the power-imbalance SLs. The Power-Imbalance-Flow trip encompasses the modules from the sensed points to the bistable contacts for total power, measured Axial Power Imbalance, and RCS flow measurement. The trip setpoint, determined from the flow and imbalance measurements, is compared to the total power measurement for trip determination. Portions of each channel are common with the Nuclear Overpower, RCS Pump-to-Power, Turbine trip, and Loss of Main Feedwater Pump trips. The ALLOWABLE VALUE for this function is given in the facility COLR because the cycle-specific changes affect the ALLOWABLE VALUE.

Nuclear Overpower RCS Flow and Measured Axial Power Imbalance trip channels are considered OPERABLE when OPERABILITY criteria 1 through 4 above are met.

The function's setpoint ALLOWABLE VALUE is selected to ensure that a trip occurs when the core power, axial power peaking, and reactor coolant flow conditions indicate an approach to DNB or fuel centerline melt limits.

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BASES (continued)

LCO (continued)

9. Main Turbine Trip (Control Oil Pressure)

The Main Turbine Trip RPS function provides an early reactor trip when the main turbine is lost at high power levels.

The turbine status is based on the control oil pressure of the main turbine. Each channel includes the instrumentation, from the oil, pressure sensing point at the turbine through the trip channel bistable contacts. Turbine trip channels are considered OPERABLE when OPERABILITY criteria 1 through 4 above are met. Each of the four turbine oil pressure switches feeds all four protection channels through buffers that continuously monitor the status of the contacts. Therefore, failure of any pressure switch renders all protection channels inoperable.

For the main turbine control oil pressure bistable, the ALLOWABLE VALUE of [45] psig is selected to provide a trip whenever feedwater pump control oil pressure drops below the normal operating range. To ensure that the trip is automatically enabled as required by the LCO, the reactor power bypass is set with an ALLOWABLE VALUE of [ ]% RTP.

The Turbine Trip is not required to protect against events that can create a harsh environment. Therefore, errors induced by harsh environments are not included in the determination of the setpoint ALLOWABLE VALUE.

10. Loss of Main Feedwater Pumps (Control Oil Pressure)

The Loss of Main Feedwater Pumps trip provides an early reactor trip at high power levels when both main feedwater pumps are lost.

The MFW pump status is determined by the oil pressure of the MFW pump turbines. Each channel includes the modules from the oil, pressure sensing point through the bistable contacts. The Loss of Main Feedwater Pumps trip channels are OPERABLE when OPERABILITY criteria 1 through 4 above are met.

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BASES (continued)

LCO (continued)

For the feedwater pump control oil, pressure bistable, the ALLOWABLE VALUE of [55] psig is selected to provide a trip whenever feedwater pump control oil pressure drops below the normal operating range. To ensure that the trip is automatically enabled as required by the LCO, the reactor power bypass is set with an ALLOWABLE VALUE of [ ]% RTP. The Loss of Main Feedwater Pumps trip is not required to protect against events that can create a harsh environment in the turbine building. Therefore, errors caused by harsh environments are not included in the determination of the setpoint ALLOWABLE VALUE.

11. Shutdown Bypass RCS High Pressure

The Shutdown Bypass RPS High Pressure is provided to allow withdrawing the CONTROL RODS prior to reaching the normal RCS Low Pressure trip setpoint. Each channel includes the modules from the RCS pressure taps through the bistable contacts. Portions of each channel are common with the RCS High Pressure, Low Pressure, and Variable Low Pressure trips. The Shutdown Bypass RCS High Pressure trip channels are considered OPERABLE when OPERABILITY criteria 1 through 3 above are met. [At this facility, the Shutdown Bypass RCS High Pressure function's ALLOWABLE VALUE is selected to ensure a trip occurs before:]

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

[For this facility, those required support systems which, upon their failure, do not result in the RPS instrumentation being declared inoperable and their justification are as follows:]

[For this facility, the supported systems impacted by the inoperability of the RPS instrumentation and the justification of whether or not each supported system is declared inoperable are as follows:] It should be noted

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BASES (continued)

LCO (continued)

that LCO 3.3.1, "Reactor Protection System (RPS) Instrumentation," may need to be augmented with additional Conditions if it is determined that the RPS provides support to other systems in the Standard Technical Specifications.

APPLICABILITY

In MODES 1 and 2, the following trips shall be OPERABLE because the reactor is critical in these MODES. These trips are designed to take the reactor subcritical to maintain the SLs during AOOs, and to assist the ESFAS in providing acceptable consequences during accidents.

1. Nuclear Overpower--High Setpoint;
2. RCS High Outlet Temperature;
3. RCS High Pressure;
4. RCS Low Pressure;
5. RCS Variable Low Pressure;
6. Reactor Building High Pressure;
7. Reactor Coolant Pump-to-Power; and
8. Nuclear Overpower RCS Flow and Measured Axial Power Imbalance.

Functions 1, 4, 5, 7 and 8 above may be bypassed in MODE 2 when RCS pressure is below [1720] psig, provided the Shutdown Bypass RCS High Pressure and the Nuclear Overpower Low setpoint trip are placed in operation. Under these conditions, the Shutdown Bypass RCS High Pressure trip and the Nuclear Overpower Low Setpoint trips act to prevent plant conditions from reaching a point where actuation of these functions is necessary.

Two other functions are required to be OPERABLE during portions of MODE 1. These are the Main Turbine Trip and the Loss of Main Feedwater Pumps trip. These functions are required to be OPERABLE above [45]% RTP and [15]% RTP, respectively. Analyses presented in Reference 7 have shown

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BASES (continued)

APPLICABILITY (continued)

that for operation below these power levels, these trips are not necessary to minimize challenges to the PORVs as required by NUREG-0737 (Ref. 5).

Because the only safety function of the RPS is to trip the CONTROL RODS, the RPS is not required to be OPERABLE in MODE 3, 4, or 5 if the reactor trip breakers are open, or the CONTROL ROD Drive (CRD) System is incapable of rod withdrawal. Similarly, th[illegible]t required to be OPERABLE in MODE 6 when [illegible]ODS are decoupled from the CRDs.

However, in MODE 3, 4, or 5, the Shutdown Bypass RCS High Pressure, and Nuclear Overpower Low Setpoint trips are required to be OPERABLE if the CRD trip breakers are closed and the CRD System is capable of rod withdrawal. Under these conditions, the Shutdown Bypass RCS High Pressure and Nuclear Overpower Low Setpoint trips are sufficient to prevent an approach to conditions that could challenge SLs.

For this LCO, a Note has been added in the APPLICABILITY to clarify that each function specified in Table 3.3.1-1 shall be treated as an independent entity with an independent Completion Time.

ACTIONS

Conditions A, B, C, and D are applicable to all RPS protection functions. A protection function channel is inoperable when it does not satisfy the OPERABILITY criteria for the function's channels. These criteria are outlined for each function in the LCO section of the Bases. All affected functions provided by an inoperable channel must be declared inoperable, and the plant must enter Condition A, B, or C. The most common causes of channel inoperability is outright failure or drift of the bistable or process module sufficient to exceed the tolerance allowed by the plant-specific setpoint analysis. Typically the drift is small, which would result in a delay of actuation rather than a total loss of function. This determination is made during the performance of a CHANNEL FUNCTIONAL TEST when the process instrument is set up for adjustment to bring it within specification. If the trip setpoint is not consistent with the ALLOWABLE VALUE in Table 3.3.1-1, the channel must be declared inoperable immediately.

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BASES (continued)

ACTIONS (continued)

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

If a facility is to take credit for topical reports as the basis for justifying Completion Times, the reports must be supported by an NRC staff Safety Evaluation Report (SER) that establishes the acceptability of each topical report for that facility.

Condition A

Condition A applies to each one of the RPS functions presented in Table 3.3.1-1.

Condition A applies if one or more functions in one protective channel become inoperable. Required Action A.1 requires placing the affected protection channel in bypass or trip. This places all RPS functions in a two-out-of-three logic configuration and prevents bypass of any other channel. In this configuration, the RPS can still perform its safety function in the presence of a random failure of any single channel. Alternatively, the inoperable channel can be placed in trip. Tripping the affected protection channel places all RPS functions in a one-out-of-three configuration.

Operation in the one-out-of-three configuration may continue indefinitely. In this configuration, the RPS is capable of performing its trip function in the presence of any single random failure. A second channel may be bypassed temporarily for surveillance testing while maintaining the RPS capability to withstand a random failure of a single channel. The 1-hour Completion Time is sufficient to perform the Required Action.

Per Required Action A.2.1 or Required Action A.2.2, if the inoperable channel is bypassed, it must either be restored to OPERABLE status or tripped within 48 hours. This action is necessary because the RPS was accepted by the NRC as a four channel design. Required Action A.2.1 is preferred because it restores the full functional capability of the

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BASES (continued)

ACTIONS (continued)

RPS. Action A.2.2 places the RPS in a one-out-of-three configuration as discussed above. The 48-hour Completion Time is based on the NRC SER of Reference 8.

Condition B

Condition B applies to each of the RPS functions presented in Table 3.3.1-1.

Condition B applies if one or more functions in two protection channels become inoperable.

The Required Action B.1 and Required Action B.2 place one of two inoperable protection channels in trip and the other in bypass. These Required Actions place all RPS functions in a one-out-of-two logic configuration to prevent bypass of a second channel. In this configuration, the RPS can still perform its safety functions in the presence of a random failure of any single channel. The 1-hour Completion Time is sufficient time to perform the Required Actions.

Required Action B.3 is necessary because the RPS was accepted by the NRC as a four channel design. Restoring one channel to OPERABLE status puts the RPS into Condition A which allows the system to be in a one-out-of-three configuration. The 48-hour Completion Time is based on the NRC SER of Reference 7.

Condition C

Condition C applies to each one of the RPS functions presented in Table 3.3.1-1.

Condition C applies if the Required Actions of Condition A or B cannot be completed within the allowed Completion Time.

Required Action C.1 and Required Action C.2 place the plant in a MODE in which the RPS is not required to be OPERABLE. The 6 hours allowed for completion is a reasonable time, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging plant systems.

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BASES (continued)

ACTIONS (continued)

Condition D

Condition D applies to each one of the RPS functions presented in Table 3.3.1-1.

Required Action D.1 verifies that all required support or supported features associated with the other redundant channel(s) are OPERABLE within a Completion Time of 1 hour. The specified Completion Time is sufficient for plant operations personnel to make this determination. If verification determines loss of functional capability, LCO 3.0.3 must be immediately entered. However, if the support or supported feature LCO or RPS LCO take into consideration the loss of function situation then LCO 3.0.3 may not need to be entered.

SURVEILLANCE REQUIREMENTS

The SRs are modified by two Notes. The first Note directs the reader to Table 3.3.1-1 to determine the correct SRs to perform for each trip function. Most functions are subject to CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION.

The second Note defines a channel as being OPERABLE for up to 8 hours while bypassed for surveillance testing. The Note allows channel bypass for testing without defining it as inoperable although during this time period it cannot actuate a reactor trip. This allowance is based on the RPS reliability analysis (Ref. 8) assumption that 8 hours is the average time required to perform channel surveillance.] The analysis demonstrated that the 8-hour testing allowance does not significantly reduce the probability that the RPS will trip when necessary. It is not acceptable to routinely remove channels from service for more than 8 hours to perform required surveillance testing. Such a practice would be contrary to the assumptions of the reliability analysis that justified the LCO's Completion Times.

If a facility is to take credit for topical reports for the basis for justifying Surveillance Frequencies, the topical reports should be supported by an NRC staff SER that establishes the acceptability of each topical report for that facility.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between two instrument channels could be an indication of excessive instrument drift in one of the channels or something more serious. CHANNEL CHECK will detect gross channel failure, thus, it is the 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 combination of the channel instrument uncertainties, including isolation, indication, and readability. If a channel is outside the match criteria, it may be an indication that the transmitter or the signal processing equipment drifted outside its limit. If the channels are within the match criteria, it is an indication that the channels are OPERABLE. If the channels are normally off scale during times when Surveillance 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 downscale.

The Surveillance Frequency, about once every shift, is based on operating experience that demonstrates the rarity of channel failure. Thus, performance of the CHANNEL CHECK guarantees that undetected, overt channel failure is limited to 12 hours. Because 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 protective function due to failure of redundant channels. The CHANNEL CHECK supplements less formal, but more frequent checks of channel OPERABILITY during normal operational use of the displays associated with the LCOs required channels.

In the case of functions that trip on a combination of several measurements, such as the Nuclear Overpower RCS flow and Measured Axial Power Imbalance function, the CHANNEL CHECK must be performed on each input.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

This places all RPS functions in a two-out-of-three logic configuration.

SR 3.3.1.2

This SR is the performance of a HEAT BALANCE CALIBRATION for the power range channels every 24 hours when greater than 15% of RTP. The outputs of the power range channels are normalized to the calorimetric. The HEAT BALANCE CALIBRATION consists of a comparison of the results of the calorimetric with the power range channel output. A Note clarifies that this Surveillance is required only if reactor power is > 15% of RTP. At lower power levels, calorimetric data are inaccurate.

The power range channel's output shall be adjusted consistent with the calorimetric results if the calorimetric is ≥ 102% of the power range channel's output. The value of 2% is adequate, because this value is assumed in the safety analyses of Reference 9. These checks, and if necessary the adjustment of the power range channels ensure that channel accuracy is maintained within the analyzed error margins. The 24-hour Frequency is adequate, based plant operating experience, which demonstrates the change in the difference between the power range indication and the calorimetric results rarely exceeds a small fraction of 2% in any 24-hour period. Furthermore, the control room operators monitor redundant indications and alarms to detect deviations in channel outputs.

SR 3.3.1.3

A comparison of power range nuclear instrumentation channels against incore detectors shall be performed at a 31-day Frequency when greater than 15% RTP. CHANNEL CALIBRATION that adjusts the measured imbalance to agree with the incore measurements is necessary when the out-of-core Measured Axial Power Imbalance minus the incore imbalance is ≥ 2% imbalance. The calculation of the ALLOWABLE VALUE envelope assumes a difference in out-of-core to incore measurements of 2.5%. Additional inaccuracies beyond those that are measured are also included in the setpoint envelope calculation. The 31-day Frequency is adequate, considering that long-term drift of the excore linear amplifiers is

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

small and burnup of the detectors is slow. Also, 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. The slow changes in neutron flux during the fuel cycle can also be detected at this interval.

SR 3.3.1.4

A CHANNEL FUNCTIONAL TEST verifies the function of the trip, interlock, and alarm functions of the channel. In the test, a simulated or actual signal is inserted as close to the sensor as practicable to verify required trips, interlocks, and alarms function when the input is beyond the trip setpoint. Where the design has made provisions for including sensors in the CHANNEL FUNCTIONAL TEST, the test signal shall be inserted at that point. "As found" and "as left" values for bistable trip setpoints are recorded. Bistable setpoints must be found within the ALLOWABLE VALUE specified in the LCO. The difference between the current "as found" and the previous "as left" setpoints must be within the drift allowance used in the setpoint analysis. Recalibration of the bistable setpoint restores the OPERABILITY of an otherwise functional component that does not meet these criteria. However, repeated failures of the same channel over a small number of test intervals should be evaluated as potentially indicating a deterministic failure that cannot be corrected by recalibration.

The setpoint and [illegible] accuracy of each channel's equipment located outside the reactor building shall be verified every [45] days on a STAGGERED TEST BASIS (i.e., Channel A at [45] days after initial startup, Channel B at [90] days, Channel C at [135] days, Channel D at [180] days, and then Channel A again). Calculations have shown that the 45-day Frequency maintains a high level of reliability of the RPS (Ref. 8).

SR 3.3.1.5

This SR is the performance of a CHANNEL CALIBRATION every [92] days. This CHANNEL CALIBRATION normalizes the power, range channel output to the calorimetric coincident, with the imbalance output being normalized to the imbalance condition predicted by the incore neutron detector system.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

The calibration for both imbalance and total power is integrated in the power imbalance detector calibration procedure. The [92]-day Surveillance Frequency specified for the Nuclear Overpower trip string is consistent with the drift assumptions made in the plant-specific setpoint analysis (Ref. 4). Furthermore, operating experience shows the reliability of the trip string is acceptable when calibrated on this interval. A Note clarifies that the neutron detectors are not required to be tested as part of the CHANNEL CALIBRATION. There is no adjustment that can be made to the detectors. Furthermore, adjustment of the detectors is unnecessary because they are passive devices, with minimal drift. Slow changes in detector sensitivity are compensated for by performing the daily calorimetric calibration and the monthly axial channel calibration.

SR 3.3.1.6

This SR is the performance of a CHANNEL CALIBRATION every [18] months. CHANNEL CALIBRATION is a complete check of the instrument channel, including the detector. The test verifies that the channel responds to the measured parameter with the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift between successive tests to ensure that the instrument channel remains operational with the setpoint within the assumptions of the plant-specific setpoint analysis. Transmitter "as found" and "as left" values are recorded and used to verify drift assumptions. For OPERABLE channels, CHANNEL CALIBRATION shall find that measurement errors and bistable setpoint errors are within the assumptions of the plant-specific setpoint analysis. Measurement and setpoint error determination and readjustment must be performed consistent with the assumptions of the plant-specific setpoint analysis.

Recalibration restores OPERABILITY of an otherwise functional component found to have errors larger than those assumed by the setpoint analysis. However, repeated failures of the same channel over a relatively small number of test intervals must be considered as potentially indicating a deterministic failure that cannot be corrected by recalibration.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

Field transmitters may be calibrated in place, removed and calibrated in a laboratory, or replaced with an equivalent, laboratory calibrated unit. RTD channels may be calibrated in place using cross-calibration techniques, or in a test bath after removal from piping. For cross calibration, at least one RTD should be replaced with a newly calibrated RTD during each refueling cycle to ensure accurate RTD cross calibration. This replacement RTD must be the same model as the remaining RTDs. Using a newly calibrated RTD as a reference ensures that RTD signal drift continues to remain random rather than systematic, and is within the limits specified in the plant-specific setpoint analysis. The replacement interval may be extended to alternate refueling if it is demonstrated that, over the extended interval, the RTDs drift is random rather than systematic, and is bounded by the plant-specific setpoint analyses assumptions. This determination may use results of statistical analysis of operating and calibration data from similar plants, using the same model of RTD in the same environmental conditions.

The Surveillance Frequency is based on the assumption of an [18]-month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

SR 3.3.1.7

This SR ensures that the channel actuation response times are less than or equal to the maximum values assumed in the accident analysis. Individual component response times are not modeled in the analyses. The analyses model the overall, or total, elapsed time from the point at which the parameter exceeds the analytical limit at the sensor to the point of rod insertion. The acceptable response times of the relevant trip channels are included in Reference 10. The response times include contributions from the sensor, the RPS processing equipment, the RTMs, the CRD breakers, and the CRD mechanism release. This test may be performed in one measurement or in overlapping segments, with verification that all components are tested.

A Note to the Surveillance indicates that neutron detectors may be excluded from RPS RESPONSE TIME testing. This Note is necessary because of the difficulty in generating an appropriate detector input signal. Excluding the detectors

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

is acceptable because the principles of detector operation together with the calibrations against heat balance calculations and incore detectors, ensure a virtually instantaneous response.

Response time tests are conducted on a [18]-month STAGGERED TEST BASIS. This results in the interval between successive tests of a given channel of n times [18] months, where n is the number of channels in the function. Testing 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 [18] months. The [18]-month test Frequency is based on plant operating experience, which shows that random failures of instrumentation components causing serious response time degradation, but not channel failure, are infrequent occurrences. Response times cannot be determined at power, because equipment operation is required.

REFERENCES

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

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

3. Title 10, Code of Federal Regulations, Part 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants."

4. [Unit Name], "[Plant-specific Setpoint Methodology]."

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

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

7. Letter from A.C. Thadani (USNRC) to C.W. Smythe (B&W Owners Group), "NRC Evaluation of BWOG Topical Report, BAW 10167" and Supplement 1, "Justification for Increasing the Reactor Trip System On-Line Test Interval," dated December 5, 1988.

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BASES (continued)

REFERENCES (continued)

8. BAW-10167, "Justification for Increasing the Reactor Trip System On-Line Test Intervals," May 1986.

B 3.3 INSTRUMENTATION

B 3.3.2 Reactor Protection System (RPS) Manual Reactor Trip

BASES

BACKGROUND

The Reactor Protection System (RPS) Manual Reactor Trip provides the operator with the capability to trip the reactor from the control room in the absence of any other trip condition. Manual trip is provided by a trip push button on the main control board. This trip is independent of the automatic trip system. As shown in Figure B 3.3.1-1, power for the CONTROL ROD drive (CRD) breaker undervoltage coils and contactor coils comes from the reactor trip modules (RTMs). The manual trip switches are located between the RTM output and the breaker undervoltage coils. Opening of the switches opens the lines to the breakers, tripping them. The switches also energize the breaker shunt trip mechanisms. There is a separate switch in series with the output of each of the four RTMs. All switches are actuated through a mechanical linkage from a single push button for each train.

APPLICABLE SAFETY ANALYSES

The Manual Reactor Trip ensures that the control room operator can initiate a reactor trip at any time. The Manual Reactor Trip function is required as a backup to the automatic trip functions, and to allow operators to shut down the reactor whenever any parameter is rapidly trending toward its trip setpoint.

The Manual Reactor Trip function satisfies Criterion 3 of the NRC Interim Policy Statement.

LCO

The LCO on the Manual Reactor Trip requires that the trip shall be operable whenever the reactor is critical or any time the CONTROL ROD breakers are closed and rods are capable of being withdrawn, including shutdown bypass. This enables the operator to terminate any reactivity excursion that in the operator's judgment requires protective action, even if no automatic trip condition exists.

(continued)

(continued)

BASES (continued)

LCO (continued)

Manual Reactor Trip functions are considered OPERABLE when:

a. All components in each protection channel that are necessary to provide a manual trip are functional and in service; and

b. Required surveillance testing is current and has demonstrated performance within each surveillance test's acceptance criteria.

[For this facility, the following support features are required to be OPERABLE to ensure RPS Manual Reactor Trip OPERABILITY:]

[For this facility, those required support features which upon their failure do not result in the RPS Manual Reactor Trip channel being declared inoperable and their justification are as follows:]

[For this facility, the supported features impacted by the inoperability of an RPS Manual Reactor Trip channel and the justification of whether or not each supported system is declared inoperable are as follows:] It should be noted that LCO 3.3.2 may need to be augmented with additional conditions if it is determined that the RPS manual reactor trip provides support to other systems in the Standard Technical Specifications.

APPLICABILITY

The Manual Reactor Trip function is required to be OPERABLE in MODES 1 and 2. It is also applicable in MODES 3, 4, and 5 if the CRD trip breakers are closed and the CRD System is capable of rod withdrawal. Because the only safety function of the RPS is to trip the CONTROL RODS, the Manual Reactor Trip function is not needed in MODE 3, 4, or 5 if the reactor trip breakers are open or the CRD System is incapable of rod withdrawal. Similarly, the RPS Manual Reactor Trip is not needed in MODE 6 when the CONTROL RODS are decoupled from the CRDs.

(continued)

BASES (continued)

ACTIONS

The Manual Reactor Trip channel is inoperable when it does not satisfy the OPERABILITY criteria for the channel's function. These criteria are outlined in the LCO section of the Bases.

Condition A

Condition A applies when one or more Manual Reactor Trip channels are found inoperable. One hour is allowed to restore channels to OPERABLE status. The automatic functions and various alternative manual trip methods, such as removing power to the RTMs, are still available. The 1-hour Completion Time is sufficient time to correct minor problems.

Condition B

With one or more Manual Reactor Trip channels inoperable and unable to be returned to OPERABLE status within one hour, the plant must be placed in a MODE in which manual trip is not required. Required Action B.1 and Required Action B.2 place the plant in at least MODE 3 with the CRD trip breakers open within 6 hours. The 6 hours allotted is a reasonable time, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging plant systems.

Condition C

Condition C is applicable to the RPS Manual Reactor Trip channels. Required Action C.1 verifies that all required support features associated with the other redundant channel(s) are OPERABLE within a Completion Time of 1 hour. The specified Completion Time is sufficient for plant operations personnel to make this determination. If verification determines loss of functional capability, LCO 3.0.3 must be immediately entered. However, if the support feature LCO or RPS manual reactor trip LCO takes into consideration the loss of function situation, then LCO 3.0.3 may not need to be entered.

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BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.3.2.1

This SR requires the performance of a CHANNEL FUNCTIONAL TEST of the Manual Reactor Trip function. This test verifies the OPERABILITY of the Manual Reactor Trip by actuation of the end devices, the CRD trip breakers. The Frequency, prior to each startup if not performed within the preceding 7 days, ensures the OPERABILITY of the manual trip function prior to achieving criticality. The Frequency was developed considering it was prudent that these surveillances only be performed during a plant outage. This was due to the plant conditions needed to perform the surveillance and the potential for unplanned plant transients if the surveillance is performed with the reactor at power. Operating experience has shown these components usually pass the surveillance when performed prior to each startup.

REFERENCES

None.

B 3.3 INSTRUMENTATION

B 3.3.3 Reactor Protection System (RPS)-Reactor Trip Module (RTM)

BASES

BACKGROUND

The RPS consists of four independent protection channels, each containing an RTM. Figure B 3.3.1-1 shows a typical RPS protection channel and the relationship of the RTM to the RPS instrumentation, manual trip, and control rod drive (CRD) trip devices. The RTM receives bistable trip signals from the functions in its own channel, as well as channel trip signals from the other three RPS-RTMs. The RTM provides these signals to its own two-out-of-four trip logic and transmits its own channel trip signal to the two-out-of-four logic of the RTMs in the other three RPS channels. Whenever any two RPS channels transmit channel trip signals, the RTM logic in each channel actuates to remove 120 V ac power from its associated CRD trip device.

The RPS trip scheme consists of series contacts that are operated by bistables. During normal plant operations, all contacts are closed and the RTM channel trip relay remains energized. However, if any trip parameter exceeds its setpoint, its associated contact opens, which de-energizes the channel trip relay.

When the RTM channel trip relay de-energizes, several things occur:

1. The KA relay de-energizes the four (4) output logic relays. Each of these relays "informs" its associated RPS channel that a reactor trip signal has occurred in RPS Channel A;

2. The KA1 contacts in the trip device circuitry powered by RPS Channel A open, but the trip device remains energized through the closed KB1, KC1, and KD1 contacts. This condition exists in each RPS-RTM. Each RPS RTM controls power to a trip device; and

3. The KA contact in parallel with the channel reset switch opens and the trip is sealed in. The channel reset switch must be depressed, after the trip condition has cleared, to re-energize the channel trip relay.

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(continued)

BASES (continued)

BACKGROUND (continued)

While the information presented pertains only to RPS Channel A, it is applicable to any RPS channel.

When the second RPS channel senses a reactor trip condition, the following occurs:

1. The channel trip relay for that channel de-energizes;

2. The output logic relays for the second channel de-energize and open contacts that supply power to the trip devices; and

3. With contacts opened by two separate RPS channels, power to the trip devices is interrupted, and the control rods fall into the core.

Note, a minimum of two out of four RTMs must sense a trip condition to cause a reactor trip. Also, since the bistable relay contacts for each function are in series with the channel trip relays, two channel trips caused by different trip functions can result in a reactor trip.

APPLICABLE SAFETY ANALYSES

Accident analyses rely on a reactor trip for protection of reactor core integrity, reactor coolant pressure boundary integrity, and reactor building OPERABILITY. A reactor trip must occur when needed to prevent accident conditions from exceeding those calculated in the accident analyses. More detailed descriptions of the applicable accident analyses are found in the bases for each of the RPS trip functions in LCO 3.3.1, "Reactor Protection System (RPS) Instrumentation."

RTM response time is included in the overall required response time for each RPS trip and is not specified separately.

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

(continued)

BASES (continued)

LCO

The RTM LCO requires all four RTMs to be OPERABLE. Failure of any RTM renders a portion of the RPS inoperable and reduces the reliability of the affected functions. RTMs are considered OPERABLE when:

- All channel components necessary to provide a reactor trip are functional and in service: and
- Required Surveillance testing is current and has demonstrated performance within each Surveillance test's acceptance criteria.

Four RTMs shall be OPERABLE to ensure that a reactor trip would occur if needed any time the reactor is critical. OPERABILITY is defined as the RTM being able to receive and interpret trip signals from its own channel and from other RPS channels and to open its associated trip device.

Requiring four channels to be OPERABLE ensures that a minimum of two RPS channels will remain OPERABLE if a single failure has occurred in one channel and a second channel has been bypassed for surveillance or maintenance. This two-out-of-four trip logic also ensures that a single RPS channel failure will not cause an unwanted reactor trip. Violation of this LCO could result in a trip signal not causing a reactor trip when needed.

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

[For this facility, those required support features which, upon their failure, do not result in the RTM being declared inoperable and their justification are as follows:]

[For this facility, the supported features impacted by the inoperability of an RTM and the justification for whether or not each supported system is declared inoperable are as follows:] It should be noted that LCO 3.3.3 may need to be augmented with additional Conditions if it is determined that the RTM provides support to other systems in the Standard Technical Specifications.

(continued)

BASES (continued)

APPLICABILITY

The RTMs are required to be OPERABLE in MODES 1 and 2 and whenever the reactor CONTROL RODS are, or can be, withdrawn. Rod withdrawal can occur in MODES 3, 4, and 5 whenever the CRD trip breakers are closed and the CRDs are capable of rod withdrawal. The RTMs are designed to ensure a reactor trip would occur if needed any time the reactor is critical. Since this condition can exist in all of these MODES, the RTMs must be OPERABLE.

ACTIONS

An RTM is inoperable when it does not satisfy the OPERABILITY criteria for the channel's function. These criteria are outlined in the LCO section.

When more than one RTM channel is inoperable, facility operation is not allowed to continue. Therefore, LCO 3.0.3 must be immediately entered, if applicable in the current mode of operation.

Condition A

Condition A applies when one RTM becomes inoperable.

When an RTM is inoperable, the associated CRD breaker must then be placed in a condition that is equivalent to a tripped condition for the RTM. Required Action A.1.1 or Required Action A.1.2 require this either by tripping the CRD breaker or removing power to the trip device. Tripping one RTM or removing power opens one set of CRD trip devices. Power to hold up control rods is still provided via the parallel CRD trip device(s). Therefore, a reactor trip will not occur until a second protection channel trips.

To ensure the trip signal is registered in the other channels, Required Action A.2 requires that the inoperable RTM be removed from the cabinet. This action causes the electrical interlocks to indicate a tripped channel in the remaining three RTMs. Operation in this condition is allowed indefinitely because the actions put the RPS into a one-out-of-three configuration. The 1-hour Completion Time is sufficient time to perform the Required Actions.

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BASES (continued)

ACTIONS (continued)

Condition B

Condition B applies if the Required Actions of Condition A are not met within the required Completion Time; in this case, the plant must be placed in a MODE in which the LCO does not apply. This is done by placing the plant in at least MODE 3, with all CRD trip breakers open or with all power to the CRD System removed, within 6 hours. The 6 hours allotted is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging plant systems.

Condition C

Condition C applies when one RTM is inoperable.

Required Action C.1 verifies that all required support features associated with the other redundant RTMs are OPERABLE within a Completion Time of 1 hour. The specified Completion Time is sufficient for plant operations personnel to make this determination. If verification determines loss of functional capability, LCO 3.0.3 must be immediately entered. However, if the support feature LCO or RPS-RTM LCO takes into consideration the loss-of-function situation, then LCO 3.0.3 may not need to be entered.

SURVEILLANCE REQUIREMENTS

SR 3.3.3.1

The SRs include performance of a CHANNEL FUNCTIONAL TEST every [45] days on a STAGGERED TEST BASIS. This test shall include verification of the OPERABILITY of the RTM and its ability to receive and properly respond to channel trip and reactor trip signals.

[For this facility, a CHANNEL FUNCTIONAL TEST includes the following:]

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

REFERENCES

None.

B 3.3 INSTRUMENTATION

B 3.3.4 CONTROL ROD Drive (CRD) Trip Devices

BASES

BACKGROUND

The Reactor Protection System (RPS) contains multiple CRD trip devices: 2 AC trip breakers, 2 DC trip breaker pairs, and 16 silicon controlled rectifier (SCR) relays. The system has two separate paths (or channels), with each path having one AC in series with a pair of DC breakers or 8 SCR relays in parallel. Each path provides independent power to the CRDs. Either path can provide sufficient power to operate the entire CRD system.

Figure B 3.3.1-2 illustrates the configuration of CRD trip devices. To trip the reactor, power to the CRDs must be removed. Loss of power causes the CRD's mechanisms to release the control rods, which then fall by gravity into the core.

Power to CRDs is supplied from two separate plant sources through the AC trip circuit breakers. These breakers are designated A and B, and their undervoltage and shunt trip coils are powered by RPS channels A and B, respectively. From the circuit breakers, the CRD power travels through voltage regulators and stepdown transformers. These devices in turn supply redundant buses that feed the DC power supplies and the regulating rod power supplies.

The DC power supplies rectify the AC input and supply power to hold the safety rods in their fully withdrawn position. One of the redundant power sources supplies phase A and the other phase C. Either phase being energized is sufficient to hold the rod. Two breakers are located on the output of each power supply. Each breaker controls power to one of the four safety rod groups. The undervoltage and shunt trip coils on the two circuit breakers on the output of one of the power supplies is controlled by RPS channel C, and the other two breakers are controlled by RPS channel D.

In addition to the DC power supplies, the redundant buses also supply power to the regulating and auxiliary power supplies. These power supplies consist of SCRs that are gated on by programming lamps. Programming lamp power is controlled by contactors (E and F), which are controlled by

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BASES (continued)

BACKGROUND (continued)

RPS power. One of the redundant programming lamp supplies is controlled by RPS channel C, and the other supply is controlled by RPS channel D.

The AC breaker and DC breakers are in series in one of the power supplies and the redundant AC breaker and DC breakers are in series in the other power supply to the CONTROL RODS. The logic required to cause a reactor trip is the opening of a circuit breaker in each of the redundant power supplies. (The pair of DC circuit breakers on the output of the power supply are treated as one breaker.) This is known as a one-out-of-two taken twice logic. The following examples illustrate the operation of the reactor trip circuit breakers.

1. If the A AC circuit breaker opens:

   a. The input power to associated DC power supply is lost; and

   b. The SCR supply from the associated power source is lost.

2. If the D DC circuit breaker(s) and F contactors open:

   a. The output of the redundant DC power supply is lost and the safety rods de-energize; and

   b. When the F contactor opens, programming lamp power is lost and the regulating rods will be de-energized.

3. The combination of 1 and 2 causes a reactor trip.

Any other combination of at least one circuit breaker opening in each power supply will cause a reactor trip.

In summary, two tripped RPS channels will cause a reactor trip. For example, a reactor trip occurs if RPS channel B senses a low Reactor Coolant System (RCS) pressure condition and RPS channel C senses a variable low RCS pressure condition. When the channel B bistable relay de-energizes, the channel trip relay de-energizes and opens its associated contacts. The same thing occurs in channel C, except the variable lower pressure bistable relay de-energizes the C

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BASES (continued)

BACKGROUND (continued)

channel trip relay. When the output logic relays de-energize, the 3 and C contacts in the undervoltage and E and F contacts de-energize, all circuit breakers open and programming lamp power is removed. All rods fall into the core, resulting in a reactor trip.

APPLICABLE SAFETY ANALYSES

Accident analyses rely on a reactor trip for protection of reactor core integrity, reactor coolant pressure boundary integrity, and reactor building OPERABILITY. A reactor trip must occur when needed to prevent accident consequences from exceeding those calculated in the accident analyses. The CONTROL ROD insertion [illegible] ensure that adequate rod worth is available upon reac[illegible] to shut down the reactor to the required SHUTDOWN MA[illegible]. Further, OPERABILITY of the CRD trip devices ensures that all CONTROL RODS (except Group 8) will trip when required. More detailed descriptions of the applicable accident analyses are found in the Bases for each of the individual RPS trip functions in LCO 3.3.1, "Reactor Protection System (RPS) Instrumentation."

The CRD trip devices satisfy Criterion 3 of the NRC Interim Policy Statement.

LCO

The LCO requires all of the CRD trip devices to be OPERABLE. Failure of any CRD trip device renders a portion of the RPS inoperable and reduces the reliability of the affected functions.

The LCO on the CRD trip device ensures that trip devices are OPERABLE. CRD trip devices are considered OPERABLE when:

- All channel components necessary to provide reactor trip are functional and in service.
- Required surveillance testing is current and has demonstrated performance within each surveillance test's acceptance criteria.

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BASES (continued)

LCO (continued)

Without reliable CRD reactor trip circuit breakers and associated support circuitry, a reactor trip cannot occur when initiated either automatically or manually.

All CRD trip devices shall be OPERABLE to ensure that the reactor remains capable of being tripped any time it is critical. OPERABILITY is defined as the CRD trip device being able to receive a reactor trip signal and to respond to this trip signal by interrupting AC power to the CRDs. Both of the AC breaker's trip devices and the breaker must be functioning properly in order for the AC breaker to be OPERABLE.

Requiring all breakers and SCR relays to be OPERABLE ensures that at least one device in each of the two power paths to the CRDs will remain OPERABLE even with a single failure. Requiring all devices OPERABLE also ensures that a single failure will not cause an unwanted reactor trip.

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

[For this facility, those required support systems which, upon their failure, do not result in the CRD trip devices being declared inoperable and their justification are as follows:]

[For this facility, the supported systems impacted by the inoperability of the CRD trip devices and the justification of whether or not each supported system is declared inoperable are as follows:] It should be noted that LCO 3.3.4 may need to be augmented with additional Conditions if it is determined that the CRD trip devices provide support to other systems in the Standard Technical Specifications.

APPLICABILITY

The CRD trip devices shall be OPERABLE in MODES 1 and 2, and in MODES 3, 4, and 5 when the CRD breakers are in the closed position and the CRD system is capable of rod withdrawal.

The CRD trip devices are designed to ensure that a reactor trip would occur if needed any time the reactor is critical.

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BASES (continued)

APPLICABILITY (continued)

Since this condition can exist in all of these MODES, the CRD trip devices shall be OPERABLE.

ACTIONS

A CRD trip device channel is inoperable when it does not satisfy the OPERABILITY criteria for the channel's function. These criteria are outlined in the LCO section above.

When the number of inoperable CRD trip device combinations stated in Condition A or B are exceeded, then facility operation is not allowed to continue in this degraded condition. Therefore, LCO 3.0.3 shall be immediately entered, if applicable in the current MODE of operation.

Condition A

Condition A represents reduced redundancy in the CRD trip function. Condition A applies if:

- One SCR relay is inoperable; or
- One diverse trip function (undervoltage or shunt trip device) is inoperable in any CRD trip breaker; or
- One diverse trip function is inoperable in both DC trip breakers associated with one protection channel. In this case, the inoperable trip function does not need to be the same for both breakers.

Condition A does not apply if a combination of one SCR relay and one CRD trip breaker [or breaker pair] diverse trip functions are inoperable.

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

If one SCR relay or one of the diverse trip functions on a trip breaker [or pair] becomes inoperable, the preferred action is to restore the inoperable device to OPERABLE status. If a CRD trip device cannot be repaired, actions must be taken to preclude the inoperable CRD trip device from preventing a reactor trip when needed. This is done by manually tripping the inoperable CRD trip device or removing power from the channel containing the inoperable CRD trip device. Either of these actions places the affected CRDs

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BASES (continued)

ACTIONS (continued)

into a one-out-of-two trip configuration, which precludes a single failure, which in turn could prevent tripping of the reactor. [For this facility, the justification of the 48-hour Completion Time is as follows:]

Condition B

Condition B represents a loss of redundancy for the CRD trip function. Condition B applies when:

- More than one SCR relay is inoperable in the same protection channel; or
- One CRD trip breaker will not function on either undervoltage or shunt trip, [or trip breaker pair]; or
- Both diverse trip functions are inoperable in one or both DC trip breakers associated with one protection channel.

Condition B does not apply to a combination of inoperable SCR relays and CRD trip breakers.

Required Action B.1 and Required Action B.2 are the same as Required Action A.2 and Required Action A.3, but the Completion Time is shortened. The 1-hour Completion Time allotted to trip or remove power to the CRD device allows the operator to take all the appropriate actions for the inoperable device and still ensures the risk involved is acceptable.

Condition C

If the Required Actions of Condition A or B are not met within the required Completion Time, the plant must be placed in a MODE in which the LCO does not apply. This is done by placing the plant in at least MODE 3, with all CRD trip breakers open or with all power to the CRD system removed, within 6 hours. The 6 hours allotted to reach MODE 3 is a reasonable time, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging plant systems.

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BASES (continued)

ACTIONS (continued)

Condition D

Condition D applies when one CRD trip device is inoperable.

Required Action D.1 verifies that all required support features associated with the other redundant CRD trip devices are OPERABLE within a Completion Time of 1 hour.

The specified Completion Time is sufficient for plant operations personnel to make this determination.

If verification determines loss of functional capability, LCO 3.0.3 shall be immediately entered. However, if the support feature LCO or CRD trip devices LCO takes into consideration the loss-of-function situation, then LCO 3.0.3 may not need to be entered.

SURVEILLANCE REQUIREMENTS

SR 3.3.4.1

SR 3.3.4.1 is to perform a CHANNEL FUNCTIONAL TEST every 31 days. This test verifies the OPERABILITY of the trip devices by actuation of the end devices. This test also includes independent verification of the undervoltage and shunt trip mechanisms of the AC breakers. The Surveillance Frequency of 31 days is based on operating experience, which has demonstrated that failure of more than one channel of a given function in any 31-day interval is a rare event.

REFERENCES

None.

B 3.3 INSTRUMENTATION

B 3.3.5 Engineered Safety Feature Actuation System (ESFAS) Instrumentation

BASES

BACKGROUND

The ESFAS initiates necessary safety systems, based upon the values of selected unit parameters, to protect against violating core design limits and reactor coolant pressure boundary, and to mitigate accidents.

ESFAS actuates the following systems:

- High pressure injection (HPI);
- Low pressure injection (LPI);
- Reactor building (RB) cooling;
- RB spray;
- RB isolation; and
- Emergency diesel generator (EDG) start.

ESFAS also provides a signal to the Emergency Feedwater Isolation and Control (EFIC) System. This signal initiates emergency feedwater (EFW) when HPI is initiated.

The ESFAS operates in a distributed manner to initiate the appropriate systems. The ESFAS does this by determining the need for actuation in each of three channels monitoring each actuation parameter. Once the need for actuation is determined, the condition is transmitted to automatic actuation logics, which perform the two-out-of-three logic to determine the actuation of each end device. Each end device has its own automatic actuation logic, although all automatic actuation logics take their signals from the same point in each channel for each parameter.

Four parameters are used for actuation:

- Low Reactor Coolant System (RCS) pressure;
- Low Low RCS pressure;
- High RB pressure; and
- High High RB pressure.

LCO 3.3.5, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," covers only the instrumentation channels that measure these parameters. These channels include all intervening equipment necessary to produce

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BASES (continued)

BACKGROUND (continued)

actuation before the measured process parameter exceeds the limits assumed by the accident analysis. This includes sensors, bistable devices, operational bypass circuitry, block timers, and output relays. LCO 3.3.6, "Engineered Safety Feature Actuation System (ESFAS) Manual Initiation," and LCO 3.3.7, "Engineered Safety Feature Actuation System (ESFAS) Automatic Actuation Logic," provide requirements on the manual initiation and automatic actuation logic functions. LCO 3.3.11, "Emergency Feedwater Initiation and Control (EFIC) Instrumentation," contains the requirements for the HPI EFW initiation logic.

The ESFAS consists of six protection channels divided into two trains of three channels each. Each protection channel includes bistable inputs from one instrumentation channel of RCS low pressure, RCS low low pressure, RB high pressure, and RB high high pressure. Specific combinations of bistable trips for these parameters cause a protection channel trip for each ESFAS System. Figure B 3.3.5-1 illustrates how instrumentation channel trips combine to cause protection channel trips. Automatic actuation logics combine the three protection channel trips in each train to actuate the individual engineered safety feature (ESF) components needed to initiate each ESF System.

The RCS pressure sensors are common to both trains and to both the high and high high pressure functions. Isolation is provided via separate bistables for each train. Separate RB pressure sensors are used for the high and high high pressure functions in each train, and separate sensors are used for each train.

The matrix below identifies the measurement channels and the function actuated by each.

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BASES (continued)

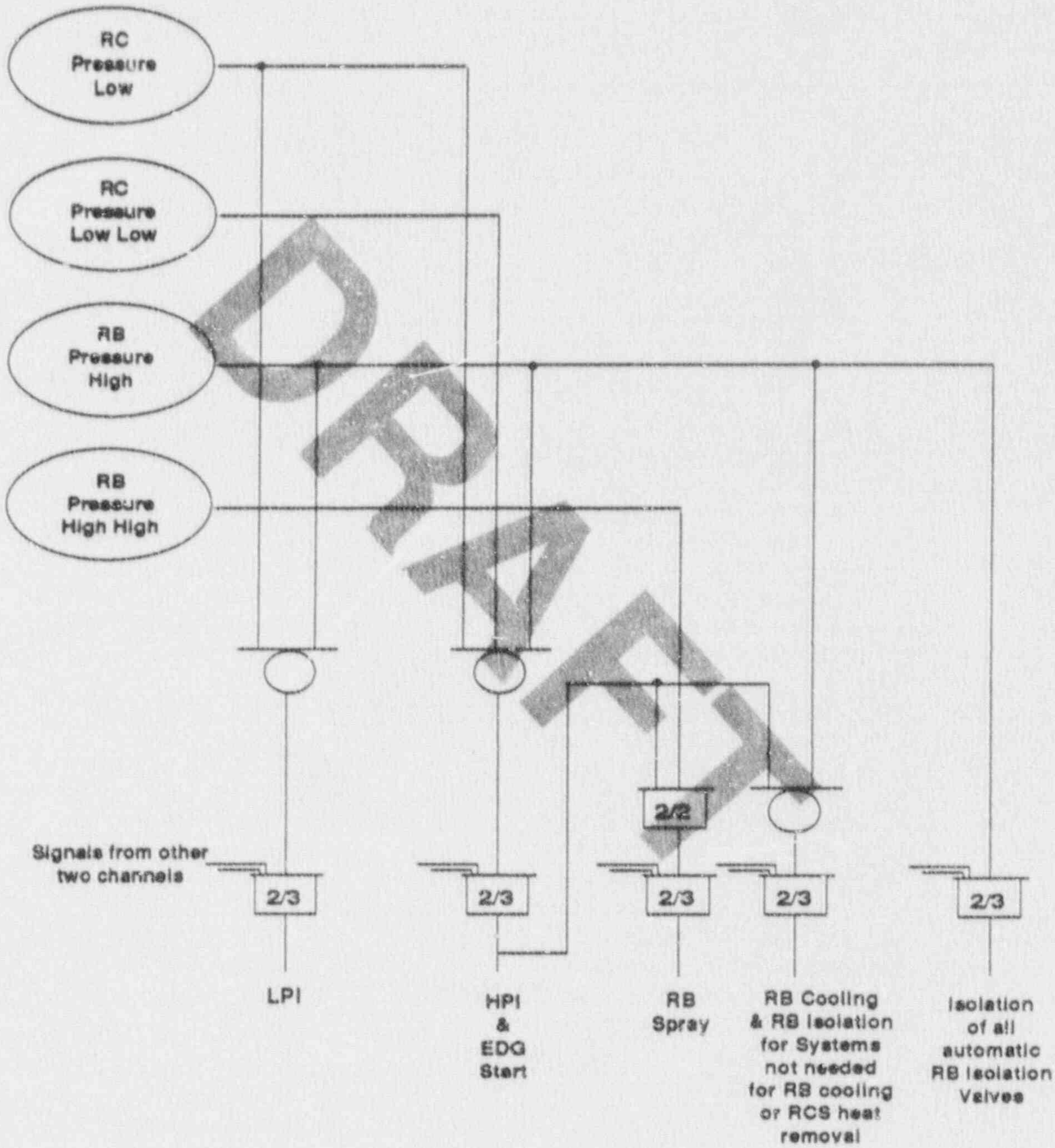

Figure B 3.3.5-1
Simplified Engineered Safety Feature Actuation System Logic

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BASES (continued)

BACKGROUND (continued)

Function	Low RCS Pressure	Low Low RCS Pressure	High RB Pressure	High High RB Pressure
HPI	X	X	X	
LPI		X	X	
RB Cooling	X	X	X	
RB Spray	**	**	**	**
RB Isolation*	X	X	X	
EDG Start	X	X	X	
Control Room Isolation	X	X	X	

* Only isolates systems not required for RB or RCS heat removal.

** Actuates on HPI initiation coincident with RB high-high pressure.

[Engineered safeguards bus undervoltage will also sequence on the HPI loads started by the HPI block timers. However, HPI will not occur unless the ESFAS HPI Signal is also present.] LCO 3.3.8, "Emergency Diesel Generator (EDG) Loss of Power Start (LOPS)," contains the requirements for the undervoltage channels.

The ESF equipment is divided between the two redundant actuation trains A and B. The division of the equipment between the two actuation trains is based on the equipment redundancy and function and is accomplished in such a manner that the failure of one of the actuation channels and the related safeguards equipment will not inhibit the overall ESF functions. Where a motor-operated or a solenoid-operated valve is driven by either of two matrices, one is from actuation channel A and one from actuation channel B. Redundant ESF pumps are controlled from separate and independent actuation channels.

The actuation of ESF equipment is also available by manual actuation switches located on the control room console.

The ESFAS, in conjunction with the actuated equipment, provides protective functions necessary to mitigate Design Basis Accidents, specifically the loss-of-coolant accident (LOCA) and steam line break (SLB) events. The ESFAS relies

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BASES (continued)

BACKGROUND (continued)

on the OPERABILITY of the automatic actuation logic for each component to perform the actuation of the selected systems of LCO 3.3.7, "Engineered Safety Feature Actuation System (ESFAS) Automatic Actuation Logic."

Engineered Safety Feature Actuation System Bypasses

No provisions are made for maintenance bypass of ESFAS instrumentation channels. Operational bypass of certain channels is necessary to allow accident recovery actions to continue and, for some channels, to allow reactor shutdown without spurious ESFAS actuation.

The ESFAS RCS pressure instrumentation channels include permissive bistables that allow manual bypass when reactor pressure is below the point at which the low and low low pressure trips are required to be OPERABLE. Once permissive conditions are sensed, the RCS pressure trips may be manually bypassed. Bypasses are automatically removed when bypass permissive conditions are exceeded.

Each RB high pressure channel may be manually bypassed after the other two channels in the function have tripped. The manual bypass allows operators to take manual control of ESF functions after initiation to allow recovery actions. The bypass may be manually removed and is automatically removed when RB pressure returns to below the trip point.

Engineered Safety Feature Actuation System Instrumentation

Reactor Coolant System Pressure

The RCS pressure is monitored by three independent pressure transmitters located in the RB. These transmitters are separate from the transmitters that feed the Reactor Protection System (RPS). Each of the pressure signals generated by these transmitters is monitored by four bistables to provide two trip signals, at [1500 psig] and at [500 psig], and two bypass permissive signals at [1700 psig] and at [900 psig].

The outputs of the three bistables associated with the low RCS pressure [1500 psig] trip drive relays in two sets (actuations A and B) of identica' and independent channels.

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BASES (continued)

BACKGROUND (continued)

These two sets of HPI channels each employ three logic channels used in two-out-of-three coincidence networks for HPI actuation. The outputs of the three bistables associated with the low low RCS pressure [500 psig] trip drive relays in two sets (actuation A and B) of identical and independent channels. These two sets of LPI channels each employ three logic channels used in two-out-of-three coincidence networks for LPI actuation. The outputs of the three low low RCS pressure bistables also trip the drive relays in the corresponding HPI actuation channel as described above.

Reactor Building Pressure

RB pressure inputs to the ESFAS are provided by 12 pressure switches. Six pressure switches are used for the RB high pressure function and six pressure switches are used for the reactor building high high pressure function. Inputs to the RPS are provided by four additional transmitters.

The output contacts of six RB high pressure switches are used in two sets of identical and independent actuation trains. These two trains each employ three logic channels. The outputs of these channels are used in two-out-of-three coincidence networks. The output contacts of the six RB pressure switches also trip the drive relays in the corresponding HPI and LPI actuation channels as described above.

The output contacts of six RB high high pressure switches are used in two sets of identical and independent actuation trains. These two trains each employ three logic channels (RB4, RB5, and RB6). The outputs of these channels are used in two-out-of-three coincident networks for RB spray actuation. Each high high pressure train actuates one RB spray train when the RB high high signal and the HPI signal are coincident in that train.

Trip Setpoints and ALLOWABLE VALUES

Trip setpoints are the nominal value at which the bistables are set. Any bistable is considered to be properly adjusted when the "as left" value is within the band for CHANNEL CALIBRATION accuracy, i.e., ± (rack calibration and comparator setting accuracy).

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BASES (continued)

BACKGROUND (continued)

The trip setpoints used in the bistables are based on the analytical limits stated in Reference 4. The selection of these trip setpoints is such that adequate protection is provided when all sensor and processing time delays are taken into account. These setpoints allow for calibration tolerances, instrumentation uncertainties, instrument drift, and severe environment induced errors for those ESFAS channels which must function in harsh environments as defined by 10 CFR 50.49 (Ref. 1). ALLOWABLE VALUES specified in Table 3.3.5-1 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 the plant-specific setpoint analysis (Ref. 3). The actual nominal trip setpoint entered into the bistable is more conservative than that specified by the ALLOWABLE VALUE, to account for changes in random measurement errors detectable by a CHANNEL FUNCTIONAL TEST. One example of such a change in measurement error is drift during the surveillance interval. If the measured setpoint does not exceed the ALLOWABLE VALUE, the bistable is considered OPERABLE.

The ALLOWABLE VALUES listed in Table 3.3.5-1 are based upon the methodology described in Reference 3, which incorporates all of the known uncertainties applicable for each channel. The magnitudes of these uncertainties are factored into the determination of each trip setpoint. All field sensors and signal processing equipment for these channels are assumed to operate within the allowances of these uncertainty magnitudes.

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

Each channel can be tested on line to verify that the setpoint accuracy is within the specified allowance requirements of Reference 5. Once a designated channel is taken out of service for testing, a simulated signal is injected in place of the field instrument signal. The process equipment for the channel in test is then tested, verified, and calibrated.

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BASES (continued)

BACKGROUND (continued)

The ESFAS LCOs in the BWOG Standard Technical Specifications (STS) are based upon a system representative of the Crystal River Unit 3 design. As discussed above, this arrangement involves measurement channels shared among all actuation functions, with separate actuation logic channels for each actuated component. In this arrangement, multiple components are affected by each instrumentation channel failure, but a single automatic actuation logic failure affects only one component. The organization of BWOG STS ESFAS LCOs reflect the described logic arrangement by identifying instrumentation requirements on an instrumentation channel rather than protective function basis. This greatly simplifies delineation of ESFAS LCOs. Furthermore, the LCO requirements on instrumentation channels, automatic actuation logics, and manual initiation are specified separately to reflect the significant different impact each has upon ESFAS OPERABILITY.

The Crystal River Unit 3 ESFAS arrangement is not standard for B&W plants. Indeed, no ESFAS design is typical of the B&W plants. Therefore, for some B&W plants the WOG STS or the CEOG STS may provide a more meaningful starting point for the development of plant-specific Technical Specifications.

APPLICABLE SAFETY ANALYSES

The following ESFASs have been assumed to function within the accident analyses.

High Pressure Injection

The ESFAS actuation of HPI has been assumed for core cooling in the LOCA analysis and is credited with boron addition in the SLB analysis.

Low Pressure Injection

The ESFAS actuation of LPI has been assumed for large-break LOCAs.

Reactor Building Spray, Reactor Building Cooling, and Reactor Building Isolation

The ESFAS actuation of the RB coolers and RB spray have been credited in RB analysis for LOCAs, both for RB performance

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

as well as for equipment environmental qualification pressure-temperature envelope definition. Accident dose calculations have credited RB isolation and RB spray.

Emergency Diesel Generator Start

The ESFAS initiated EDG start has been assumed in the LOCA analysis to ensure that emergency power is available throughout the limiting LOCA scenarios.

The small- and large-break LOCA analyses assume a conservative 35-second delay time for the actuation of HPI and LPI (Ref. 6). This delay time includes allowances for EDG starting, EDG loading, ECCS pump starts, and valve openings. Similarly, the RB cooling, RB isolation, and RB spray have been analyzed with delays appropriate for the entire system analyzed. Typical values used in the analysis are 35 seconds for RB cooling, 60 seconds for RB isolation, and 58 seconds for RB spray.

Control Room Isolation

[For this facility the applicable safety analyses for the ESFAS control room isolation function are as follows:]

Accident analyses rely on automatic ESFAS actuation for protection of the core temperature and containment pressure limits, and for limiting off-site dose levels following an accident. These include LOCA, SLB, and feedwater line break events which result in RCS inventory reduction or severe loss of RCS cooling.

The ESFAS channels satisfy Criterion 3 of the NRC Interim Policy Statement.

LCO

The LCO requires all channels of instrumentation for ESFAS functions to be OPERABLE. Failure of any instrument renders the affected channel(s) inoperable and reduces the reliability of the affected functions. ESFAS instrumentation channels are considered OPERABLE when:

- All channel components necessary to provide an ESFAS actuation signal are functional and in service;

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BASES (continued)

LCO (continued)

- Channel measurement uncertainties are known (via test, analysis, or design information) to be within the assumptions of the setpoint calculations;
- Required surveillance testing is current and has demonstrated performance within each surveillance test's acceptance criteria; and
- The associated operational bypass is not enabled except under the conditions specified by the LCO Applicability statement for the function.

The LCO requires all channels to be OPERABLE to ensure system reliability, testability, and redundancy.

Only the ALLOWABLE VALUE is specified for each ESFAS function in the LCO. Nominal trip setpoints are specified in the plant-specific setpoint calculations. The nominal trip setpoints are selected to ensure the setpoint measured by CHANNEL FUNCTIONAL TESTS do not exceed the ALLOWABLE VALUE if the bistable 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 is consistent with the assumptions of the plant-specific setpoint calculations. Each ALLOWABLE VALUE specified is more conservative than the analytical limit assumed in the safety analysis in order to account for instrument uncertainties appropriate to the trip function. These uncertainties are defined in the plant-specific setpoint methodology (Ref. 3).

The ALLOWABLE VALUES for bypass removal functions are stated in the Application column of Table 3.3.5-1.

Three ESFAS instrumentation channels shall be OPERABLE in each ESFAS train to ensure that a single failure in one channel will not result in loss of the ability to automatically actuate the required safety systems.

The bases for the LCO on ESFAS functions are:

<u>Reactor Coolant System Pressure</u>

Three channels each of RCS Pressure--Low and RCS Pressure--Low Low are required OPERABLE in each train. Each channel

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BASES (continued)

LCO (continued)

includes a sensor, trip bistable, bypass bistable, bypass relays, output relays, and block timers. The analog portion of each pressure channel is common to both trains of both RCS pressure functions. Therefore, failure of one analog channel renders one channel of the low pressure and low low pressure functions in each train inoperable. The bistable portions of the channels are function and train specific. Therefore, a bistable failure renders only one function in one train inoperable. Failure of a bypass bistable or bypass circuitry such that a trip channel cannot be bypassed does not render the channel inoperable. Output relays and block timer relays are train specific, but may be shared among functions. Therefore, output or block timer relay failure renders all affected functions in one train inoperable.

1. Reactor Coolant System Pressure--Low Setpoint

The RCS Pressure--Low setpoint is based on HPI actuation for small-break LOCAs. The setpoint ensures that the HPI will be actuated at a pressure greater than or equal to the value assumed in accident analyses plus the instrument uncertainties. The maximum value assumed for the setpoint of the RCS low pressure trip of HPI in safety analyses is [1480] psig. The setpoint for the low RCS and ALLOWABLE VALUE of [1600] psig for the low pressure function is selected to ensure actuation occurs when actual RCS pressure is above [1480] psig. The RCS pressure instrumentation must function while subject to the severe environment created by a LOCA. Therefore, the trip setpoint ALLOWABLE VALUE must account for severe environment-induced errors.

To ensure the RCS Pressure--Low trip is not bypassed when required to be OPERABLE by the safety analysis, each channel's bypass removal bistable must be set with an ALLOWABLE VALUE of ≤ [1800] psig. The bypass removal does not need to function for accidents initiated from RCS pressures below the bypass removal setpoint. Therefore, the bypass removal setpoint ALLOWABLE VALUE need not account for severe environment-induced errors.

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BASES (continued)

LCO (continued)

2. Reactor Coolant System Pressure--Low Low Setpoint

The RCS Pressure--Low Low setpoint LPI actuation occurs in sufficient time to ensure LPI flow prior to the emptying of the core flood tanks during a large-break LOCA. The ALLOWABLE VALUE of ≥ [400] psig ensures sufficient overlap of the core flood tank flow and the LPI flow to keep the reactor vessel downcomer full during a large-break LOCA. The RCS pressure instrumentation must function while subject to the severe environment created by a LOCA. Therefore, the trip setpoint ALLOWABLE VALUE must account for severe environment-induced errors.

To ensure the RCS Pressure--Low Low trip is not bypassed when assumed OPERABLE by the safety analysis, each channel's bypass removal bistable must be set with an ALLOWABLE VALUE of ≤ [900] psig. The bypass removal does not need to function for accidents initiated by RCS pressure below the bypass removal setpoint. Therefore, the bypass removal setpoint ALLOWABLE VALUE need not account for severe environment-induced errors.

Reactor Building Pressure

Three channels each of RCS Pressure--Low and RB Pressure--High are required to be OPERABLE in each train. Each channel includes a pressure switch, bypass relays, and output relays. The high pressure channels also include block timers. Each pressure switch is function and train specific, so there are 12 pressure switches total. Therefore, a pressure switch renders only one function in one train inoperable. Output relays and block timer relays are train specific, but may be shared among functions. Therefore, output or block timer relay failure renders all affected functions in one train inoperable.

The RB pressure switches may be subjected to high radiation conditions during the accidents which they are intended to mitigate. The sensor portion of the switches are also exposed to the steam environment present in the RB following

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BASES (continued)

LCO (continued)

a LOCA or high energy line break. Therefore, the trip setpoint ALLOWABLE VALUE must account for measurement errors induced by these environments.

3. Reactor Building Pressure--High Setpoint

The RB Pressure--High setpoint ALLOWABLE VALUE ≤ [5] psig was selected to be low enough to detect a rise in RB pressure that would occur due to a small-break LOCA, thus ensuring that the RB high pressure actuation of the safety systems will occur for a wide spectrum of break sizes. The trip setpoint also causes the RB coolers to shift to emergency mode to prevent damage to the cooler fans due to the increase in the density of the air-steam mixture present in the containment following a LOCA.

4. Reactor Building Pressure--High High Setpoint

The ≤ [30] psig RB Pressure--High High setpoint ALLOWABLE VALUE was chosen to be high enough to avoid actuation during an SLB, but also low enough to ensure a timely actuation during a large-break LOCA.

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

[For this facility, those required support systems which upon their failure do not result in the ESFAS instrumentation being declared inoperable and their justification are as follows:]

[For this facility, the supported systems impacted by the inoperability of the ESFAS instrumentation and the justification of whether or not each supported system is declared inoperable are as follows:]

APPLICABILITY

Three channels of ESFAS instrumentation for each function listed below shall be OPERABLE in each ESFAS train.

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BASES (continued)

APPLICABILITY (continued)

1. Reactor Coolant System Pressure--Low Setpoint

[At this facility, the low RCS pressure actuation function shall be OPERABLE during operation above [1800] psig for the following reasons:] Below [1800] psig the function may be bypassed to avoid actuation during normal plant cooldowns.

[At this facility, the low RCS pressure function is not necessary to plant safety when RCS pressure < [1800] psig for the following reasons:]

In MODES 5 and 6, there is adequate time for the operator to evaluate plant conditions and respond by manually starting individual systems, pumps, and other equipment to mitigate the consequences of an abnormal condition or accident. Plant pressure and temperature are very low and many ESF components are administratively locked out or otherwise prevented from actuating to prevent inadvertent overpressurization of plant systems.

2. Reactor Coolant System Pressure--Low Low Setpoint

[At this facility, the low low RCS pressure actuation function shall be OPERABLE during operation above [900] psig for the following reasons:] Below [900] psig the ESFAS function shall be bypassed to avoid LPI initiation during normal plant cooldowns. Or, [At this facility, the low low RCS pressure function is not necessary to plant safety when RCS pressure < [900] psig for the following reasons:]

In MODES 5 and 6, there is adequate time for the operator to evaluate plant conditions and respond by manually starting individual systems, pumps, and other equipment to mitigate the consequences of an abnormal condition or accident. Plant pressure and temperature are very low and many ESF components are administratively locked out or otherwise prevented from actuating to prevent inadvertent overpressurization of plant systems.

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BASES (continued)

APPLICABILITY (continued)

3, 4. Reactor Building Pressure--High and Reactor Building Pressure--High High Setpoints

The RB high and high high pressure actuation functions of ESFAS shall be OPERABLE in MODES 1, 2, 3, and 4 when the potential for a high energy line break exists. In MODES 5 and 6, the plant conditions are such that there is insufficient energy in the primary and secondary systems to raise the containment pressure to either the RB high or RB high high pressure setpoints. Furthermore, in MODES 5 and 6, there is adequate time for the operator to evaluate plant conditions and respond by manually starting individual systems, pumps, and other equipment to mitigate the consequences of an abnormal condition or accident. Plant pressure and temperature are very low and many ESF components are administratively locked out or otherwise prevented from actuating to prevent inadvertent overpressurization of plant systems.

A Note has been added in the Applicability to provide clarification that for this LCO, each function specified in Table 3.3.5-1 shall be treated as an independent entity with an independent Completion Time.

ACTIONS

Required Action A, Required Action B, and Required Action C apply to all ESFAS instrumentation functions listed in Table 3.3.5-1.

ESFAS is inoperable when it does not satisfy the OPERABILITY criteria for the channel's function. These criteria are outlined for each function in the LCO section of the Bases. The most common cause of channel inoperability is outright failure or drift of the bistable or process module sufficient to exceed the tolerance allowed by the plant-specific setpoint 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 setpoint drift is generally made during the performance of a CHANNEL FUNCTIONAL TEST when the process instrument is set up for adjustment to bring it within specification. If the trip setpoint is less conservative than the ALLOWABLE VALUE in Table 3.3.5-1, the channel shall be declared inoperable

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BASES (continued)

ACTIONS (continued)

immediately, and the appropriate Conditions from Table 3.3.5-1 shall be entered immediately.

In the event a channel's trip setpoint is found non-conservative with respect to the ALLOWABLE VALUE, or the transmitter, instrument loop, signal processing electronics, or ESFAS bistable is found inoperable, then all affected functions provided by that channel should be declared inoperable and the plant must enter the Conditions for the particular protection function affected.

When the number of inoperable channels in a trip function exceed those specified, then the plant is outside the safety analysis. Therefore, LCO 3.0.3 shall be immediately entered if applicable in the current MODE of operation.

Conditions A, B, and C

These conditions are applicable to all ESFAS protection functions.

Condition A

Required Action A.1 applies if one channel becomes inoperable. If one ESFAS channel is inoperable, placing it in a tripped condition leaves the system in a one-out-of-two condition for actuation. Thus, if another channel were to fail, the ESFAS instrumentation could still perform its actuation functions. This action is completed when all of the affected output relays and block timers are tripped. This can normally be accomplished by tripping the affected bistables or tripping the individual output relays and block timers. [At this facility, the specific output relays associated with each ESFAS instrumentation channel are listed in the following document:]

The 1-hour Completion Time is sufficient time to perform the Required Action.

Condition B

Required Action B.1 and Required Action B.2 apply if the Required Action A.1 is not met within the required Completion Time. The plant shall be placed in a MODE in which the LCO does not apply. This is done by placing the

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BASES (continued)

ACTIONS (continued)

plant in at least MODE 3 within 6 hours and in MODE 5 within 36 hours. The 6 hours and 36 hours are reasonable times, based on operating experience, to reach MODE 3 and MODE 5, respectively, from full power in an orderly manner and without challenging plant systems.

Condition C

Condition C is applicable to each one of the ESFAS functions presented in Table 3.3.5-1.

Required Action C.1 verifies that the Required Actions have been initiated for those supported systems declared inoperable because of the inoperability of the support channel(s) within a Completion Time of 1 hour. The specified Completion Time is sufficient for plant operations personnel to make this determination.

Required Action C.1 ensures that those identified Required Actions associated with supported systems impacted by the inoperability of channel(s) associated with each ESFAS function have been initiated. This can be accomplished by entering the supported systems' LCOs independently or as a group of Required Actions that need to be initiated every time Condition C is entered. [For this facility, the identified supported systems' Required Actions associated with each ESFAS function are as follows:]

Required Action C.2 verifies that all required support or supported features associated with the other redundant channel(s) are OPERABLE within a Completion Time of 1 hour. The specified Completion Time is sufficient for plant operations personnel to make this determination. If verification determines loss of functional capability, LCO 3.0.3 shall be immediately entered. However, if the support or supported feature LCO takes into consideration the loss of function situation, then LCO 3.0.3 may not need to be entered.

SURVEILLANCE REQUIREMENTS

All ESFAS functions listed in Table 3.3.5-1 are subject to CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, CHANNEL CALIBRATION, and response time testing. The operational bypasses associated with each ESFAS instrumentation channel are also

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

subject to these SRs to ensure OPERABILITY of the ESFAS instrumentation channel.

SR 3.3.5.1

SR 3.3.5.1 is the performance of a CHANNEL CHECK. Performance of the CHANNEL CHECK every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or something more serious. CHANNEL CHECK will detect gross channel failure; thus, it is the key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff, based on a combination of the channel instrument uncertainties, including isolation, indication, and readability. If a channel is outside the match criteria, it may be an indication that the transmitter or the signal-processing equipment has drifted outside its limit. If the channels are within the match criteria, it is an indication that the channels are OPERABLE. If the channels are normally off scale during times when surveillance 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 downscale.

The Surveillance Frequency, about once every shift, is based on operating experience that demonstrates the rarity of channel failure. Thus, performance of the CHANNEL CHECK ensures that undetected overt channel failure is limited to 12 hours. 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 protective function due to failure of redundant channels. 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.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.5.2

SR 3.3.5.2 is the performance of a CHANNEL FUNCTIONAL TEST every 31 days. A CHANNEL FUNCTIONAL TEST verifies the function of the trip, interlock, and alarm functions of the channel. The test inserts a simulated or actual signal as close to the sensor as practicable and verifies required trip, bypass removal interlocks, and alarms function when the input is beyond the trip point. Where the design has made provisions for including sensors in the CHANNEL FUNCTIONAL TEST, the test signal shall be inserted at that point. "As found" and "as left" values for bistable trip setpoints are recorded. Bistable setpoints shall be found within the ALLOWABLE VALUE specified in the LCO. (Note that for bypass removal functions the ALLOWABLE VALUES are given in terms of limits on the associated trip function Applicability in Table 3.3.5-1.) The difference between the current "as found" and the previous "as left" setpoints shall be within the drift allowance used in the setpoint analysis. Recalibration of the bistable setpoint restores the OPERABILITY of an otherwise functional component that does not meet these criteria. However, repeated failures of the same channel over a small number of test intervals should be evaluated as potentially indicating a deterministic failure which cannot be corrected by recalibration.

The Surveillance Frequency of 31 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 31-day interval is a rare event.

SR 3.3.5.3

SR 3.3.5.3 is the performance of a CHANNEL CALIBRATION every [18] months. CHANNEL CALIBRATION is a complete check of the instrument channel, including the detector. The test verifies that the channel responds to a measured parameter with the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift between successive tests, to ensure that the instrument channel remains operational with the setpoint within the assumptions of the plant-specific setpoint analysis. Transmitter "as found" and "as left" values are recorded and

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

used to verify drift assumptions. For OPERABLE channels, CHANNEL CALIBRATION shall find that measurement errors and bistable setpoint errors are within the assumptions of the plant-specific setpoint analysis. Measurement and setpoint error determination and readjustment shall be performed consistent with the assumptions of the plant-specific setpoint analysis.

Recalibration restores OPERABILITY of an otherwise functional component found to have errors larger than assumed by the setpoint analysis. However, repeated failures of the same channel over relatively small number of test intervals shall be considered as potentially indicating a deterministic failure that cannot be corrected by recalibration.

Field transmitters may be calibrated in place, removed and calibrated in a laboratory, or replaced with an equivalent, laboratory-calibrated unit.

This Surveillance Frequency is based upon the assumption of an [18]-month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

SR 3.3.5.4

SR 3.3.5.4 ensures that the ESFAS actuation channel response times are verified on a STAGGERED TEST BASIS. The response time values are the maximum values assumed in the safety analyses. Individual component response times are not modeled in the analyses. The analyses model the overall or total elapsed time, from the point at which the parameter exceeds the actuation setpoint value at the sensor, to the point at which the end device is actuated. Thus, this SR encompasses the automatic actuation logic components covered by LCO 3.3.7, "Engineered Safety Feature Actuation System (ESFAS) Automatic Actuation Logic," and the operation of the mechanical ESF components. This Surveillance also encompasses the EFW initiation function of HPI.

Each function's response shall be verified for one channel in each train every [18] months on a STAGGERED TEST BASIS (i.e., channel A at [18] months after initial startup, channel B at [36] months, channel C at [54] months, and then channel A again). Thus, for ESFAS the maximum interval

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

between response time testing of the same channel is [54] months. The test may be performed in one measurement or in overlapping segments with verification that all components are measured. Testing 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 [18] months. The [18]-month 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. Response times cannot be determined at power since equipment operation is required.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants."

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

3. [Unit Name], "[Plant-Specific Setpoint Methodology]."

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

5. [Unit Name] FSAR, Section [7], "[Instrumentation and Control]."

6. BAW-10103A, Rev. 3, "ECCS Analysis of B&W's 177-FA Lowered Loops NSS," July 1977.

B 3.3 INSTRUMENTATION

B 3.3.6 Engineered Safety Feature Actuation System (ESFAS) Manual Initiation

BASES

BACKGROUND

The ESFAS Manual Initiation capability allows the operator to actuate ESFAS functions from the main control room in the absence of any other initiation condition. Manually actuated functions include High Pressure Injection, Low Pressure Injection, Reactor Building (RB) Spray, Reactor Building Cooling, and Reactor Building Isolation. This function is provided in the event the operator determines that an ESFAS function is needed and has not been automatically actuated. Furthermore, the Manual Initiation function allows operators to rapidly initiate engineered safety features (ESF) functions if the trend of plant parameters indicates that ESF actuation will be needed.

This LCO covers only the system level manual initiation of these functions. LCO 3.3.5, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," and LCO 3.3.7, "Engineered Safety Feature Actuation System (ESFAS) Automatic Actuation Logic," provide requirements on the portions of the ESFAS that automatically initiate the above functions.

The ESFAS Manual Initiation function relies on the OPERABILITY of the automatic actuation logic (LCO 3.3.7, "Engineered Safety Feature Actuation System (ESFAS) Automatic Actuation Logic") for each component to perform the actuation of the systems. A manual trip push button is provided on the ESF panel of the control room console for each of the levels of protection for each actuation. Operation of the push button energizes relays whose contacts perform a logical "OR" function with the matrices of the automatic actuation, except for the matrices which are part of the ESF buses loading sequence. Manual actuation of the ESF buses loading sequence is made by de-energizing the timed output relays. The power supply for the manual trip relays is taken from the station batteries. Different batteries are used for the two actuations.

The Manual Initiation channel is defined as the instrumentation between the console switch and the automatic

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BASES (continued)

BACKGROUND (continued)

actuation logic, which actuates the end devices. Other means of manual initiation, such as controls for individual ESF devices, may be available in the control room and other plant locations. These alternative means are not required by this LCO, nor may they be credited to fulfill the requirements of this LCO.

APPLICABLE SAFETY ANALYSES

The ESFAS, in conjunction with the actuated equipment, provides protective functions necessary to mitigate Design Basis Accidents, specifically the loss-of-coolant accident and steam line break events.

The ESFAS Manual Initiation ensures that the control room operator can rapidly initiate ESF functions at any time. The Manual Initiation trip function is required as a backup to automatic trip functions and to allow operators to initiate ESFAS whenever any parameter is rapidly trending toward its trip setpoint. Furthermore, the Manual Initiation may be specified in operating procedures for verification that ESF systems are running.

The ESFAS Manual Initiation functions satisfy Criterion 3 of the NRC Interim Policy Statement.

LCO

Two Manual Initiation channels of each ESFAS function shall be OPERABLE whenever conditions exist that could require ESF protection of the reactor or RB. Two OPERABLE channels ensure that no single random failure will prevent system level Manual Initiation of any ESFAS function. The Manual Initiation function allows the operator to initiate protective action prior to automatic initiation or in the event the automatic initiation does not occur.

The ESFAS Manual Initiation channels are considered OPERABLE when:

a. All channel components necessary to provide an ESFAS actuation are functional and in service; and

b. Required surveillance testing is current and has demonstrated performance within each surveillance test's acceptance criteria.

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BASES (continued)

LCO (continued)

[For this facility, the following support features systems are required to be OPERABLE to ensure ESFAS Manual Initiation OPERABILITY:]

[For this facility, those required feature support systems which upon their failure do not result in the ESFAS Manual Initiation circuitry being declared inoperable and their justification are as follows:]

[For this facility, the supported features systems impacted by the inoperability of ESFAS Manual Initiation circuitry and the justification of whether or not each supported system is declared inoperable are as follows:]

APPLICABILITY

The ESFAS Manual Initiation functions shall be OPERABLE in MODES 1, 2, and 3, and in MODE 4 when the associated engineered safeguard equipment is required to be OPERABLE. The Manual Initiation channels are required because ESF functions are designed to provide protection in these MODES. In MODES 5 and 6, the systems initiated by ESFAS are either reconfigured or disabled for shutdown cooling operation. Accidents in these MODES are slow to develop. Adequate time is available to evaluate plant conditions and to respond by manually operating the ESF components if required. Accidents in these MODES are slow to develop and would be mitigated by manual operation of individual components.

A Note has been added in Applicability to provide clarification that for this LCO, each function specified shall be treated as an independent entity with an independent Completion Time.

ACTIONS

A Manual Initiation channel is inoperable when it does not satisfy the OPERABILITY criteria for the channel's function. These criteria are outlined in the LCO section of the Bases.

When both ESFAS Manual Initiation channels are inoperable in any function(s), facility operation is not allowed to

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BASES (continued)

ACTIONS (continued)

continue in this degraded condition. Therefore, LCO 3.0.3 must be immediately entered, if applicable in the current MODE of operation.

Condition A

Condition A applies when one Manual Initiation channel of one or more ESFAS functions becomes inoperable. Required Action A.1 must be taken to restore the channel to OPERABLE status within the next [72] hours. The Completion Time of [72] hours is based on plant operating experience and administrative controls, which provide alternative means of ESFAS function initiation via individual component controls. The [72]-hour Completion Time is consistent with the allowed outage time for the safety systems actuated by ESFAS.

Condition B

Required Action B.1 and Required Action B.2 apply if the Required Action A.1 cannot be met within the required Completion Time. The plant must be placed in a MODE in which the requirement does not apply. This is done by placing the plant in at least MODE 3 within 6 hours and in MODE 5 within 36 hours. The Completion Times are reasonable, based on operating experience, to reach the required MODES from full power in an orderly manner and without challenging plant systems.

Condition C

Condition C applies when one ESFAS Manual Initiation channel of one or more ESFAS functions becomes inoperable. Required Action C.1 verifies that the Required Actions have been initiated for those supported systems declared inoperable because of the inoperability of the support channel(s) within a Completion Time of 1 hour. The specified Completion Time is sufficient for plant operations personnel to make this determination.

Required Action C.1 ensures that those identified Required Actions associated with supported systems impacted by the inoperability of channel(s) associated with each ESFAS Manual Initiation function have been initiated. This can be accomplished by entering the supported systems' LCOs or

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BASES (continued)

ACTIONS (continued)

independently as a group of Required Actions needed to be initiated every time Condition C is entered. [For this facility, the identified supported systems' Required Actions associated with each ESFAS Manual Initiation function are as follows:]

Required Action C.2 verifies that all required support or supported features associated with the other redundant channel(s) are OPERABLE within a Completion Time of 1 hour. The specified Completion Time is sufficient for plant operations personnel to make this determination. If verification determines loss of functional capability, LCO 3.0.3 must be immediately entered. However, if the support or supported feature system LCO takes into consideration the loss of function situation, then LCO 3.0.3 may not need to be entered.

SURVEILLANCE REQUIREMENTS

SR 3.3.6.1

SR 3.3.6.1 is the performance of a CHANNEL FUNCTIONAL TEST every [18] months. This test verifies that the initiating circuitry is OPERABLE and will actuate the end device (i.e., pump, valves, etc.). The test also includes trip devices that would normally actuate the end device. The [18]-month Frequency was developed considering it was prudent that these surveillances only be performed during a plant outage. This was due to the plant conditions needed to perform the surveillance and the potential for unplanned plant transients if the surveillance is performed with the reactor at power. Operating experience has shown these components usually pass the surveillance when performed on the [18]-month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

None.

B 3.3 INSTRUMENTATION

B 3.3.7 Engineered Safety Feature Actuation System (ESFAS) Automatic Actuation Logic

BASES

BACKGROUND

The automatic actuation logic channels of ESFAS are defined as the logic between the buffers of the sensing channels and the controllers which actuate ESFAS equipment. Each of the components actuated by the ESFAS functions has an associated automatic actuation logic channel. If two-out-of-three ESFAS instrumentation channels indicate a trip, or system level Manual Initiation occurs, the automatic actuation logic is activated and the associated component is actuated. The purpose of requiring OPERABILITY of the ESFAS automatic actuation logic is to ensure that the functions of ESFAS can be automatically initiated in the event of an accident. Automatic actuation of some functions is necessary to prevent the plant from exceeding the Emergency Core Cooling Systems (ECCS) limits (Ref. 1). It should be noted that OPERABLE automatic actuation logic channels alone will not ensure that each function can be activated; the instrumentation channels and actuated equipment associated with each function must also be OPERABLE to ensure that the functions can be automatically initiated during an accident.

This LCO covers only the automatic actuation logic that initiates these functions. LCO 3.3.5, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," and LCO 3.3.6, "Engineered Safety Feature Actuation System (ESFAS) Manual Initiation," provide requirements on the instrumentation and manual initiation channels that input to the automatic actuation logic.

The ESFAS, in conjunction with the actuated equipment, provides protective functions necessary to mitigate Design Basis Accidents, specifically the loss-of-coolant accident (LOCA) and steam line break (SLB) events. The ESFAS relies on the operability of the automatic actuation logic for each component to perform the actuation of the selected systems.

The small- and large-break LOCA analyses assume a conservative 35-second delay time for the actuation of high pressure injection (HPI) and low pressure injection (LPI)

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BASES (continued)

BACKGROUND (continued)

(Ref. 2). This delay time includes allowances for emergency diesel generator (EDG) starts, EDG loading, ECCS pump starts, and valve openings. Similarly, the reactor building (RB) cooling, reactor building isolation, and reactor building spray have been analyzed with delays appropriate for the entire system.

Typical values used in the analysis are [35] seconds for reactor building cooling, [60] seconds for reactor building isolation, and [58] seconds for reactor building spray.

The ESFAS automatic initiation of engineered safety features (ESF) functions to mitigate accident conditions is assumed in the DBA analysis and is required to ensure that consequences of analyzed events do not exceed the accident analysis predictions. Automatically actuated features include HPI, LPI, reactor building cooling, reactor building spray, and reactor building isolation.

The ESFAS LCOs in the BWOG Standard Technical Specifications (STS) are based on a system representative of the Crystal River Unit 3 design. As discussed above, this arrangement involves measurement channels shared among all actuation functions, with separate actuation logic channels for each actuated component. In this arrangement, multiple ESF components are affected by a measurement channel failure, but a single automatic actuation logic failure affects only one component. The organization of BWOG STS ESFAS LCOs reflect the described logic arrangement by linking actions for automatic actuation logic failures directly to the actions for the affected ESF component. The overall philosophy is that if an automatic actuation logic fails, the affected component is put into its engineered safeguard configuration. This action eliminates the need for the automatic actuation logic. If the affected component cannot be placed in its engineered safeguard configuration, actions are taken to address the inoperability of the supported system component. This greatly simplifies delineation of ESFAS LCOs. Furthermore, the LCO requirements on instrumentation channels, automatic actuation logics, and manual initiation are specified separately to reflect the significant different impact each has upon ESFAS operability.

The Crystal River Unit 3 ESFAS arrangement is not standard for B&W plants. Indeed, ESFAS designs vary significantly

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BASES (continued)

BACKGROUND (continued)

among the B&W plants. Therefore, for some B&W plants the WOG STS or the CEOG STS may provide a more meaningful starting point for the development of plant-specific Technical Specifications.

APPLICABLE SAFETY ANALYSES

Accident analyses rely on automatic ESFAS actuation for protection of the core temperature and reactor building pressure limits, and for limiting off-site dose levels following an accident. These include LOCA, SLB, and feedwater line break, events which result in Reactor Coolant System (RCS) inventory reduction or severe loss of RCS cooling. The automatic actuation logic is an integral part of the ESFAS.

The ESFAS automatic actuation logics satisfy Criterion 3 of the NRC Interim Policy Statement.

LCO

The automatic actuation logic for each component actuated by the ESFAS is required to be OPERABLE whenever conditions exist which could require engineered safety features protection of the reactor or the reactor building. This ensures automatic initiation of the engineered safety features required to mitigate the consequences of accidents.

The automatic actuation logic is considered OPERABLE when:

a. All channel components necessary to provide an ESFAS actuation are functional and in service; and

b. Required surveillance testing is current and has demonstrated performance within each surveillance test's acceptance criteria.

[For this facility, the following support systems are required to be OPERABLE to ensure ESFAS automatic actuation logic instrumentation OPERABILITY:]

[For this facility, those required support systems which upon their failure do not result in the ESFAS automatic actuation logic instrumentation being declared inoperable and their justification are as follows:]

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BASES (continued)

APPLICABILITY

[For this facility, the supported systems impacted by the inoperability of the ESFAS automatic actuation logic instrumentation and the justification of whether or not each supported system is declared inoperable are as follows:]

Automatic actuation logic function shall be OPERABLE in MODES 1, 2, and 3, and in MODE 4 when the associated ESFAS equipment is required, because ESF functions are designed to provide protection in these MODES. Automatic actuation in MODE 5 or 6 is not required because the systems initiated by ESFAS 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. Adequate time is available to evaluate plant conditions and respond by manually operating the ESF components if required.

A Note has been added in Applicability to provide clarification that for this LCO, each function specified is treated as an independent entity with an independent Completion Time.

ACTIONS

An automatic actuation logic channel is inoperable when it does not satisfy the OPERABILITY criteria for the channel's function. These criteria are outlined for each function in the LCO section of the Bases.

Condition A

When one or more automatic actuation logic matrices are inoperable, the plant must enter Condition A. Required Action A.1 requires the automatic actuation logic be restored to OPERABLE status. This is preferred as it completely restores ESFAS OPERABILITY.

If the inoperable automatic actuation logic cannot be repaired, the associated component can be placed in its engineered safeguard configuration. Required Action A.2 is equivalent to the automatic actuation logic performing its safety function ahead of time. In some cases, placing the component in its engineered safeguard configuration would violate plant safety or operational considerations. In

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BASES (continued)

ACTIONS (continued)

these cases the component status should not be changed, but the supported system component must be declared inoperable. Conditions which would preclude the placing of a component in its engineered safeguard configuration include, but are not limited to, violation of system separation, activation of fluid systems which could lead to thermal shock, or isolation of fluid systems which are normally functioning. [At this facility, the following documentation is used to determine if the affected component may be placed in the engineered safeguard configuration and identifies the engineered safeguard configuration:] The Completion Time of 1 hour is based on operating experience and reflects the urgency associated with the inoperability of a safety system component.

Required Action A.3 requires entry into the Required Actions of the affected supported systems, since the true effect of automatic actuation logic failure is inoperability of the supported system. The Completion Time of 1 hour is based on operating experience and reflects the urgency associated with the inoperability of a safety system component.

Condition B

Condition B applies when one or more automatic actuation logic matrices becomes inoperable.

Required Action B.1 verifies that the Required Actions have been initiated for those supported systems declared inoperable because of the inoperability of the support ESFAS automatic actuation logics within a Completion Time of 1 hour. The specified Completion Time is sufficient for plant operations personnel to make this determination.

Required Action B.1 ensures that those identified Required Actions associated with supported systems impacted by the inoperability of the ESFAS automatic actuation logic have been initiated. This can be accomplished by entering the supported systems' LCOs or independently as a group of Required Actions needed to be initiated every time Condition B is entered. [For this facility, the identified supported systems' Required Actions associated with each ESFAS automatic actuation logic are as follows:]

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BASES (continued)

ACTIONS (continued)

Required Action B.2 verifies that all required support or supported features associated with the other redundant ESFAS automatic actuation logic(s) are OPERABLE within a Completion Time of 1 hour. The specified Completion Time is sufficient for plant operations personnel to make this determination. If verification determines loss of functional capability, LCO 3.0.3 must be immediately entered. However, if the support or supported feature LCO takes into consideration the loss of function situation, then LCO 3.0.3 may not need to be entered.

SURVEILLANCE REQUIREMENTS

SR 3.3.7.1

SR 3.3.7.1 is the performance of a CHANNEL FUNCTIONAL TEST on a 31-day STAGGERED TEST BASIS. The test demonstrates that every automatic actuation logic associated with one of the two safety system trains successfully perform the two-out-of-three logic combinations every 31 days. All automatic actuation logics are thus retested every 62 days. The test simulates the required one-out-of-three inputs to the logic circuit and verifies the successful operation of the automatic actuation logic. The Surveillance Frequency is based upon operating experience that demonstrates the rarity of more than one channel failing within the same 31-day interval.

Automatic actuation logic response time testing is incorporated into the response time testing required by LCO 3.3.5, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation."

REFERENCES

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

2. BAW-10103A, Rev. 3, "ECS Analysis of B&W's 177-FA Lowered Loop NSS," July, 1977.

B 3.3 INSTRUMENTATION

B 3.3.8 Emergency Diesel Generator (EDG) Loss of Power Start (LOPS)

BASES

BACKGROUND

The EDGs provide a source of emergency power when offsite power is either unavailable or is insufficiently stable to allow safe plant operation. Undervoltage protection will generate a LOPS in the event a loss of voltage or degraded voltage condition occurs in the switchyard. There are two LOPS functions for each 4.16 kV vital bus.

Three undervoltage relays with [inverse voltage time] characteristics are provided on each 4.16 kV Class 1E instrument bus for the purpose of detecting a sustained undervoltage condition or a loss of bus voltage. The relays are combined in a two-out-of-three logic to generate a LOPS if the voltage is below 75% for a short time or below 90% for a long time. The LOPS-initiated ACTIONS are described in the FSAR (Ref. 1).

Trip Setpoints and ALLOWABLE VALUE

The trip setpoints used in the bistables are based on the analytical limits presented in accident analysis (Ref. 2). 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 instrument drift, ALLOWABLE VALUES specified in SR 3.3.8.3 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 the plant-specific setpoint methodology (Ref. 5). The actual nominal trip setpoint entered into the bistable is more conservative than that required by the plant-specific setpoint calculations. If the measured setpoint does not exceed the documented surveillance test acceptance criteria, the bistable is considered OPERABLE.

Setpoints in accordance with the ALLOWABLE VALUE will assure that limits of Specification 2.0, "Safety Limits," are

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BASES (continued)

BACKGROUND (continued)

not violated during anticipated operational occurrences (AOOs) and that the consequences of accidents will be acceptable, providing the plant is operated from within the LCOs at the onset of the AOO or accident, and the equipment functions as designed.

The undervoltage protection scheme has been designed to protect the plant from spurious trips caused by the offsite power source. This is made possible by the inverse voltage time] characteristics of the relays used. A complete loss of offsite power will result in approximately a [1]-second delay in LOPS actuation. The EDG starts and is available to accept loads within a [10]-second time interval on Engineered Safety Feature Actuation System (ESFAS) or LOPS. Emergency power is established within the maximum time delay assumed for each event analyzed in the accident analysis (Ref. 2).

Since there are three protective channels in a two-out-of-three trip logic for each division of the 4.16 kV power supply, no single failure will cause or prevent protective system actuation. This arrangement meets IEEE-279 criteria (Ref. 4).

APPLICABLE SAFETY ANALYSES

The EDG LOPS is required for the engineered safety features (ESFs) 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 EDG based on the loss of offsite power during a loss-of-coolant accident (LOCA). The actual EDG 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 EDG supplied power following a loss of offsite power. The analysis assumes a nonmechanistic EDG loading, which does not explicitly account for each individual component of the loss of power detection and subsequent actions. The total actuation time for the limiting systems, high pressure injection and low

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

pressure injection, is [35] seconds. This delay time includes contributions from the EDG start, EDG loading, and safety injection system component actuation. The response of the EDG 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 LOPS, in conjunction with the ESF systems powered from the EDGs, provide plant protection in the event of any of the analyzed accidents discussed in accident analysis (Ref. 2), in which a loss of offsite power is assumed. LOPS channels are required to meet the redundancy and testability requirements of 10 CFR 50, Appendix A, GDC 21 (Ref. 3).

The delay times assumed in the safety analysis for the ESF equipment include the [10]-second EDG start delay, and the appropriate sequencing delay, if applicable. The response of the EDG to a loss of power must be demonstrated to fall within this analysis response time. The response times for ESFAS-actuated equipment in LCO 3.3.5, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," include the appropriate EDG loading and sequencing delay.

The EDG LOPS channels satisfy Criterion 3 of the NRC Interim Policy Statement.

LCO

The LCO for the LOPS requires that three channels per bus of each LOPS instrumentation function shall be OPERABLE in MODES 1, 2, 3, and 4 when the LOPS supports safety systems associated with the ESFAS. In MODES 5 and 6, the three channels must be OPERABLE whenever the associated EDG is required to be OPERABLE to ensure that the automatic start of the EDG is available when needed.

Loss of LOPS function could result in the delay of safety systems initiation when required. This could lead to the violation of the Safety Limits during certain AOOs, or unacceptable consequences during accidents. During the loss of offsite power, which is an AOO, the EDG powers the motor-driven emergency feedwater pumps. Failure of these pumps to

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BASES (continued)

LCO (continued)

start would leave only the one turbine-driven pump, as well as an increased potential for a loss of decay heat removal through the secondary system.

Only ALLOWABLE VALUES are specified for each function in the LCO. Nominal trip setpoints are specified in the plant-specific setpoint calculations. The nominal setpoints are selected to ensure that the setpoint measured by CHANNEL FUNCTIONAL TESTS does not exceed the ALLOWABLE VALUE if the bistable is performing as required. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within the ALLOWABLE VALUE, is acceptable provided that operation and testing is consistent with the assumptions of the plant-specific setpoint calculation. Each ALLOWABLE VALUE specified is more conservative than the analytical limit assumed in the transient and accident analysis in order to account for instrument uncertainties appropriate to the trip function. These uncertainties are defined in the plant-specific setpoint methodology.

[For this facility, relay configuration is as follows:]

[For this facility, the trip meets single-failure criterion for single-phasing events as follows:]

[For this facility, the time-delay setpoint is controlled as follows:]

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

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

[For this facility, those required support systems which upon their failure do not result in the EDG LOPS instrumentation being declared inoperable and their justification are as follows:]

[For this facility, the supported systems impacted by the inoperability of the EDG LOPS instrumentation and the justification of whether or not each supported system is declared inoperable are as follows:]

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BASES (continued)

APPLICABILITY

The EDG LOPS actuation function shall be OPERABLE in MODES 1, 2, 3, and 4 because ESF functions are designed to provide protection in these MODES. Actuation in MODE 5 or 6 is required whenever the required EDG shall be OPERABLE, so that it can perform its function on a loss of power or degraded power to the vital bus.

A Note has been added in Applicability to provide clarification that for this LCO, each function is treated as an independent entity with an independent Completion Time.

ACTIONS

A channel is inoperable when it does not satisfy the OPERABILITY criteria for the channel's function. These criteria are outlined in the LCO LOPS section of the Bases. The most common cause of channel inoperability is outright failure or drift of the bistable or process module sufficient to exceed the tolerance allowed by the plant-specific setpoint 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 setpoint drift is generally made during the performance of a CHANNEL FUNCTIONAL TEST when the instrument is set up for adjustment to bring it within specification. If the trip setpoint is less conservative than the ALLOWABLE VALUE, the channel shall be declared inoperable immediately and the appropriate Conditions shall be entered immediately.

In the event a channel's trip setpoint is found non-conservative with respect to the ALLOWABLE VALUE, or the channel is found inoperable, then the function that the channel provides must be declared inoperable and the LCO Condition entered for the particular protection function affected. Since the required channels are specified on a per EDG basis, the Condition may be entered separately for each EDG.

Condition A

Condition A applies if one channel is inoperable for one or more functions per EDG bus.

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BASES (continued)

ACTIONS (continued)

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

Required Action A.1 to restore channel OPERABILITY is the preferred action because it restores full functional capability of the LOPS. The 1-hour Completion Time is reasonable to evaluate and take action to correct a degraded condition in an orderly manner and takes into account the low probability of an event requiring LOPS occurring during this interval.

If the channel cannot be restored to OPERABLE status in compliance with Required Action A.1, Required Action A.2.1 requires that the channel be tripped within 1 hour. With a channel in trip, the LOPS channels are configured to provide a one-out-of-two logic to initiate a trip of the incoming offsite power. In trip, one additional valid actuation will cause a LOPS signal on the bus. The 1-hour Completion Time is justified on the same basis as for Required Action A.1.

Required Action A.2.2 requires restoring the channel prior to the next CHANNEL FUNCTIONAL TEST, a time period which could be as long as 31 days. Restoring the channel before the next CHANNEL FUNCTIONAL TEST should allow ample time to repair most failures. The LOPS is still capable of performing its design function given an additional failure.

Restoring one channel to OPERABILITY is the preferred action. The 1-hour Completion Time should allow ample time to repair most failures and takes into account the low probability of an event requiring LOPS occurring during this interval.

Condition B

Condition B applies if the Required Action of Condition A and associated Completion Times are not met, or when two undervoltage or two degraded voltage channels in a single bus are inoperable.

Required Action B.1 ensures that the affected diesel generator is declared inoperable and the actions specified in LCO 3.8.1, "AC Sources—Operating," or LCO 3.8.2 "AC—Sources Shutdown," are required immediately.

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BASES (continued)

ACTIONS (continued)

Also, other supported systems affected by LOPS channel inoperability are declared inoperable and the corresponding LCOs entered. [For this facility, the supported systems impacted by LOPS channel inoperability are as follows:]

Condition C

Condition C is applicable to each one of the EDG LOPS functions.

Required Action C.1 verifies that the Required Actions have been initiated for those supported systems declared inoperable because of the inoperability of the support channel(s) within a Completion Time of 1 hour. The specified Completion Time is sufficient for plant operations personnel to make this determination.

Required Action C.1 ensures that those identified Required Actions associated with supported systems impacted by the inoperability of channel(s) associated with each EDG LOPS function have been initiated. This can be accomplished by entering the supported systems' LCOs or independently as a group of Required Actions needed to be initiated every time Condition C is entered. [For this facility, the identified supported systems' Required Actions associated with each EDG LOPS function are as follows:]

Required Action C.2 verifies that all required support or supported features associated with the other redundant channel(s) are OPERABLE within a Completion Time of 1 hour. The specified Completion Time is sufficient for plant operations personnel to make this determination. If verification determines loss of functional capability, LCO 3.0.3 shall be immediately entered. However, if the support or supported feature LCO takes into consideration the loss of function situation, then LCO 3.0.3 may not need to be entered.

SURVEILLANCE REQUIREMENTS

SR 3.3.8.1

SR 3.3.8.1 is the performance of the CHANNEL CHECK once every 12 hours to ensure that a gross failure of

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

instrumentation has not occurred. A CHANNEL CHECK is a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or 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 combination of the channel instrument uncertainties, including isolation, indication, and readability. If a channel is outside the match criteria, it may be an indication that the transmitter or the signal processing equipment has drifted outside its limit. If the channels are within the match criteria, it is an indication that the channels are OPERABLE.

The surveillance interval, about once every shift, is based on operating experience that demonstrates the rarity of channel failure. Thus, performance of the CHANNEL CHECK guarantees that undetected overt channel failure is limited to 12 hours. 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 channel OPERABILITY during normal operational use of the displays associated with this LCO's required channels.

SR 3.3.8.2

SR 3.3.8.2 is the performance of a CHANNEL FUNCTIONAL TEST every 31 days to ensure that the entire channel will perform its intended function when needed. [For this facility, the 31-day Frequency is justified as follows:]

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

This test checks trip devices that provide actuation signals directly. [For this facility, a CHANNEL FUNCTIONAL TEST constitutes the following:]

SR 3.3.8.3

SR 3.3.8.3 is the performance of a CHANNEL CALIBRATION every 18 months. The CHANNEL CALIBRATION verifies the accuracy of each component within the instrument channel. This calibration includes calibration of the undervoltage relays and demonstrates that the equipment falls within the specified operating characteristics defined by the manufacturer. For OPERABLE channels, CHANNEL CALIBRATION shall find that measurement errors and bistable setpoint errors are within the assumptions of the plant-specific setpoint analysis. Recalibration restores OPERABILITY of an otherwise functional component that does not meet these criteria. However, repeated failures of the same channel over a relatively small number of test intervals must be considered as potentially indicating a deterministic failure that cannot be corrected by recalibration. Completion of this test results in the channel being properly adjusted and expected to remain within the "as found" tolerance assumed by the setpoint analysis until the next scheduled surveillance. Measurement and setpoint error determination and readjustment must be performed consistent with the assumptions of the plant-specific setpoint analysis in Reference 5.

The setpoints, as well as the response to a loss of voltage and a degraded voltage test, shall include a single point verification that the trip occurs within the required delay time, as shown in Reference 1. [For this facility, the frequency is justified as follows:]

REFERENCES

1. [Unit Name] FSAR, Section [8.3], "[Onsite Power Systems]."

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

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BASES (continued)

REFERENCES (continued)

3. Title 10, Code of Federal Regulations, Part 50, Appendix A, General Design Criterion 21, "[Title]."

4. Institute of Electrical and Electronic Engineers, IEEE-279, "Criteria for Protection Systems for Nuclear Power Generating Stations."

5. [Unit Name], "[Plant-Specific Setpoint Methodology]."

B 3.3 INSTRUMENTATION

B 3.3.9 Source Range Neutron Flux Channels

BASES

BACKGROUND

The source range neutron flux channels provide the operator with an indication of the approach to criticality at lower power levels than can be seen on the intermediate range neutron flux instrumentation. These channels also provide the operator with a flux indication that reveals changes in reactivity and helps to verify that SHUTDOWN MARGIN is being maintained.

The source range instrumentation has two redundant count rate channels originating in two high-sensitivity proportional counters. Two source range detectors are externally located on opposite sides of the core 180°. These channels are used over a counting range of 0.1 to 1E6 cps and are displayed on the operator's control console in terms of log count rate. The channels also measure the rate of change of the neutron flux level, which is displayed for the operator in terms of start-up rate from -0.5 to +5 decades per minute. An interlock is provided per minute (i.e., a control rod withdraw "inhibit" on a high startup rate of +2 decades per minute in either channel).

The proportional counters of the source range channels are BF_3 chambers. The detector high voltage is automatically turned off when the flux level is approximately 1 decade above the useful operating range. Conversely, the high voltage is turned on automatically when the flux level returns to within approximately 1 decade of the detectors' maximum useful range. High voltage will be automatically turned off when the flux level is above 1E-9 amps in both intermediate range channels, or 10% power in power range channels.

APPLICABLE SAFETY ANALYSES

The source range neutron flux channels are necessary to monitor core reactivity changes. It is the primary means for detecting and triggering operator actions to respond to reactivity transients initiated from conditions in which

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)	the Reactor Protection System (RPS) is not required to be OPERABLE. It also triggers operator actions to anticipate RPS actuation in the event of reactivity transients starting from shutdown or low power conditions. The source range neutron flux channels LCO requirements support compliance with GDC 13 of 10 CFR 50, Appendix A (Ref. 1). The source range neutron flux channels satisfy Criterion 2 of the NRC Interim Policy Statement.
LCO	Two source range neutron flux channels shall be OPERABLE whenever the control rods are capable of being withdrawn to provide the operator with redundant source range neutron instrumentation. The source range instrumentation is the primary power indication at low power levels < 1E-10 amps on intermediate range instrumentation and must remain OPERABLE for the operator to continue increasing power. Source range neutron flux instrumentation is considered OPERABLE when: • All channel components necessary to provide source range indication and supply control function inhibits are functional and in service; • Channel measurement uncertainties are known via test analysis or design information to be within the assumptions of the surveillance acceptance criteria; • Required surveillance testing is current and has demonstrated performance within each surveillance test acceptance criteria; and • The control rod withdrawal "inhibit" on a high start-up rate of +2 decades per minute is OPERABLE. [For this facility, the following support systems are required to be OPERABLE to ensure source range neutron flux instrumentation OPERABILITY:] [For this facility, those required support systems which, upon their failure, do not result in the source range neutron flux instrumentation being declared inoperable and their justification are as follows:]

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BASES (continued)

APPLICABILITY

Two source range neutron flux instrumentation shall be OPERABLE in MODE 2 to provide redundant indication during an approach to criticality. Neutron flux level is sufficient for monitoring on the intermediate range and power range instrumentation prior to entering MODE 1, therefore, source range instrumentation is not required in MODE 1.

A Note has been added allowing detector high voltage to be de-energized above 1E-9 amps on the intermediate range instrumentation. Above this point the source range instrumentation is no longer the primary power indicator. As such, the high voltage to the source range detectors may be de-energized.

In MODES 3, 4, and 5, source range neutron flux instrumentation shall be OPERABLE to provide the operator with a means of monitoring changes in SHUTDOWN MARGIN and to provide an early indication of reactivity changes.

The requirements for source range neutron flux instrumentation during MODE 6 refueling operations are addressed in LCO 3.9.2., "Nuclear Instrumentation."

ACTIONS

A source range is inoperable when it does not satisfy the OPERABILITY criteria for the channel's function. These criteria are outlined for each function in the LCO section of the Bases.

Condition A

The Required Action for one channel of the source range neutron flux indication inoperable while < 1E-9 amps on intermediate range neutron instrumentation is to delay increasing reactor power until the channel is repaired and restored to OPERABLE status. This limits power increases in the range where the operators rely solely on the source range instrumentation for power indication. The Completion Time ensures the source range is available prior to further power increases. Furthermore, it ensures that power remains below the point where the intermediate range channels

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(continued)

BASES (continued)

ACTIONS (continued)

provides primary protection until both source range channels are available to support the overlap verification required by SR 3.3.9.4.

Condition B

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

With both source range neutron flux channels inoperable when reactor power is < 1E-9 amps on the intermediate range instrumentation, the operators must place the reactor in the next lowest condition for which source range instrumentation is not required. This is done by suspending positive reactivity additions, inserting all control rods, and opening the control rod drive trip breakers. Periodic SHUTDOWN MARGIN verification is then required to provide a means for detecting the slow reactivity changes that could be caused by mechanisms other than control rod withdrawal or operations involving positive reactivity changes. Since the source range instrumentation provides the only reliable direct indication of power in this condition, the operators must continue to verify the SHUTDOWN MARGIN every 12 hours until at least one channel of the source range instrumentation is returned to OPERABLE status. The 1-hour Completion Time for Required Action B.3 and Required Action B.4 provides sufficient time for operators to accomplish the actions. Required Action B.1, Required Action B.2, and Required Action B.3 preclude rapid positive reactivity additions. The 12-hour Frequency for repeating SR 3.1.1.1 ensures the reactivity changes possible with control rods inserted are detected before SHUTDOWN MARGIN limits are challenged.

Condition C

With reactor power above 1E-9 amps on the intermediate range instrumentation, continued operation is allowed with one or more source range channels inoperable. The ability to continue operation is justified since the instrumentation does not provide a safety function during high power operation. However, actions are initiated within 1 hour to restore the channels to OPERABLE status such that the channels are available in case they are needed in the future. The Completion Time of 1 hour is sufficient to initiate the actions. The actions must continue until channels are restored to OPERABLE status.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.3.9.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is a comparison of the indicated parameter on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or something more serious. CHANNEL CHECK will detect gross channel failure, thus it is the key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff, based on a combination of the channel instrument uncertainties, including isolation, indication, and readability. If a channel is outside the match criteria, it may be an indication that the transmitter or the signal processing equipment has drifted outside its limit. If the channels are within the match criteria, it is an indication that the channels are OPERABLE. If the channels are normally off scale during times when surveillance 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 downscale.

The Surveillance Frequency, about once every shift, is based on operating experience that demonstrates the rarity of channel failure. Thus, performance of the CHANNEL CHECK guarantees that undetected overt channel failure is limited to 12-hours. 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 protective function due to failure of redundant channels. 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. When operating in Required Action A.1, CHANNEL CHECK is still required. However, in this condition a redundant source range is not available for comparison. CHANNEL CHECK may still be performed via comparison with intermediate range detectors, if available, and verification that the

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(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

OPERABLE source range channel is energized and indicating a value consistent with current plant status.

SR 3.3.9.2

SR 3.3.9.2 is the performance of a CHANNEL CALIBRATION. A CHANNEL CALIBRATION is performed every [18] months. The test is a complete check and readjustment of the channels, from the preamplifier input to the indicators. The test verifies the channel responds to measured parameter with the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift between successive tests to ensure that the instrument channel remains operational. For OPERABLE channels, CHANNEL CALIBRATION shall find that measurement errors are sufficiently small such that measurement and indication errors will not mislead operators into actions that would challenge plant safety limits.

Recalibration restores OPERABILITY of an otherwise functional component that does not meet these criteria. However, repeated failures of the same channel over a relatively small number of test intervals must be considered as potentially indicative of a deterministic failure which cannot be corrected by recalibration.

The SR is modified by a Note excluding neutron detectors from the calibration. It is not necessary to test the detectors 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.

The Surveillance Frequency of [18] months is based on demonstrated instrument CHANNEL CALIBRATION reliability over an [18]-month interval such that the instrument is not adversely affected by drift.

SR 3.3.9.3

SR 3.3.9.3 is the performance of a CHANNEL FUNCTIONAL TEST within 7 days prior to reactor startup. This ensures that the source range instrumentation is functioning properly prior to the operators using the source range neutron flux power indications during startup.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

A CHANNEL FUNCTIONAL TEST verifies the function of the trip, interlock, and alarm functions of the channel. The test inserts a simulated or actual signal as close to the sensor as practicable and verifies required trip, interlock, and alarm functions when the input is beyond the trip point. Where the design has made provisions for including sensors in the CHANNEL FUNCTIONAL TEST, the test signal shall be inserted at that point. "As found" and "as left" values for bistable trip setpoints are recorded. The difference between the current "as found" and the previous "as left" setpoints must be within the drift allowance used in the setpoint analysis. Recalibration of the bistable setpoint restores the OPERABILITY of an otherwise functional component that does not meet these criteria. However, repeated failures of the same channel over a small number of test intervals should be evaluated as potentially indicative of a deterministic failure which cannot be corrected by recalibration.

[At this facility, the test may be omitted if performed within the previous 7 days for the following reasons:]

SR 3.3.9.4

SR 3.3.9.4 is the verification of 1 decade of overlap with the intermediate range instrumentation within 7 days prior to startup. This ensures a continuous source of power indication during the approach to criticality. Failure to perform this surveillance leaves the plant in a safe, subcritical condition until the verification can be made. [At this facility, the test may be omitted if performed within the previous 7 days for the following reasons:]

[For this facility, the provisions of SR 3.0.4 are not applicable for the following reasons:]

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix A, General Design Criterion 13, "Instrumentation and Control."

B 3.3 INSTRUMENTATION

B 3.3.10 Intermediate Range Neutron Flux

BASES

BACKGROUND

The intermediate range neutron flux channels provide the operator with an indication of reactor power at higher power levels than the source range instrumentation and lower power levels than the power range instrumentation.

The intermediate range instrumentation has two log N channels originating in two electrically identical gamma-compensated ion chambers. Each channel provides 8 decades of flux level information in terms of the log of ion chamber current from 1E-10 to 1E-2 amperes. The channels also measure the rate of change of the neutron flux level, which is displayed for the operator in terms of startup rate from -0.5 to +5 decades per minute. A high startup rate of +3 decades per minute in either channel will initiate a control rod withdrawal inhibit.

The intermediate range compensated ion chambers are of the electrically adjustable gamma-compensating type. Each detector has a separate adjustable high-voltage power supply and an adjustable compensating voltage supply.

APPLICABLE SAFETY ANALYSES

Intermediate range neutron flux channels are necessary to monitor core reactivity changes and is the primary indication to trigger operator actions to anticipate Reactor Protection System actuation in the event of reactivity transients starting from low power conditions. The intermediate range neutron flux channels meet the design requirements of 10 CFR 50, Appendix A (Ref. 1).

The intermediate range neutron flux channels satisfy Criterion 2 of the NRC Interim Policy Statement.

LCO

Two intermediate range neutron flux instrumentation channels shall be OPERABLE to provide the operator with redundant

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BASES (continued)

LCO (continued)

neutron flux indication. These enable operators to control the increase in power and to detect neutron flux transients. This indication is used until the power range instrumentation is on scale. Violation of this requirement could prevent the operator from detecting and controlling neutron flux transients which could result in reactor trip during power escalation.

Intermediate range neutron flux instrumentation is considered OPERABLE when:

- All channel components necessary to provide intermediate range indication are functional and in service;
- Channel measurement uncertainties are known (via test, analysis, or design information) to be within the assumptions of the surveillance acceptance criteria;
- Required surveillance testing is current and has demonstrated performance within each surveillance test acceptance criteria; and
- The high startup rate of +3 decades per minute control rod withdrawal inhibit is OPERABLE.

[For this facility, the following support systems are required to be OPERABLE to ensure intermediate range neutron flux instrumentation OPERABILITY:]

[For this facility, those required support systems which, upon their failure, do not result in the intermediate range neutron flux instrumentation being declared inoperable and their justification are as follows:]

[Of the three intermediate range functions above—log current display, startup rate display, and control rod withdrawal inhibit—only the following are covered by LCO 3.3.10, "Intermediate Range Neutron Flux," at this facility:]

APPLICABILITY

The intermediate range channels shall be OPERABLE in MODE 1 when the reactor is critical and with THERMAL POWER below

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BASES (continued)

APPLICABILITY (continued)

10% RATED THERMAL POWER (RTP), or in MODE 2, and in MODES 3, 4, and 5 when the control rod drive (CRD) trip breakers are closed and the CRD System is capable of rod withdrawal.

The intermediate range instrumentation is designed to detect power changes during initial criticality and power escalation when the power range and source range instrumentation cannot provide reliable indications. Since the conditions can exist in all of these modes, the intermediate range instrumentation must be OPERABLE.

ACTIONS

An Intermediate Range Neutron Flux channel is inoperable when it does not satisfy the OPERABILITY criteria for the channel's function. These criteria are outlined for each function in the LCO section of the Bases.

Condition A

A.1 and A.2

If one intermediate range channel becomes inoperable when the channels indicate 1E-9 amps, the plant is exposed to the possibility that a single failure will disable all neutron-monitoring instrumentation. To avoid this, the inoperable channel must be repaired or power must be reduced to the point where source range channels can provide neutron flux indication. Completion of Required Action A.2 places the plant in this state, and LCO 3.3.9, "Source Range Neutron Flux," requires OPERABILITY of two source range detectors once this state is reached. If the one-channel failure occurs when indicated power is < 1E-9 amps, the Required Action prohibits increases in power above the source range capability.

The 2-hour Completion Time allows controlled reduction of power into the source range and is based on plant operating experience that demonstrates the improbability of the second intermediate range channel failing during the allowed interval.

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BASES (continued)

ACTIONS (continued)

Condition B

B.1 and B.2

With two intermediate range neutron flux channels inoperable when THERMAL POWER ≤ 10% RTP, the operators must place the reactor in the next lowest condition for which the intermediate range instrumentation is not required. The Required Actions immediately suspend operations involving positive reactivity changes and within 1 hour place the reactor in a tripped condition with the CRD trip breakers open where the source range instrumentation provides the power level indication. The Completion Times are based on plant operating experience and allow the operators sufficient time to manually insert the control rods prior to opening the CRD breakers.

SURVEILLANCE REQUIREMENTS

SR 3.3.10.1

SR 3.3.10.1 is the performance of a CHANNEL CHECK.

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; thus, it is the key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria is determined by the plant staff, based on a combination of the channel instrument uncertainties, including isolation, indication, and readability. If a channel is outside the match criteria, it may be an indication that the transmitter or the signal-processing equipment has drifted outside its limit. If the channels are within the match criteria, it is an indication that the

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

channels are OPERABLE. If the channels are normally off scale during times when surveillance 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 downscale.

The Surveillance Frequency, about once every shift, is based on operating experience that demonstrates the rarity of channel failure. Thus, performance of the CHANNEL CHECK ensures that undetected overt channel failure is limited to 12 hours. 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 protective function due to failure of redundant channels. 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.

When operating in Required Action A.1 CHANNEL CHECK is still required. However, in this condition a redundant intermediate range is not available for comparison. CHANNEL CHECK may still be performed via comparison with power or source range detectors, if available, and verification that the OPERABLE intermediate range channel is energized and indicating a value consistent with current plant status.

SR 3.3.10.2

SR 3.3.10.2 is the performance of a CHANNEL CALIBRATION every [18] months. The test is a complete check and readjustment of the channels, from the preamplifier input to the indicators. The test verifies the channel responds to the measured parameter with the necessary range and accuracy. CHANNEL calibration leaves the channel adjusted to account for instrument drift between successive tests to ensure that the instrument channel remains operational. For OPERABLE channels, CHANNEL calibration shall find that measurement errors are sufficiently small such that measurement and indication errors will not mislead operators into actions that would challenge plant safety limits.

Recalibration restores OPERABILITY of an otherwise functional component that does not meet these criteria.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

However, repeated failures of the same channel over a relatively small number of test intervals must be considered as potentially indicative of a deterministic failure that cannot be corrected by recalibration.

The SR is modified by a Note excluding neutron detectors from the calibration. It is not necessary to test the detectors because generating a meaningful test signal is difficult. In addition, the detectors are of simple construction, and any failures in the detectors will be apparent as a change in channel output.

The Surveillance Frequency of [18] months is based on demonstrated instrument CHANNEL CALIBRATION reliability over an [18]-month interval such that the instrument is not adversely affected by drift.

<u>SR 3.3.10.3</u>

SR 3.3.10.3 is the performance of a CHANNEL FUNCTIONAL TEST within 7 days prior to reactor startup. This ensures that the intermediate range instrumentation is functioning properly prior to the operators using the intermediate range neutron flux power indications during startup.

A CHANNEL FUNCTIONAL TEST verifies the function of the trip, interlock, and alarm functions of the channel. The test inserts a simulated or actual signal as close to the sensor as practicable and verifies required trip, interlock, and alarm functions when the input is beyond the trip point. Where the design has made provisions for including sensors in the CHANNEL FUNCTION TEST, the test signal shall be inserted at that point. "As found" and "as left" values for bistable trip setpoints are recorded. The difference between the current "as found" and the previous "as left" setpoints must be within the drift allowance used in the setpoint analysis. Recalibration of the bistable setpoint restores the OPERABILITY of an otherwise functional component that does not meet these criteria. However, repeated failures of the same channel over a small number of test intervals should be evaluated as potentially indicative of a deterministic failure that cannot be corrected by recalibration.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

[At this facility, the test may be omitted if performed within the previous 7 days for the following reasons:]

SR 3.3.10.4

SR 3.2.10.4 is the verification within 7 days prior to startup of 1 decade of overlap with the power range instrumentation. This ensures a continuous source of power indication during the approach to criticality. Failure to perform this Surveillance leaves the plant in a condition where the intermediate range channels provide adequate protection until the verification can be made.

[At this facility, the test may be omitted if performed within the previous 7 days for the following reasons:]

[For this facility, the provisions of SR 3.0.4 are not applicable for the following reasons:]

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50 Appendix A, General Design Criterion 13, "Instrumentation and Control."

B 3.3 INSTRUMENTATION

B 3.3.11 Emergency Feedwater Initiation and Control (EFIC) Instrumentation

BASES

BACKGROUND

The EFIC System instrumentation is designed to provide safety-grade means of controlling the secondary system as a heat sink for core decay heat removal. In order to ensure the secondary system remains a heat sink, the EFIC System takes action to initiate emergency feedwater (EFW) when the primary source of feedwater is lost and to isolate functional components from hydraulic faults within the secondary system. These actions ensure that a source of cooling water is available to be fed to a steam generator (SG) which has a controlled steam pressure, thereby fixing the heat sink temperature at the saturation temperature of the secondary system. The EFIC functions that are supported and the parameters that are needed for each of these functions are described below.

The EFIC instrumentation contains devices and circuitry which generate the following signals when monitored variables reach levels that are indicative of conditions requiring protective actions.

1. EFW initiation;
2. EFW vector valve control;
3. Main steam line isolation; and
4. Main feedwater (MFW) isolation.

EFW is initiated to restore a source of cooling water to the secondary system when conditions indicate that the normal source of feedwater is insufficient to continue heat removal. The two indications used for this are the loss of both MFW pumps and a low level in the once-through steam generator (OTSG). EFW is also initiated when action is being taken to isolate the MFW from the OTSG during conditions of uncontrolled depressurizations. This is done by initiating EFW when steam pressure reaches the low OTSG pressure setpoint for isolation of main steam and MFW, and EFW vector valve control. Finally, EFW is initiated when the primary system experiences a total loss of forced

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BASES (continued)

BACKGROUND (continued)

circulation. This initiation, on the loss of all reactor coolant pumps (RCPs) ensures the EFW is available to raise SG levels to promote natural circulation cooling. Additionally, this ensures that EFW is available under the worst case, small-break loss-of-coolant accident (LOCA) conditions when secondary system cooling with high SG water levels is necessary.

The EFIC System also isolates main steam and MFW to an SG that has lost pressure control. With the loss of pressure control, the heat sink temperature control is lost, and the heat removal rate cannot be controlled. The main steam and MFW are isolated to an OTSG when the steam pressure reaches a low setpoint, a condition which is beyond the normal operating point of the secondary system.

The EFIC System also performs an EFW control function to avoid delivering EFW to a depressurized SG when the other SG remains pressurized. This continues the function of isolating functional components from an OTSG whose pressure cannot be controlled. This function precludes the delivery of fluid to a depressurized SG, thereby avoiding an uncontrolled cooling condition as long as the other SG remains pressurized. When both of the SGs are depressurized, the EFIC logic provides EFW flow to both SGs until a significant pressure difference between the two OTSGs is developed, which ensures that core cooling is maintained.

Trip Setpoints and ALLOWABLE VALUES

The trip setpoints are the nominal value at which the bistables are set. Any bistable is considered to be properly adjusted when the "as left" value is within the band for CHANNEL CALIBRATION accuracy, i.e., ± (rack calibration + comparator setting accuracy).

The trip setpoints used in the bistables are based on the analytical limits stated in Reference 5. The selection of these trip setpoints is such that adequate protection is provided when all sensor and processing time delays are taken into account. The ALLOWABLE VALUES specified in Table 3.3.11-1 are conservatively adjusted with respect to the analytical limits, to allow for calibration tolerances, instrumentation uncertainties, instrument drift, and severe

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BASES (continued)

BACKGROUND (continued)

environment errors for those EFIC channels that must function in harsh environments as defined by 10 CFR 50.49 (Ref. 4). A detailed description of the methodology used to calculate the trip setpoints, including their explicit uncertainties, is provided in plant-specific setpoint methodology (Ref. 3). The actual nominal trip setpoint entered into the bistable is more conservative than that specified by the ALLOWABLE VALUE to account for changes in random measurement errors detectable by a CHANNEL FUNCTIONAL TEST. One example of such a change in measurement error is drift during the surveillance interval. If the measured setpoint does not exceed the ALLOWABLE VALUE, the bistable is considered OPERABLE.

Setpoints in accordance with the ALLOWABLE VALUE ensure that the consequences of Design Basis Accidents (DBAs) are acceptable providing the plant is operated from within the LCOs at the onset of the DBA and that the equipment functions as designed.

Each channel can be tested on line to verify that the setpoint accuracy is within the specified allowance requirements of the FSAR (Ref. 6). Once a designated channel is taken out of service for testing, a simulated signal is injected in place of the field instrument signal. The process equipment for the channel in test is then tested, verified, and calibrated. The SRs for the channels are specified in the SRs section.

The ALLOWABLE VALUES listed in Table 3.3.11-1 are based upon the plant-specific setpoint methodology (Ref. 3) which incorporates all of the known uncertainties applicable for each channel. The magnitudes of these uncertainties are factored into the determination of each trip setpoint. All field sensors and signal processing equipment for these channels are assumed to operate within the allowances of these uncertainty magnitudes.

Figure B 3.3.11-1 illustrates EFIC EFW initiation logic operation.

Each EFIC train actuates on a one-out-of-two taken twice combination of trip signals from the instrumentation channels. Each EFIC channel can issue an initiate command, but an EFIC actuation will take place only if at least two

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BASES (continued)

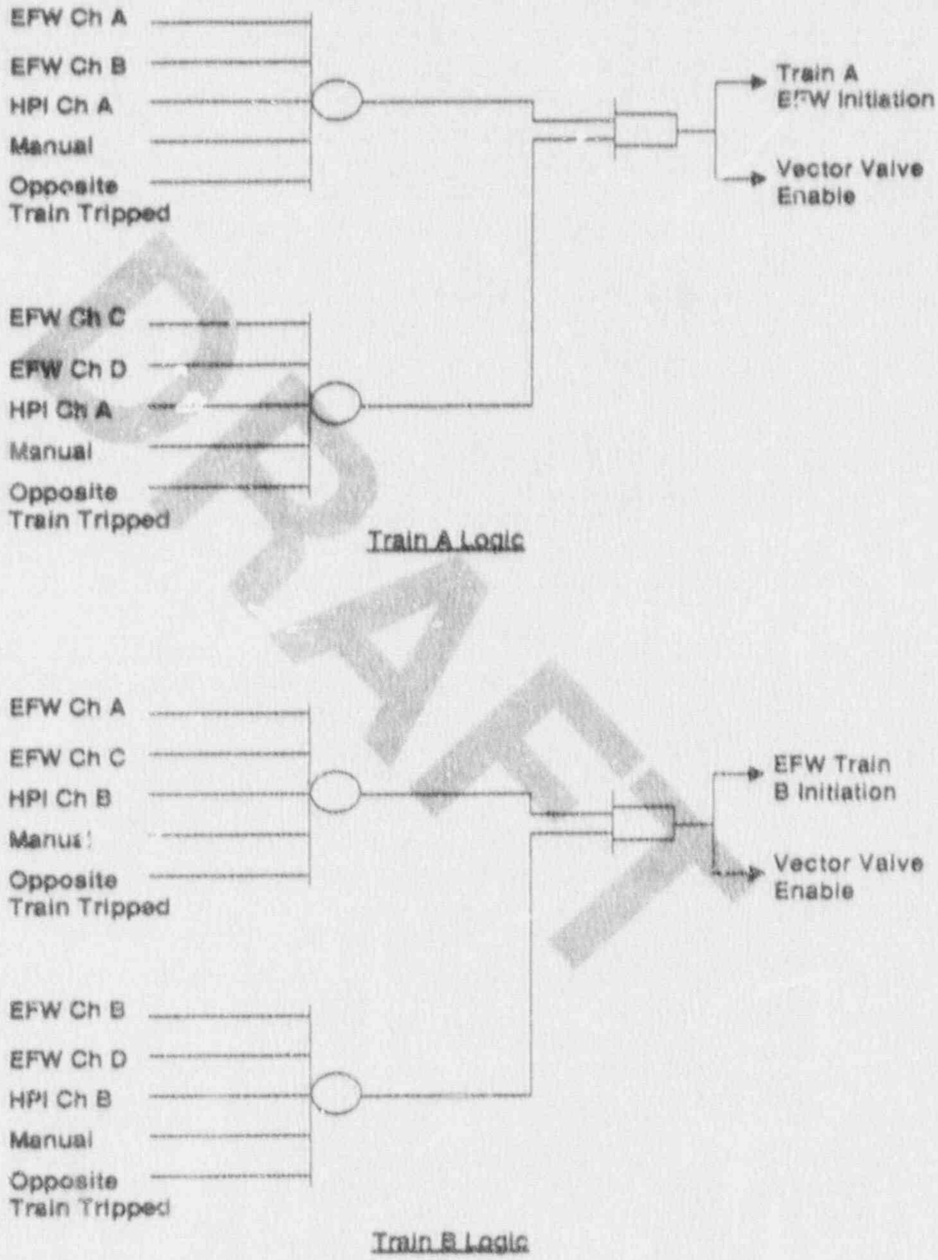

Figure B 3.3.11-1
Emergency Feedwater Initiation and Control Emergency Feedwater Initiation Logic

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BASES (continued)

BACKGROUND (continued)

channels issue initiate commands. The one-out-of-two logic combinations are transposed between trains so that failure of two channels prevents actuation of, at most, one train.

More detailed descriptions of the EFIC instrumentation are provided below.

1. EFW Initiation

Figure B 3.3.11-2 illustrates one channel of the EFIC EFW initiation channel. The individual instrumentation channels that serve this function are discussed below.

1.a. Loss of MFW Pumps

Loss of both MFW pumps is one of the four parameters within the EFIC System that automatically initiates EFW. Loss of MFW pumps is detected by MFW pump turbine control oil pressure. The MFW pump status instrumentation is a part of the Nuclear Instrumentation System (NIS) and Reactor Protection System (RPS). Each RPS channel receives an input from two control oil pressure switches, one on each pump. If both switches in a single channel trip, the associated RPS channel trips. Each RPS channel provides both MFW pumps tripped signal to the associated EFIC channel. The RPS bypasses the trip function when THERMAL POWER ≤ 20% RATED THERMAL POWER (RTP).

Loss of both MFW pumps was chosen as an EFW automatic initiating parameter because it is a direct and immediate indicator of loss of MFW.

1.b. OTSG Level--Low Low

Four EFIC dedicated low range level transmitters per SG are used to generate the signals used for detection for low low level conditions for EFW actuation. There is one transmitter for each of the four channels A, B, C, and D. The signals are also used after EFW is actuated to control OTSG level at the low level setpoint [30 inches] when one or more RCPs are operational.

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BASES (continued)

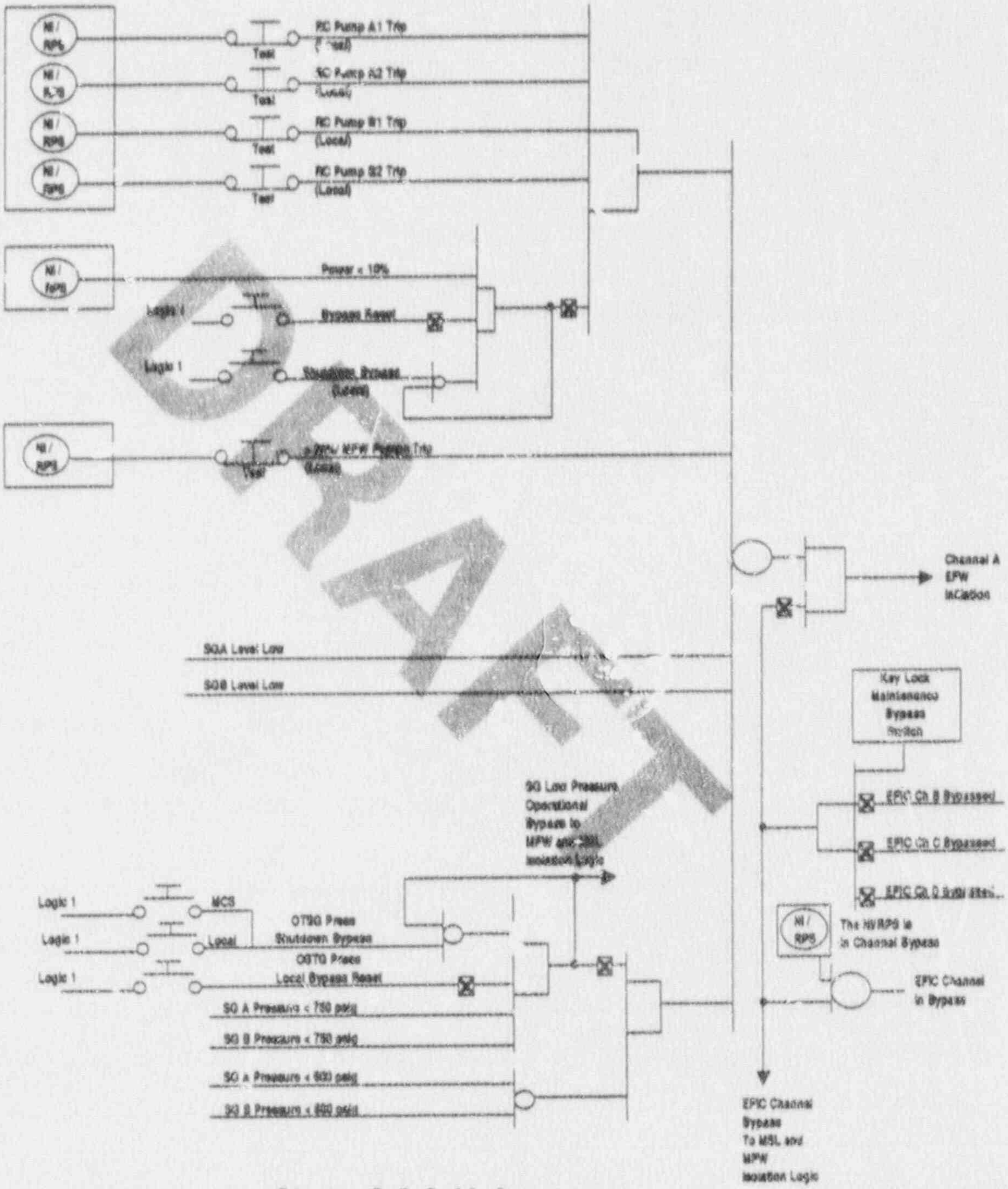

Figure B 3.3.11-2
Emergency Feedwater Initiation and Control Emergency
Feedwater Initiation Channel

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BASES (continued)

BACKGROUND (continued)

The lower and upper taps for the low range level transmitters are located at [6 inches] and [277 inches], respectively, above the upper face of the SG's lower tube sheet. The calibrated range is [0-150 inches].

OTSG Level--Low Low was chosen as an EFW automatic initiating parameter because it indicates that the primary feedwater source is insufficient to meet the heat removal requirements and therefore additional cooling water is necessary to ensure core decay heat removal.

1.c. OTSG Pressure--Low

Four transmitters per SG provide the EFIC System with channels A through D of SG pressure. These are the same transmitters used by the MFW and main steam line isolation functions. When the SG pressure drops below the bistable setpoint of [600] psig on a given channel, an EFW initiation signal is sent to the automatic actuation logic. The low pressure function may be manually bypassed when both SGs are less than [750] psig. If either SG input channel exceeds [750] psig, the EFIC channel bypass is automatically removed. The low pressure operational bypass allows for normal cooldown without EFIC initiation.

SG low pressure is a primary indication and actuation signal for steam line breaks (SLBs) or feedwater line breaks. For small breaks, which do not depressurize the SG or take a long time to depressurize, automatic actuation is not required. The operator has time to diagnose the problem and take the appropriate actions.

1.d. RCP Status

A loss of power to all four RCPs is an indication of a pending loss of forced flow in the Reactor Coolant System. These sensing signals are input into the four channels of EFIC.

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BASES (continued)

BACKGROUND (continued)

When at least two channels issue initiate commands based on loss of all RCPs, the EFIC System will automatically actuate EFW and switch the level control setpoint to approximately 50% in the SG. This higher setpoint provides a thermal center in the SG at a higher elevation than that of the reactor to ensure natural circulation of the reactor coolant.

To allow heatup and cooldown operations without actuation, a bypass permissive of 10% RTP is used. The 10% bypass permissive was chosen because it was an available, qualified Class 1E signal at the time the EFIC System was designed. When the first RCP is started, the "loss of four RCPs" initiation signal may be manually reset. If the bypass is not manually reset, it will be automatically reset when the plant reaches 10% power. During cooldown, the bypass may be inserted at any time the power has been reduced below 10%. However, for most operating conditions, it is recommended that this trip function remain active until after the Decay Heat Removal System has been initiated and the system is ready for the last RCP to be tripped.

2. EFW Vector Valve Control

Figure B 3.3.11-3 illustrates one channel of the EFIC EFW vector valve control logic. The function of the EFW vector logic is to determine whether EFW should not be fed to one or the other SG. This is to preclude the continued addition of EFW to a depressurized SG and thus minimize the overcooling effects of a steam leak.

There are four sets of vector logic, one in each channel of EFIC. Each set of vector logic receives SG pressure information from bistables located in the input logic of the same EFIC channel. The pressure information received is:

a. SG A pressure less than [600] psig;

b. SG B pressure less than [600] psig;

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BASES (continued)

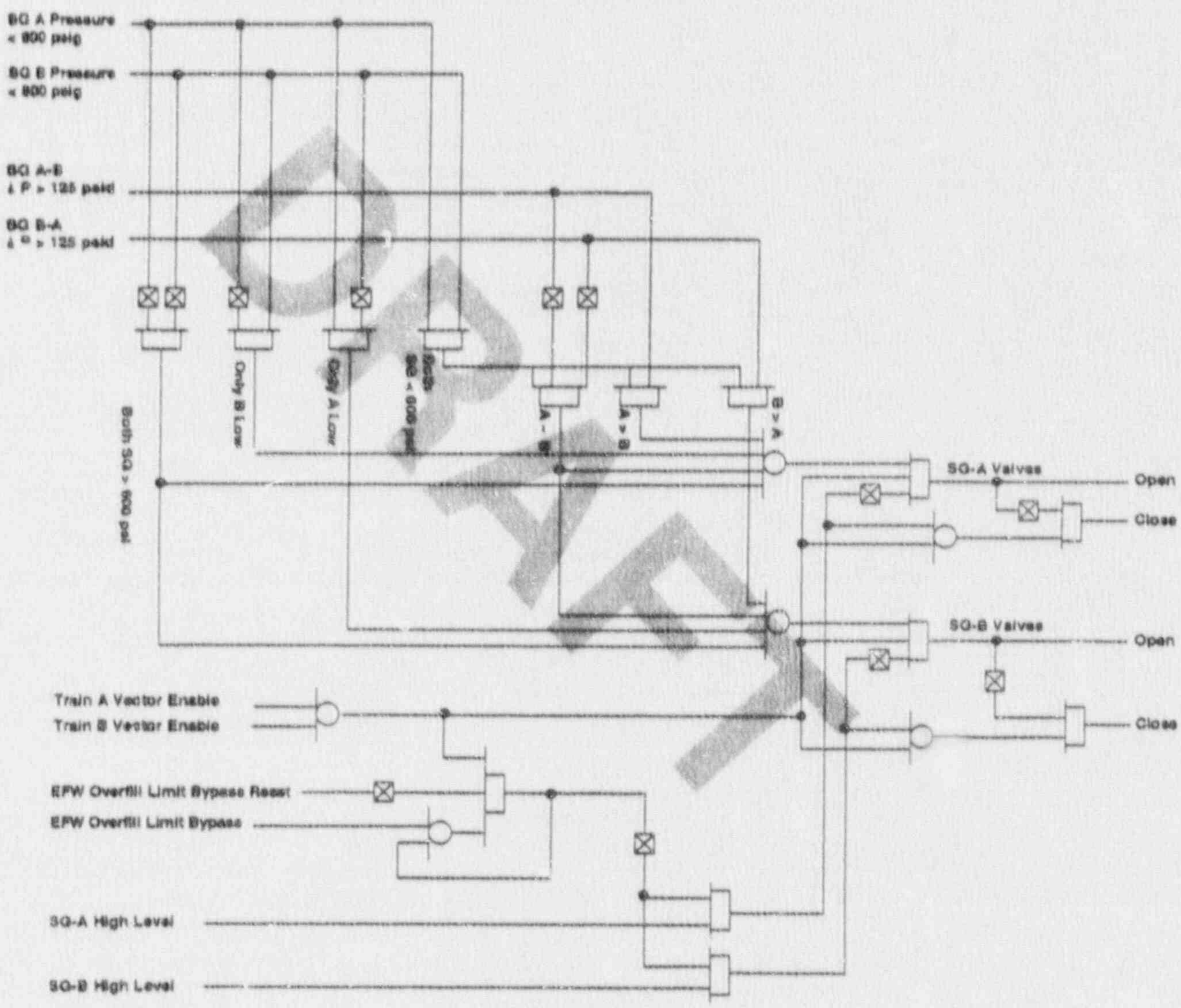

Figure B 3.3.11-3
Emergency Feedwater Initiation and Control
Emergency Feedwater Vector Valve Control Logic

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BASES (continued)

BACKGROUND (continued)

c. SG A pressure [125] psid greater than SG B pressure; and

d. SG B pressure 125 psid greater than SG A pressure.

Each vector logic also receives a vector/control enable signal from both EFIC channel A and channel B when EFW is initiated. [Each logic also receives an OTSG high level signal. High level in an SG prevents opening the associated vector valves and enables closing the valves without either EFIC train vector valve enable.]

The vector logic develops signals to open or close SG A and B EFW valves.

The vector logic outputs are in a neutral state until enabled by the control/vector enable from the channel A or B trip logics. When enabled, the vector logic can issue close commands to the EFW control valves and open or closed commands to the EFW isolation valves per the selected channel assignments.

Each vector logic may isolate EFW to only one SG or the other, never both.

The valve open or close commands are determined by the relative values of SG pressures as follows:

Pressure Status	SG A Valves	SG B Valves
SGA and SGB > [600] psig	open	open
| SGA - SGB | < [125] psid	open	open
SGA or SGB $\leq$ [600] psig and SGA - SGB $\geq$ 125 psid	open	close
SGA or SGB $\leq$ [600] psig and SGB - SGA $\geq$ 125 psid	close	open

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BASES (continued)

BACKGROUND (continued)

Bypass

One of the four initiation channels can be put into "maintenance bypass." Bypassing one initiation logic channel bypasses all functions in that initiation channel. An interlock feature prevents bypassing more than one channel at a time. In addition, since the EFIC System receives signals from the NIS or RPS, the maintenance bypass from the NIS or RPS is interlocked with the EFIC System. If one channel of the NIS or RPS is in maintenance bypass, only the corresponding channel of the EFIC may be bypassed (e.g., channel A, NIS or RPS, and channel A, EFIC). This ensures that only corresponding channels of the EFIC and NIS or RPS are placed in maintenance bypass at the same time.

EFIC channel maintenance bypass does not bypass EFW initiation from Engineered Safety Feature Actuation System (ESFAS) high pressure injection (HPI).

The ESFAS HPI initiation function can, however, be bypassed from within ESFAS.

Operational bypass of the OTSG Pressure- Low and Loss of MFW Pumps initiation functions is provided to allow normal plant shutdown.

The operational bypass provisions were discussed as part of the individual functions above.

Operational bypass of the OSTG Level- High input to the vector valve logic is possible after EFIC initiation. [For this facility, bypassing the overfill function is for the following reasons:]

3, 4. Main Steam Line and MFW Isolation

Figure B 3.3.11-4 illustrates one channel of the EFIC main steam line and MFW isolation logic. Four pressure transmitters per SG provide EFIC with channels A through D logic of SG pressure. The channels are as described for EFW initiation above.

Once isolated, manual action is required to defeat the isolation command if desired. The EFIC System is designed to perform its intended function with one channel in maintenance bypass (in effect, inoperable) with a single failure in one of the remaining channels. This is in

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BASES (continued)

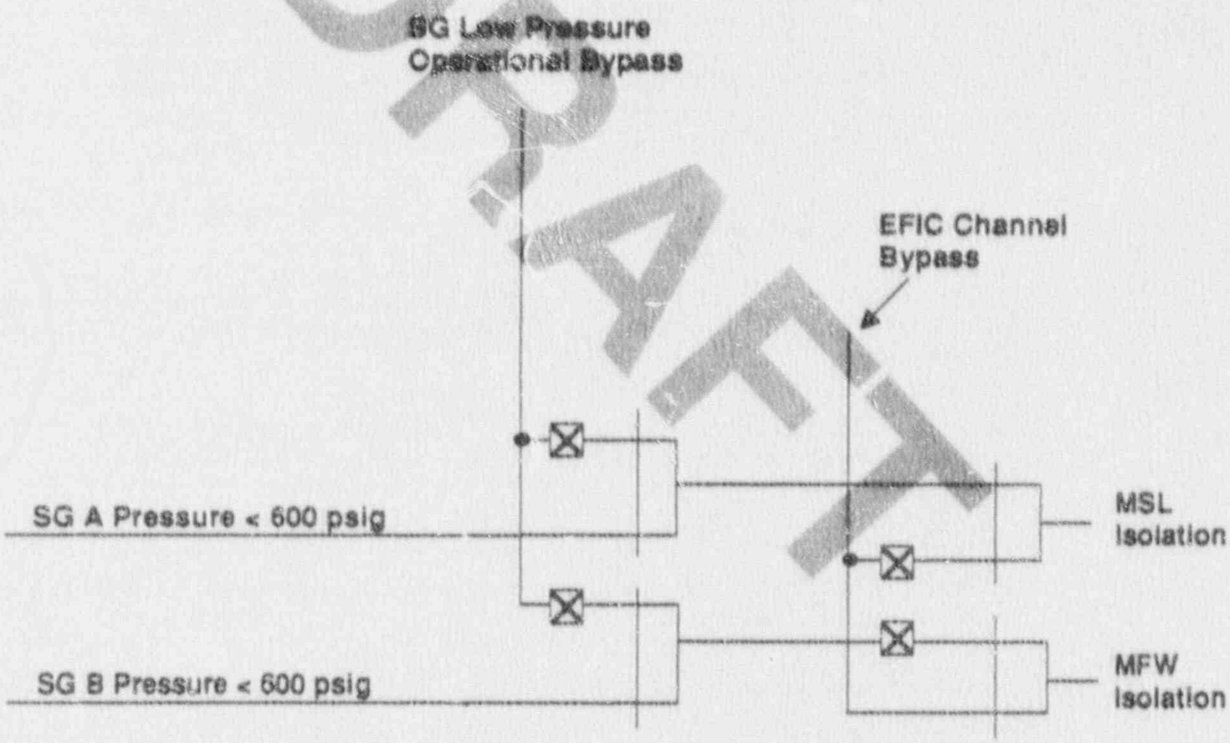

Figure B 3.3.11-4
Main Steam Line and Main Feedwater Line Isolation Logic

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BASES (continued)

BACKGROUND (continued)

compliance with IEEE-279 (Ref. 1) and GDC 21 in 10 CFR 50, Appendix A (Ref. 2) due to the redundancy and independence in the EFIC design.

In order to take full advantage of the four channel design, all wiring associated with the RPS must meet separation requirements of Regulatory Guide 1.75, and each channel must be energized from a separate inverter and station batteries. Plants meeting these requirements may operate in a two-out-of-three logic with one channel removed from service indefinitely if such design was approved by the NRC staff (Ref. [ ]). Designs not approved by the NRC staff can operate in a two-out-of-three logic for a limited time (72 hours) only.

APPLICABLE SAFETY ANALYSES

1. <u>EFW Initiation</u>

Although loss of both MFW pumps is a direct and immediate indicator of loss of MFW, other scenarios such as valve closures potentially could be the cause for the loss of feedwater. The loss of MFW analysis, therefore, conservatively assures the actuation of EFW on low SG level. If the loss of feedwater is due to loss of MFW pumps, EFW will be actuated much earlier than assumed in the analysis, which will increase the SG heat transfer capability and lessen the severity of the transient.

The DBA parameters selected for initiation of the EFW systems is a loss of MFW transient. In the analysis of this transient, OTSG Level--Low Low is the parameter assumed to automatically initiate EFW. This assumption yields the least SG inventory available for heat removal and is, therefore, conservative.

OTSG Level--Low Low would be an indicator of all accidents involving a loss of primary-to-secondary heat removal. OTSG Pressure--Low is a primary indication and provides the actuation signal for SLBs or feedwater line breaks. For small breaks, which do not depressurize the SG or take a long time to

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

depressurize, automatic actuation is not required. The operator has time to diagnose the problem and take the appropriate actions.

Loss of four RCPs is a primary indicator of the need for EFW in the safety analyses for loss of electric power and loss of coolant flow. It also serves as a backup indicator for SLBs and small-break LOCAs.

2. EFW Vector Valve Control

Most of the FSAR SLB analyses were performed prior to the development of the safety-grade EFIC System. Therefore, the EFIC vector valve control was not credited in the original licensing basis for a main SLB analysis. Instead, operator action was credited with isolating EFW to the affected SG within the first 60 seconds. However, isolating the affected SG is a function automatically performed by the EFIC System. Therefore, the FSAR analysis remains conservative relative to the inclusion of the vector valve logic.

3, 4. Main Steam Line and MFW Isolation

The FSAR analysis assumed integrated control system action for feedwater and SG isolation. The analysis took credit for turbine stop valve closure and feedwater valve isolation on reactor trip and considered the isolation functions occurring on SG pressure < [600] psig as backup. These isolation functions are currently provided by the safety grade EFIC System. Use of the EFIC System in the original safety analysis would have been consistent with the licensing position allowing mitigative functions to be performed by safety-grade systems in accident analysis. For these reasons, the SLB accident analysis remains conservative with the assumed integrated control system actions.

The EFIC System satisfies Criterion 3 of the NRC Interim Policy Statement.

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BASES (continued)

LCO

All instrumentation performing an EFIC System function shall be OPERABLE. Failure of any instrument renders the affected channel(s) inoperable and reduces the reliability of the affected functions.

Four channels are required OPERABLE for all EFIC instrumentation channels to ensure that no single failure prevents actuation of a train. Each EFIC instrumentation channel is considered to include the sensors and measurement channels for each function, the operational bypass switches and permissives, and the logic that combines the instrumentation functions to cause a protective channel trip. Failures that disable the capability to place a channel in operational bypass but which do not disable the trip function do not render the protection channel inoperable.

EFIC instrumentation channels are OPERABLE when:

- All channel components necessary to provide an EFIC signal are functional and in service;
- Channel measurement uncertainties are known via test, analysis, or design information to be within the assumptions of the setpoint calculations;
- Required surveillance testing is current and has demonstrated performance within each surveillance test's acceptance criteria; and
- The associated operational bypass is not enabled except under the conditions specified by the LCO applicability statement for the function.

Only the ALLOWABLE VALUES are specified for each EFIC initiation and bypass removal function in the LCO. In Table 3.3.11-1, ALLOWABLE VALUES for the bypass removal functions are specified in terms of Applicability limits on the associated trip function. Nominal trip setpoints are specified in the plant-specific setpoint calculations. The nominal setpoints are selected to ensure the setpoint measured by CHANNEL FUNCTIONAL TESTS do not exceed the ALLOWABLE VALUE if the bistable is performing as required. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its ALLOWABLE VALUE, is

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BASES (continued)

LCO (continued)

acceptable provided that operation and testing is consistent with the assumptions of the plant-specific setpoint calculations. Each ALLOWABLE VALUE specified is more conservative than the analytical limit assumed in the safety analysis in order to account for instrument uncertainties appropriate to the trip function. These uncertainties are defined in the plant-specific setpoint methodology (Ref. 3).

The Bases for the LCO requirements of each specific EFIC function are discussed below.

Loss of MFW Pumps

Four EFIC channels shall be OPERABLE with MFW pump turbines A and B control oil low pressure actuation setpoints of ≥ [55] psig. The [55] psig setpoint is about half of the normal operating control oil pressure. The [55] psig setpoint ALLOWABLE VALUE was arbitrarily chosen as a good indication of loss of MFW pump. Analysis only assumes loss of MFW pumps and a specific value of MFW pump control oil pressure is not used in the analysis. The loss of MFW pumps function includes a bypass enable and removal function from the RPS. [At this facility the basis for the [20]% RTP setpoint ALLOWABLE VALUE for the bypass removal function is as follows:]

OTSG Level--Low Low

Four EFIC dedicated low range level transmitters per SG shall be OPERABLE to generate the signals used for detection for low low level conditions for EFW actuation. There is one transmitter for each of the four channels A, B, C, and D. The signals are also used after EFW is actuated to control at the low level setpoint of [30] inches when one or more RCPs are OPERABLE with OTSG Level--Low Low actuation setpoints of ≥ [9] inches. In the determination of the low low level setpoint, it is desired to place the setpoint as low as possible, considering instrument errors, to give the maximum operability margin between the integrated control system low load control setpoint and the EFW actuation setpoint. This will minimize spurious or unwanted initiation of EFW. Credit is only taken for low low level actuation for those transients which do not involve a

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BASES (continued)

LCO (continued)

degraded environment. Therefore, normal environment errors only are used for determining the SG level low low level setpoint.

OTSG Pressure--Low

Four EFIC channels per SG shall be OPERABLE with SG low pressure actuation setpoints of ≥ [600] psig. The setpoint is chosen to avoid actuation under transient conditions not requiring secondary system isolation, preferring to maintain a steaming path to the condenser if possible. Small-break LOCA analyses have indicated minimum secondary system pressures of approximately [700] psig. The OSTG Pressure--Low function includes a bypass enable and removal function. [At this facility, the basis for the ≥ [750] psig ALLOWABLE VALUE for the bypass removal function is as follows:]

OTSG Differential Pressure--High

Four EFIC channels for SG differential pressure shall be OPERABLE with setpoints of ≤ [125] psid. The setpoint ensures that automatic EFW isolation to a depressurized SG occurs for the range of sizes of SLBs that require rapid actuation early in the event. The setpoint has also been chosen to avoid spurious isolation of EFW during conditions due to relatively small deviations in SG pressures which can be caused by primary system conditions. The OSTG Differential Pressure--High function includes a bypass enable and removal function. [At this facility, the basis for the ≥ [750] psig ALLOWABLE VALUE, for the bypass removal function is as follows:]

RCP Status

Four EFIC channels for RCP status shall be OPERABLE. This ensures that upon the loss of four RCPs, the EFW control level will automatically be raised to approximately 50%, providing a higher SG level for establishing and maintaining natural circulation conditions when the forced reactor coolant flow is lost. No setpoint is specified since the status indication as used by EFIC is binary in nature. The RCP status function includes a bypass enable and removal function from the RPS. [At this facility the basis for the ≥ [10]% RTP ALLOWABLE VALUE is as follows:]

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BASES (continued)

LCO (continued)

[For this facility, the basis for OTSG Level--High Signal is as follows:]

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

[For this facility, those required support systems which, upon their failure, do not result in the EFIC instrumentation being declared inoperable and their justification are as follows:]

[For this facility, the supported systems impacted by the inoperability of the EFIC instrumentation and the justification of whether or not each supported system is declared inoperable are as follows:]

[It should be noted that LCO 3.3.11 may need to be augmented with additional conditions, if it is determined that the EFIC instrumentation provides support to other systems in the Standard Technical Specifications.]

APPLICABILITY

The EFIC instrumentation functions shall be OPERABLE in accordance with Table 3.3.11-1 at all times that automatic secondary heat sink protection is required. Each function has its own requirements based on the specific accidents and conditions which it is designed to protect against.

The initiation of EFW on the loss of MFW pumps shall only be required above [20%] RTP, when core power production and heat removal requirements are the greatest. Below this power level, the EFW System actuation on low OTSG level is rapid enough to avoid unnecessary primary system overheating.

EFW initiation on low low OTSG level shall be OPERABLE at all times the SG is required for heat removal. These conditions include MODES 1, 2, and 3. However, to avoid automatic actuation of the EFW pumps during normal heatup and cooldown transients, the low OTSG pressure function can be bypassed at or below a secondary pressure of [750] psig. This secondary pressure can normally only be reached during MODE 3 operation.

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BASES (continued)

APPLICABILITY (continued)

The trip of all RCPs function shall be operable in MODES 1, 2, and 3. It is possible to bypass the function below [10%] RTP, and in order to prevent actuation of the EFW pumps it must be bypassed prior to stopping the last RCP.

For all lower MODES, the primary system temperatures are too low to allow the SGs to effectively remove energy.

The MFW, main steam line isolation, and EFW vector valve control functions shall be OPERABLE in MODES 1, 2, and 3 because the SG inventory can be at a high energy level and contribute significantly to the peak reactor building pressure with a secondary-side break. Both the normal feedwater and the EFW must be able to be isolated on each SG to limit overcooling of the primary and mass and energy releases to the reactor building. Once the SG pressures have decreased below [750] psig, the main steam line and MFW isolation functions can be bypassed to avoid actuation during normal plant cooldowns. The EFW vector valve control logic will not perform any function when both SG pressures are low, thus the logic can also be bypassed at the same point. In MODES 4, 5, and 6 the energy level is low and the secondary-side feedwater flow rate is low or non-existent.

A Note has been added to the Applicability to provide clarification that, for this LCO, each function specified in Table 3.3.11-1 shall be treated as an independent entity with an independent Completion Time.

ACTIONS

A channel is inoperable when it does not satisfy the OPERABILITY criteria for the function's channel. These criteria are outlined in the LCO section of the Bases. A protection function may be rendered inoperable because of overt failure or because the bistable or process module has drifted sufficiently so as to exceed the ALLOWABLE VALUE. Typically, the drift is small 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 FUNCTIONAL TEST when the process instrument is set up for adjustment to bring it within specification. If the trip setpoint is not consistent with the ALLOWABLE VALUE in Table 3.3.11-1, the channel must be declared inoperable

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BASES (continued)

ACTIONS (continued)

immediately, and the appropriate Condition from Table 3.3.11-1 must be entered.

In the event a channel's trip setpoint is found nonconservative with respect to the ALLOWABLE VALUE, or any of the transmitter, signal processing electronics, or EFIC channel cabinet modules are found inoperable, then all affected functions provided by that channel must be declared inoperable and the plant must enter the Conditions for the particular protection function affected.

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

Condition A

Condition A applies to failures of a single EFW initiation, main steam line isolation, or MFW isolation instrumentation channel. This includes failure of the same instrumentation channel in any combination of the functions.

With one channel inoperable in one or more EFW initiation, main steam line isolation, or MFW isolation functions, the channel must be placed in bypass or trip within 1 hour. This condition applies to failures which occur in a single channel, channel A for example, which when bypassed will remove all functions within the channel from service. Since the RPS and EFIC channels are linked, only the corresponding channel in each system may be bypassed at any time. This condition is forced by an electrical interlock. If testing of another channel in either the EFIC or RPS is required, the EFIC channel must be placed in trip to allow the other channel to be bypassed. With the channel in trip, the resultant logic is one-out-of-two. The Completion Time of 1 hour is adequate to perform the required action.

Required Action A.2.1 and Required Action A.2.2 provide for restoring the channel to OPERABLE status within 72 hours or placing the channels in trip.

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BASES (continued)

ACTIONS (continued)

A single inoperable EFIC instrumentation channel affects at most one train of EFW, main steam line isolation, and MFW isolation. Therefore, the 72-hour Completion Time was selected to be consistent with the allowed out-of-service time for the EFW, main steam line isolation, and MFW isolation functions.

Condition B

Condition B applies to a situation where instrumentation channels for multiple protection channels of EFW initiation, main steam line isolation, or MFW isolation instrumentation are inoperable. For example, Condition B applies if channel A and B of the EFW initiation function are inoperable.

Condition B does not apply if one channel of different functions are inoperable in the same protection channel. That condition is addressed by Condition A.

With two EFW initiation, main steam line isolation, or MFW isolation protection channels inoperable, one channel must be placed in bypass (Required Action B.1) to ensure the remaining OPERABLE channels are not bypassed. Bypassing one of the remaining OPERABLE channels would leave the EFIC unable to complete its safety functions. Therefore, the second channel must be tripped (Required Action B.2) to prevent a single failure from causing loss of the EFIC function. The Completion Time of 1 hour is adequate to perform the Required Action.

One of the channels must be returned to OPERABLE status (Required Action B.3) to preclude spurious signals from initiating EFW, main steam line isolation, and MFW isolation. Restoring one channel changes system status to Condition A. A single inoperable EFIC channel affects at most one train of EFW, main steam line isolation, and MFW isolation. Therefore the 72-hour Completion Time was selected to be consistent with the allowed out-of-service time for the EFW, main steam line isolation, and MFW isolation functions.

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BASES (continued)

ACTIONS (continued)

Condition C

The function of the EFW control and isolation valve complex and the vector logic is to meet the single failure criterion while being able to provide EFW on demand and isolate an SG when required. These conflicting requirements result in the necessity for two valves in series, in parallel with two valves in series, and a four channel valve command system.

With one EFW vector valve channel inoperable the system cannot meet the single failure criterion and still meet the dual functional criteria above. This condition is analogous to having one EFW train inoperable. Therefore, when one vector valve channel is inoperable, the channel must be restored to OPERABLE status (Required Action C.1) within 72 hours, which is consistent with the Completion Time associated with the loss of one train of EFW.

Condition D

If the Required Actions cannot be met within the required Completion Time, the plant must be placed in a MODE in which the requirement does not apply. This is done by placing the plant in at least MODE 3 within 6 hours (Required Action D.1) and in MODE 4 within 12 hours (Required Action D.2). The allowed Completion Times are reasonable, based on operating experience, to reach the required MODES from full power in an orderly manner and without challenging plant systems.

Condition E

Condition E is applicable to each one of the EFIC instrumentation functions presented in Table 3.3.11-1.

Required Action E.1 verifies that the Required Actions have been initiated for those supported systems declared inoperable because of the inoperability of the support channel(s) within a Completion Time of 1 hour. The specified Completion Time is sufficient for plant operations personnel to make this determination.

Required Action E.1 ensures that those identified Required Actions associated with supported systems impacted by the inoperability of channel(s) associated with each EFIC

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BASES (continued)

ACTIONS (continued)

instrumentation function have been initiated. This can be accomplished by entering the supported systems' LCOs or independently as a group of Required Actions needed to be initiated every time Condition E is entered. [For this facility, the identified supported systems' Required Actions associated with each EFIC instrumentation function are as follows:]

Required Action E.2 verifies that all required support or supported features associated with the other redundant channel(s) are OPERABLE within a Completion Time of 1 hour. The specified Completion Time is sufficient for plant operations personnel to make this determination. If verification determines loss of functional capability, LCO 3.0.3 must be immediately entered. However, if the support or supported feature LCO takes into consideration the loss of function situation, then LCO 3.0.3 may not need to be entered.

SURVEILLANCE REQUIREMENTS

The SRs for any particular EFIC instrumentation function are found in the SRs column of Table 3.3.11-1. All functions are subject to CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION. The OSTG--Low Low Level function is the only function that was modeled in transient analysis and thus is the only EFIC function subjected to response time testing. Individual EFIC subgroup relays must also be tested, one at a time, to verify the individual EFIC components will actuate when required. Some components cannot be tested at power since their actuation might lead to plant trip or equipment damage. These are specifically identified and must be tested when shutdown. The various SRs account for individual functional differences and test frequencies applicable specific to the functions listed in Table 3.3.11-1. The operational bypasses associated with each EFIC instrumentation channel are also subject to these SRs to ensure OPERABILITY of the EFIC instrumentation channel.

SR 3.3.11.1

SR 3.3.11.1 is the performance of a CHANNEL CHECK. Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

CHANNEL CHECK is a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or even something more serious. CHANNEL CHECK will detect gross channel failure, thus it is the key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff, based on a combination of the channel instrument uncertainties, including isolation, indication, and readability. If a channel is outside the match criteria, it may be an indication that the transmitter or the signal processing equipment has drifted outside its limit. If the channels are within the match criteria, it is an indication that the channels are OPERABLE. If the channels are normally off scale during times when surveillance 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 downscale.

The Surveillance Frequency, about once every shift, is based on operating experience that demonstrates the rarity of channel failure. Thus, performance of the CHANNEL CHECK guarantees that undetected overt channel failure is limited to 12 hours. 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 protective function due to failure of redundant channels. 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.

SR 3.3.11.2

SR 3.3.11.2 is the performance of a CHANNEL FUNCTIONAL TEST every 31 days. A CHANNEL FUNCTIONAL TEST verifies the function of the trip, interlock, and alarm functions of the channel. The test inserts a simulated or actual signal as close to the sensor as practicable and verifies required trips, interlocks, and alarm functions when the input is

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

beyond the trip point. Where the design has made provisions for including sensors in the CHANNEL FUNCTIONAL TEST, the test signal shall be inserted at that point. "As found" and "as left" values for bistable trip setpoints are recorded. Bistable setpoints for both trip and bypass removal functions must be found within the ALLOWABLE VALUE specified in the LCO. (Note that the ALLOWABLE VALUES for the bypass removal functions are specified in the Applicability column of Table 3.3.11-1 as limits on Applicability for the trip functions.) The difference between the current "as found" and the previous "as left" setpoints must be within the drift allowance used in the setpoint analysis. Recalibration of the bistable setpoint restores the operability of a otherwise functional component that does not meet these criteria. However, repeated failures of the same channel over a small number of test intervals should be evaluated as potentially indicating a deterministic failure which cannot be corrected by recalibration.

The Surveillance Frequency of 31 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 31-day interval is a rare event.

SR 3.3.11.3

SR 3.3.11.3 is the performance of a CHANNEL CALIBRATION every [18] months.

CHANNEL CALIBRATION is a complete check of the instrument channel including the detector. The test verifies the channel responds to a measured parameter with the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift between successive tests to ensure that the instrument channel remains operational with the setpoint within the assumptions of the plant specific setpoint analysis. Transmitter "as found" and "as left" values are recorded and used to verify drift assumptions. For OPERABLE channels, CHANNEL CALIBRATION shall find that measurement errors and bistable setpoint errors are within the assumptions of the plant-specific setpoint analysis. Measurement and setpoint error

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

determination and readjustment must be performed consistent with the assumptions of the plant-specific setpoint analysis.

Recalibration restores operability of an otherwise functional component found to have errors larger than assumed by the setpoint analysis. However, repeated failures of the same channel over a relatively small number of test intervals must be considered as potentially indicating a deterministic failure which cannot be corrected by recalibration.

Field transmitters may be calibrated in place, removed and calibrated in a laboratory, or replaced with an equivalent laboratory-calibrated unit.

The Surveillance Frequency is based upon the assumption of an [18]-month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

SR 3.3.11.4

SR 3.3.11.4 is the performance of the EFIC RESPONSE TIME test on an [18]-month STAGGERED TEST BASIS. The EFIC RESPONSE TIME values are the maximum values assumed in the safety analyses. Individual component response times are not modeled in the analyses. The analyses model the overall or total elapsed time from the point at which the parameter exceeds the actuation setpoint value at the sensor to the point at which the end device is actuated. [For this facility, the EFIC RESPONSE TIME acceptance criteria are found in the following document:]

EFIC RESPONSE TIME tests are conducted on an [18]-month STAGGERED TEST BASIS. This results in the interval between successive tests of a given channel of n times [18] months, where n is the number of channels in the function. Testing of the final actuation devices, which make up the bulk of the EFIC RESPONSE TIME, is included in the testing of each channel. Therefore, staggered testing results in response time verification of these devices every [18] months. The [18]-month test 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

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)	occurrences. EFIC RESPONSE TIMES cannot be determined at power since equipment operation is required.

REFERENCES

1. Institute of Electrical and Electronic Engineers, IEEE-279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations," April 5, 1972.

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

3. [Unit Name], "[Plant-Specific Setpoint Methodology]."

4. Title 10, Code of Federal Regulations, Part 50.49, "Environment Qualification of Electric Equipment Important to Safety in Nuclear Power Plants."

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

6. [Unit Name] FSAR, Section [7], "[Instrumentation and Control]."

B.3.3 INSTRUMENTATION

B 3.3.12 Emergency Feedwater Initiation and Control (EFIC) Manual Initiation

BASES

BACKGROUND

The EFIC manual initiation capability provides the operator with the capability to actuate EFIC functions from the control room in the absence of any other initiation condition. Manually actuated functions include main feedwater (MFW) isolation for steam generator (SG) A, MFW isolation for SG B, main steam line isolation for SG A, main steam line isolation for SG B, and emergency feedwater (EFW) initiation. These functions are provided in the event the operator determines that an EFIC function was not automatically actuated. This is a backup function to those performed by EFIC.

The EFIC manual initiation circuitry satisfies the manual initiation and single failure criterion requirements of IEEE 279 (Ref. 1).

APPLICABLE SAFETY ANALYSES

Normally, EFIC functions are automatic. However, the manual initiation functions are required as backups to the automatic trip functions and allow operators to initiate EFW, main steam line isolation, or MFW isolation whenever any parameter is rapidly trending toward its trip setpoint. Furthermore, the manual initiation of EFW, main steam line isolation, and MFW isolation may be specified in plant operating procedures.

The EFIC manual initiation functions satisfy Criterion 3 of the NRC Interim Policy Statement.

LCO

All instrumentation performing an EFIC manual initiation function shall be OPERABLE. Failure of any instrument renders the affected channel(s) inoperable and reduces the reliability of the affected functions.

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BASES (continued)

LCO (continued)

EFIC manual initiation is OPERABLE when:

a. All channel components necessary to provide a reactor trip are functional and in service; and

b. Required surveillance testing is current and has demonstrated performance within each surveillance test's acceptance criteria.

Two manual initiation switches per actuation channel (A and B) of each function (MFW isolation, main steam line isolation, and EFW initiation) are required to be OPERABLE whenever the SGs are being relied upon to remove heat. Each function (MFW isolation, main steam line isolation, and EFW initiation) has two actuation or "trip" channels, channels A and B. Within each channel A actuation logic there are two manual trip switches. When one manual switch is depressed, a half-trip occurs. When both manual switches are depressed, a full trip of channel A actuation occurs. Similarly, channel B actuation logic for each function has two manual trip switches. Both switches per actuation channel must be OPERABLE and be depressed in order to get a full manual trip of that channel. The use of two manual trip switches for each channel of actuation logic allows for testing without actuating the end devices and also reduces the possibility of accidental manual actuation.

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

[For this facility, those required support systems which, upon their failure, do not result in the EFIC manual initiation circuit being declared inoperable and their justification are as follows:]

[For this facility, the supported systems impacted by the inoperability of the EFIC manual initiation circuit and the justification of whether or not each supported system is declared inoperable are as follows:]

APPLICABILITY

The MFW and main steam line isolation manual actuation functions shall be OPERABLE in MODES 1, 2, and 3 because SG

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BASES (continued)

APPLICABILITY (continued)

inventory can be at a high energy level and contribute significantly to the peak containment pressure during a secondary-side break. In MODES 4, 5, and 6, the SG energy level is low and secondary-side feedwater flow rate is low or nonexistent.

The EFW manual actuation function shall be OPERABLE in MODES 1, 2, and 3 because the SGs are relied upon for Reactor Coolant System heat removal. In MODES 4, 5 and 6, heat removal requirements are reduced and can be provided by the decay heat removal system.

A Note has been added to the Applicability to provide clarification that for this LCO the EFIC manual initiation function is treated as an independent entity with an independent Completion Time.

ACTIONS

With two inoperable EFIC manual initiation switches in any actuation channel, facility operation is not allowed to continue in this degraded condition. Therefore, LCO 3.0.32 must be immediately entered if applicable in the current mode of operation.

Condition A

With one manual initiation channel of one or more EFIC functions inoperable, the channel must be placed in the tripped condition within 1 hour. With the channel in the tripped condition, the single failure criterion is met and the operator can still initiate one actuation channel. Failure to perform this Required Action A.1 could allow a single failure of another switch to prevent manual actuation of at least one of two trip channels. The time allotted to trip the channel allows the operator to take all the appropriate actions for the failed channel and still ensure that the risk involved in operating with the failed channel is acceptable.

With one channel in trip, the second manual initiate channel cannot be surveillance tested without causing a full actuation channel trip. Therefore, an inoperable switch must be restored to OPERABLE prior to the next required

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BASES (continued)

ACTIONS (continued)

manual CHANNEL FUNCTIONAL TEST of the affected EFIC function.

Condition B

If the Required Action A.1 cannot be met within the required Completion Time, the plant must be placed in a MODE in which the requirement does not apply. This is done by placing the plant in at least MODE 3 within 6 hours and in MODE 4 within 12 hours. The Completion Times are reasonable, based on operating experience, to reach the required MODES from full power in an orderly manner and without challenging plant systems.

Condition C

Condition C applies if one manual initiation switch per actuation channel of one or more EFIC functions are inoperable.

Required Action C.1 verifies that the Required Actions have been initiated for those supported systems declared inoperable because of the inoperability of the support manual initiation switches within a Completion Time of 1 hour. The specified Completion Time is sufficient for plant operations personnel to make this determination.

Required Action C.1 ensures that those identified Required Actions associated with supported systems impacted by the inoperability of manual initiation switches associated with the EFIC manual initiation function have been initiated.

This can accomplished by entering the supported systems LCOs or independently as a group of Required Actions associated with the EFIC manual initiation function are as follows:]

Required Action C.2 verifies that all required support or supported features associated with the other redundant manual initiation switches are OPERABLE within a Completion Time of 1 hour. The specified Completion Time is sufficient for plant operations personnel to make this determination. If verification determines loss of functional capability, LCO 3.0.3 must be immediately entered. However, if the

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BASES (continued)

ACTIONS (continued)

support or supported feature LCO takes into consideration the loss of function situation, then LCO 3.0.3 may not need to be entered.

SURVEILLANCE REQUIREMENTS

SR 3.3.12.1

SR 3.3.12.1 is the performance of a CHANNEL FUNCTIONAL TEST to ensure that the channels can perform their intended functions and shall be performed once every 31 days. This test verifies that the initiating circuitry is OPERABLE and will actuate the end device (i.e., pump, valves, etc.). However, for MFW and main steam line isolation the test need not include actuation of the end device. This is due to the risk of a plant transient caused by the closure of valves associated with MFW and main steam line isolation during testing at power. The surveillance interval of 31 days is based upon operating experience that demonstrates the rarity of more than one channel failing within the same 31-day interval.

REFERENCES

1. Institute of Electrical and Electronic Engineers, IEEE-279-1971 "Criteria for Protection Systems for Nuclear Power Generating Stations," April 5, 1972.

B 3.3 INSTRUMENTATION

B 3.3.13 Emergency Feedwater Initiation and Control (EFIC) Logic

BASES

BACKGROUND

Main Steam Line and Main Feedwater (MFW) Isolation

The four emergency feedwater initiation and control (EFIC) channels sensing a steam generator (SG) low outlet pressure condition input their initiate commands to the trip logic modules. Figure B 3.3.11-3 illustrates the main steam line and MFW isolation logic. The trip logic modules are physically located in the "A" and "B" EFIC channel cabinets. Train "A" actuation logic initiates when instrumentation channel "A" or "B" initiates and channel "C" or "D" initiates, which in simplified logic is:

"A" actuation = (A and C) or (A and D) or (B and C) or (B and D)

Train "B" actuation logic initiates when instrumentation channel "A" or "C" initiates and channel "B" or "D" initiates, which in simplified logic is:

"B" actuation = (A and B) or (A and D) or (B and C) or (C and D)

The four functions (SG A main feedwater isolation, SG B main feedwater isolation, SG A main steam line isolation, and SG B main steam line isolation) have a channel "A" and a channel "B" of the automatic actuation logic.

Both channels "A" and "B" of the SG A main feedwater isolation automatic actuation logic send closure signals to the SG A main feedwater pump suction valve, the three SG A block valves, and the MFW pump discharge cross-connect valve. In addition, SG A EFIC "A" and "B" channels trip MFW pump "A."

Both channels "A" and "B" of the SG A main steam line isolation automatic actuation logic send closure signals to both of the SG A main steam isolation valves.

SG B isolation automatic actuation logics respond similarly for the SG B valves and MFW pump "B."

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BASES (continued)

BACKGROUND (continued)

Emergency Feedwater (EFW) Initiation

The four EFIC channels for each of the parameters being sensed input their initiate commands to the trip logic modules. Figure B 3.3.11-1 illustrates the EFW initiation logic. These trip logic modules are physically located in the "A" and "B" EFIC channel cabinets.

EFW initiation functions are the same logic combinations as MFW and main steam line isolation. EFW initiation also occurs on high pressure injection (HPI) initiation. Each train of HPI initiation drives the corresponding train of EFIC EFW initiation.

EFIC automatically initiates the EFW System when any of the following conditions exist:

1. All four reactor coolant pumps are tripped;
2. Both MFW pumps are tripped and reactor power is > 20% RTP with the nuclear instrumentation Reactor Protection System not in shutdown bypass;
3. Low level in either SG;
4. Low pressure in either SG; or
5. HPI actuation on both A and B Engineered Safety Feature Actuation System channels.

Each EFIC train actuates both EFW trains.

Vector Valve Enable

The EFW module logic is responsible for sending open or close signals to the EFW control and isolation valves. Figure B 3.3.11-4 illustrates the vector valve logic. The vector module logic outputs are in a neutral state (neither commanding open or close) until a signal is received from the vector enable logic. The vector enable logic monitors the channel A and B EFW initiation logics. When an EFW initiation occurs, the vector enable logic enables the vector logic to generate open or close signals to the EFW valves depending on the relative values of SG pressures.

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BASES (continued)

APPLICABLE SAFETY ANALYSES

Automatic isolation of MFW and main steam line was assumed in the safety analyses to mitigate the consequences of main steam line or MFW line ruptures. The FSAR analyses for steam line breaks (SLBS) were generated before the development and installation of the safety-grade EFIC System, which currently performs these automatic safety functions. The FSAR analysis, for example, assumes main steam line isolation through turbine stop valve closure based on an integrated control system signal. This same function is provided by the EFIC System by a safety-grade signal that closes the main steam line isolation valves. The analyses are bounding and the use of the EFIC System is consistent with the licensing position to take credit for safety-grade systems to mitigate the consequences of an accident.

Similarly, logic valve control was not credited in the FSAR SLB analysis. Operator action was credited with isolating EFW to the affected SG within the first 60 seconds. This function would be automatically performed by EFIC. Therefore, the FSAR analysis remains conservative relative to the inclusion of the vector valve logic.

Automatic initiation of EFW is credited in the loss of main feedwater analysis. The analysis assumed both MFW pumps tripped to initiate the event. The automatic actuation was based on the once-through steam generator low level function of EFIC, although EFIC would initiate EFW based on the loss of both MFW pumps.

The EFIC logic satisfies Criterion 3 of the NRC Interim Policy Statement.

LCO

Two channels of each MFW and main steam line isolation and EFW initiation automatic actuation logic shall be OPERABLE. There are only two channels of automatic actuation logic per function. Therefore, violation of this LCO could result in a complete loss of the automatic function assuming a single failure of the other channel.

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BASES (continued)

LCO (continued)

EFIC logic is OPERABLE when:

- All channel components necessary to provide a reactor trip are functional and in service; and
- Required surveillance testing is current and has demonstrated performance within each surveillance test's acceptance criteria.

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

[For this facility, those required support systems which, upon their failure, do not result in the EFIC logic being declared inoperable and their justification are as follows:]

[For this facility, the supported systems impacted by the inoperability of EFIC logic and the justification of whether or not each supported system is declared inoperable are as follows:]

APPLICABILITY

The MFW and main steam line isolation automatic actuation functions shall be OPERABLE in MODES 1, 2, and 3 because SG inventory can be at a high-energy level and contribute significantly to the peak containment pressure during a secondary-side line break. In MODES 4, 5, and 6, the energy level is low and the secondary-side feedwater flow rate is low or nonexistent.

The EFW automatic initiation and vector valve logics shall be OPERABLE in MODES 1, 2, and 3 because the SGs are being used for heat removal from the primary system. During these MODES, the core power and heat-removal requirements are the greatest, and if the normal source of feedwater is lost, EFW must be initiated rapidly to minimize the overheating of the primary system.

For MODES 4, 5, and 6, the primary system temperatures are too low to allow the SGs to effectively remove energy.

For this LCO, a Note has been added to the Applicability statement to provide clarification that each EFIC logic function shall be treated as an independent entity with an independent Completion Time.

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BASES (continued)

ACTIONS

A EFIC logic is inoperable when it does not satisfy the OPERABILITY criteria for the channel's function. These criteria are outlined for each function in the LCO section of the Bases.

In the event a channel is found inoperable, then all affected functions provided by that channel must be declared inoperable and the LCO Condition entered for the particular protection function affected.

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

Condition A

Condition A applies when one or more EFIC logic functions in the same channel are inoperable (i.e., channel A could be inoperable for all five EFIC logic functions and Condition A would still be applicable). This Condition is equivalent to failure of one EFW, main steam line isolation, and MFW isolation train.

The EFIC System has not been analyzed for failure of one train of one function and the opposite train of another function. In this condition, the potential for system interactions that disable heat removal capability on EFW has not been dismissed. Consequently, failure of any logic function in channel A and the same or different logic function in channel B is a potential loss of EFIC function. Therefore, any combination of failures in both channel A and B is not covered by Condition A and must be addressed by entry into LCO 3.0.3.

A.1

With one automatic actuation logic channel of one or more EFIC functions inoperable, the associated EFIC train must be declared inoperable since automatic actuation is unavailable. Since there are only two automatic actuation logic channels per EFIC function, the condition of one

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BASES (continued)

ACTIONS (continued)

channel inoperable is analogous to having one train of a two-train engineered safety feature (ESF) system inoperable. The system safety function can be accomplished, however, a single failure can not be taken. Therefore, the failed channel(s) must be restored to OPERABLE status to reestablish the system's single-failure tolerance.

Since Condition A can be equivalent to failure of a single train of the EFW, main steam line isolation, and MFW isolation capability, the Completion Time is set at 72 hours to be consistent with Completion Times for restoring one inoperable ESF system train.

Condition B

If the Required Action A.1 can not be met within the required Completion Time, the plant must be placed in a MODE in which the requirement does not apply. This is done by placing the plant in at least MODE 3 within 6 hours and in MODE 4 within 12 hours. The Completion Times are reasonable, based on operating experience, to reach the required MODES from full power in an orderly manner and without challenging plant systems.

Condition C

Condition C is applicable with one or more channel A or B functions inoperable.

Required Action C.1 verifies that the Required Actions have been initiated for those supported systems declared inoperable because of the inoperability of the support channel(s) within a Completion Time of 1 hour. The specified Completion Time is sufficient for plant operations personnel to make this determination.

Required Action C.1 ensures that those identified Required Actions associated with supported systems impacted by the inoperability of channel(s) associated with each EFIC logic function have been initiated. This can be accomplished by entering the supported systems' LCOs or independently as a group of Required Actions that need to be initiated every time Condition C is entered. [For this facility, the identified supported systems' Required Actions associated with each EFIC logic function are as follows:]

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BASES (continued)

ACTIONS (continued)

Required Action C.2 verifies that all required support or supported features associated with the other redundant channel(s) are OPERABLE within a Completion Time of 1 hour. The specified Completion Time is sufficient for plant operations personnel to make this determination. If verification determines loss of functional capability, LCO 3.0.3 must be immediately entered. However, if the support or supported feature LCO takes into consideration the loss-of-function situation, then LCO 3.0.3 may not need to be entered.

SURVEILLANCE REQUIREMENTS

SR 3.3.13.1

SR 3.3.13.1 is the performance of a CHANNEL FUNCTIONAL TEST once every 31 days to ensure that the channels can perform their intended functions. This test verifies MFW and main steam line isolation and EFW initiation automatic actuation logics are functional. The test simulates the required inputs to the logic circuit and verifies successfully operation of the automatic actuation logic. The test need not include actuation of the end device. This is due to the risk of a plant transient caused by the closure of valves associated with MFW and main steam line isolation or actuation of EFW during testing at power. The Surveillance Frequency of 31 days is based on operating experience, which has demonstrated the rarity of more than one channel failing within the same 31-day interval.

REFERENCES

None.

B 3.3 INSTRUMENTATION

B 3.3.14 Reactor Building Purge Isolation--High Radiation

BASES

BACKGROUND

The Reactor Building Purge Isolation--High Radiation System closes the reactor building purge valves. This action isolates the reactor building atmosphere from the environment to minimize releases of radioactivity in the event of an accident. The high-radiation signal indicates a failure of a barrier to the fuel radioactivity, and most likely a loss-of-coolant accident. The purge valves must begin to shut rapidly to ensure they reach a completely closed position prior to excessive pressures in the reactor building, against which the valves may not close.

The radiation monitoring system measures the activity in a representative sample of air drawn in succession through a particulate sampler, an iodine sampler, and a gas sampler. The LCO addresses only the gas sampler portion of this system. The sensitive volume of the gas sampler is shielded with lead and monitored by a G-M detector. The air sample is taken from the center of the purge exhaust duct through an isokinetic nozzle installed in the duct at a point selected for reduced turbulence.

Should a gaseous activity flow rate of approximately 1E-2 μCi/sec (Kr-85) be exceeded, the monitor will alarm and initiate closure of the purge valves. This activity flow rate is selected on the basis of 250,000 scfm flow rate in the purge exhaust and a gas monitor setpoint equal to two times the expected background at the location of the monitor, which will provide fast detection of any release. The alarm setpoints for the particulate and iodine channels are such that an alarm is obtained after the monitor samples a maximum permissible concentration level for 8 hours. Therefore, a maximum of 1.3 mCi of Cs-137 or 67 μCi of DOSE EQUIVALENT I-131 will be released to the atmosphere during this period. Alarm setpoints can also be based on isotopic analyses.

Purge isolation occurs on RCS low pressure and reactor building high pressure signals. Engineered Safety Feature Actuation System (ESFAS) purge isolation functions are covered by LCO 3.3.5, "Engineered Safety Feature Actuation

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BASES (continued)

BACKGROUND (continued)

System (ESFAS) Instrumentation," LCO 3.3.6, "Engineered Safety Feature Actuation System (ESFAS) Manual Initiation," and LCO 3.3.7, "Engineered Safety Feature Actuation System (ESFAS) Automatic Actuation Logic".

[At this facility, the purge isolation function on high radiation satisfies the single-failure criterion as follows:]

The closure of the purge valves ensures the reactor building remains as a barrier to fission product release. There is no bypass for this function. The closure of the purge valves provides a reactor building isolation assumed in the accident analysis.

Trip Setpoints and ALLOWABLE VALUE

The trip setpoints are the nominal value at which the bistables are set. Any bistable is considered to be properly adjusted when the "as left" value is within the band for CHANNEL CALIBRATION accuracy, i.e., ± (rack calibration + comparator setting accuracy).

The trip setpoints used in the bistables are based on the analytical limits derived from the FSAR (Ref. 1). 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, instrument drift, and severe environment errors for those channels that must function in harsh environments as defined by 10 CFR 50.49 (Ref. 2). ALLOWABLE VALUES specified in LCO 3.3.14 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 the plant-specific setpoint methodology (Ref. 3). The actual nominal trip setpoint entered into the bistable is normally still more conservative than that specified by the ALLOWABLE VALUE to account for changes in random measurement errors detectable by a CHANNEL FUNCTIONAL TEST. One example of such a change in measurement error is drift during the surveillance interval. If the measured setpoint does not exceed the ALLOWABLE VALUE, the bistable is considered OPERABLE.

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BASES (continued)

BACKGROUND (continued)	Setpoints in accordance with the ALLOWABLE VALUE will ensure that 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 LCO 3.3.14, the ALLOWABLE VALUE of SR 3.3.14.3 is the LSSS. These ALLOWABLE VALUES are established to prevent violation of the safety limits during normal plant operation and AOOs. The ALLOWABLE VALUE in SR 3.3.14.3 is based on the methodology described in Reference 3, which incorporates all of the known uncertainties applicable for each channel. The magnitudes of these uncertainties are factored into the determination of each trip setpoint. All field sensors and signal-processing equipment for these channels are assumed to operate within the allowances of these uncertainty magnitudes. Each channel of the process control equipment can be tested on line to verify that the signal or setpoint accuracy is within the specified allowance requirements. Once a designated channel is taken out of service for testing, a simulated signal is injected in place of the field instrument signal. The process equipment for the channel in test is then tested, verified, and calibrated.
APPLICABLE SAFETY ANALYSES	The analysis for the maximum hypothetical accident assumes the reactor building remains intact, with penetrations that are unnecessary for core cooling isolated early in the event, within approximately [60] seconds. The closure of the purge valves ensures the reactor building integrity assumed in the analysis is maintained. The isolation of the reactor building has not been analyzed mechanistically in the dose calculations, although its rapid isolation is assumed. The Reactor Building Purge Isolation System satisfies Criterion 3 of the NRC Interim Policy Statement.

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BASES (continued)

LCO

The instrumentation necessary to initiate a purge valve closure on a high-radiation signal shall be OPERABLE at all times that the purge valves are open. This provides the capability to isolate the reactor building if an accident were to occur.

Only the ALLOWABLE VALUES are specified for each reactor building purge isolation trip function in the LCO. Nominal trip setpoints are specified in the plant-specific setpoint calculations. The nominal setpoints are selected to ensure the setpoint measured by CHANNEL FUNCTIONAL TESTS does not exceed the ALLOWABLE VALUE if the bistable 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 is consistent with the assumptions of the plant-specific setpoint calculations. Each ALLOWABLE VALUE specified is more conservative than the analytical limit assumed in the safety analysis in order to account for instrument uncertainties associated with the trip function. These uncertainties are defined in the plant-specific setpoint analysis (Ref. 3) and the Offsite Dose Calculation Manual.

Reactor building purge isolation high radiation is OPERABLE when the following criteria are met:

- All channel components necessary to provide a reactor building purge valve closure of a high-radiation signal are functional and in service. For sampling systems, OPERABILITY requires correct valve lineups, sample pump operation, and filter motor operation, as well as detector OPERABILITY, when these sampling features are necessary to initiate a trip as assumed by the safety analysis or setpoint analysis.
- Channel measurement uncertainties are known (via test analysis, or design information) to be within the assumptions of the setpoint calculations.
- Required surveillance testing is current and has demonstrated performance within each surveillance test's acceptance criteria.

[For this facility, the basis for the setpoint ALLOWABLE VALUE is as follows:]

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BASES (continued)

LCO (continued)

[For this facility, the following support systems are required to be OPERABLE to ensure reactor building purge isolation high radiation instrumentation OPERABILITY:]

[For this facility, those required support systems which, upon their failure, do not result in the reactor building purge isolation high radiation instrumentation being declared inoperable and their justification are as follows:]

[For this facility, the supported systems impacted by the inoperability of reactor building purge isolation high radiation instrumentation and the justification of whether or not each supported system is declared inoperable are as follows:]

APPLICABILITY

The reactor building purge isolation high radiation shall be OPERABLE whenever the reactor building purge valves are open in MODES 1, 2, 3, and 4. Outside of these MODES, the purge isolation must be OPERABLE whenever CORE ALTERATIONS or movement of irradiated fuel is taking place. These conditions are those under which the potential for fuel damage, and thus radiation release, is the greatest. While in MODES 5 and 6 without fuel handling in progress, the Purge Valve Isolation System need not be OPERABLE since the potential for a radioactive release is minimized and operator action is sufficient to ensure post-accident offsite doses are maintained within the limits of 10 CFR 100. The need to use the purge valves in MODES 5 and 6 is in preparation for entry. This capability is required to minimize doses for personnel entering the building and is independent of the automatic isolation capability.

ACTIONS

A protection function channel is inoperable when it does not satisfy the OPERABILITY criteria for the channel's function. These criteria are outlined in the LCO section of the Bases.

The most frequent occurrence to render a protection function inoperable is the determination that a bistable or process module has drifted sufficiently to exceed the ALLOWABLE VALUE. Typically, the drift is small and results in a delay of actuation rather than a total loss of function. This

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BASES (continued)

ACTIONS (continued)

determination is generally made during the performance of an ANALOG CHANNEL OPERATIONAL TEST, when the process instrumentation is set up for adjustment to bring it within specification. If the trip setpoint is less conservative than the ALLOWABLE VALUE, the channel must be declared inoperable immediately and the appropriate Condition must be entered.

In the event a channel's trip setpoint is found nonconservative with respect to the ALLOWABLE VALUE, or the transmitter, & rack module, or an SSPS module is found inoperable, then the function which that channel provides must be declared inoperable and the LCO Condition entered for the particular protection function affected.

When the required channels are specified on a per bus or per train basis, then the Condition may be entered separately for each bus or train as appropriate.

Condition A

Condition A applies to failure of the purge isolation function in MODE 1, 2, 3, or 4.

A.1, A.2.1, A.2.2, A.2.3, A.3.1, and A.3.2

If the inoperable channel cannot be restored, operation may continued as long as the Containment Purge System supply and exhaust valves are maintained in the closed position as required by Required Action A.2. This action accomplishes the safety function of the Containment Purge Isolation--Radiation High function. The 1-hour Completion Time is reasonable considering the time required to isolate the penetration and the relative importance of maintaining containment OPERABILITY during MODES 1, 2, 3, and 4.

In the event that neither Required Action A.1 nor Required Action A.2 can be met within the associated Completion Times, the plant must be placed in a MODE in which the LCO does not apply. This is done by placing the plant in at least MODE 3 within the following 6 hours and in MODE 5 within the following 36 hours. The times allowed to reach MODES 3 and 5 from MODE 1 are reasonable, based on operating experience, to reach the required MODES from full power in an orderly manner and without challenging plant systems.

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BASES (continued)

ACTIONS (continued)

Condition B

Condition B applies to failure of the high-radiation purge function during CORE ALTERATIONS or movement of irradiated fuel in the reactor building.

B.1, B.2, B.3.1, and B.3.2

With the purge isolation instrumentation inoperable, the channel must be restored, or the purge valves must be closed, or CORE ALTERATIONS and movement of irradiated fuel within the reactor building must be suspended. Required Action B.2 accomplishes the function of the high-radiation channel. Required Action B.3.1 and Required Action B.3.2 place the plant in a configuration in which purge isolation on high radiation is not required. The Completion Time of "immediately" is consistent with the urgency associated with the loss of reactor building isolation capability under conditions in which fuel-handling accidents are possible, and the high-radiation function provides the only automatic actions to mitigate radiation release.

Condition C

Required Action C.1 verifies that the Required Actions have been initiated for those supported systems declared inoperable because of the inoperability of the support channel[(s)] within a Completion Time of 1 hour. The specified Completion Time is sufficient for plant operations personnel to make this determination.

Required Action C.1 ensures that those identified Required Actions associated with supported systems impacted by the inoperability of channel[(s)] associated with the reactor building purge isolation function have been initiated. This can be accomplished by entering the supported systems' LCOs. [Alternatively, the appropriate Required Actions for the supported systems may be listed in the Required Actions for Condition C of this LCO.] [For this facility, the identified supported systems associated with the reactor building purge isolation function are as follows:]

[Required Action C.2 verifies that all required support or supported features associated with the other redundant

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BASES (continued)

ACTIONS (continued)

channel(s) are OPERABLE within a Completion Time of 1 hour. The specified Completion Time is sufficient for plant operations personnel to make this determination. If verification determines loss of functional capability, LCO 3.0.3 must be immediately entered. However, if the support or supported feature LCO takes into consideration the loss-of-function situation, then LCO 3.0.3 may not need to be entered.]

SURVEILLANCE REQUIREMENTS

SR 3.3.14.1

SR 3.3.14.1 is the performance of the CHANNEL CHECK for the reactor building purge isolation--high radiation instrumentation once every 12 hours to ensure that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. Performance of the CHANNEL CHECK helps ensure the instrumentation continues to operate properly between each CHANNEL CALIBRATION. The high-radiation instrumentation should be compared to similar plant instruments located throughout the plant. If the radiation monitor employs keep-alive sources or check sources operable from the control room, the CHANNEL CHECK should also note the detector's response to these sources.

[For those facilities with only one radiation channel, the CHANNEL CHECK constitutes the following:]

Agreement criteria are determined by the plant staff, based on a combination of the channel instrument uncertainties, including isolation, indication, and readability. If a channel is outside the match criteria, it may be an indication that the transmitter or the signal-processing equipment has drifted outside its limit. If the channels are within the match criteria, it is an indication that the channels are OPERABLE. The surveillance interval, about once every shift, is based on operating experience that demonstrates the rarity of channel failure. Thus,

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.14.1 ensures that loss of function will be identified in 12 hours or less. [At this facility, the following administrative controls and design features (e.g., downscale alarms) immediately alert operators to loss of function:]

SR 3.3.14.2

SR 3.3.14.2 is the performance of a CHANNEL FUNCTIONAL TEST once every 31 days to ensure that the channels can perform their intended functions. This test verifies the capability of the instrumentation to provide the reactor building isolation. In MODES 1, 2, 3, and 4, the test does not include the actuation of the purge valves, as these valves are normally closed.

[For this facility the justification of a 31-day Frequency, in view of the fact that there is only one channel, is as follows:]

SR 3.3.14.3

SR 3.3.14.3 is the performance every 92 days of a CHANNEL CALIBRATION, with a setpoint ALLOWABLE VALUE of ≤ [25] mR/hr, to ensure that the instrument channel remains operational with the correct setpoint. The CHANNEL CALIBRATION is a complete check of the instrumentation and detector. While in MODES 1, 2, 3, and 4, the CHANNEL CALIBRATION does not include the actuation of the purge valves, since they are normally closed.

The Surveillance Frequency is based upon the assumption of a 92-day calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis. This test is a complete check of the process control instrument loop and the transmitter. Transmitter "as found" and "as left" values are recorded and used to verify drift assumptions. The radiation monitor may be calibrated in place or on a bench using test equipment, or it may be replaced by an equivalent, laboratory-calibrated unit. For OPERABLE channels, CHANNEL CALIBRATION shall find that measurement errors and bistable setpoint errors are within the assumptions of the plant-specific setpoint analysis. Recalibration restores OPERABILITY of an otherwise functional component that does not meet these criteria. However, repeated failures of the same channel over a

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

relatively small number of test intervals must be considered as potentially indicating a deterministic failure that cannot be corrected by recalibration. Completion of this test results in the channel being properly adjusted and expected to remain within the "as found" tolerance assumed by the setpoint analysis until the next scheduled surveillance. Measurement and setpoint error determination and readjustment must be performed consistent with the assumptions of the plant-specific setpoint analysis.

REFERENCES

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

2. Title 10, Code of Federal Regulation, Part 50.49 "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants."

3. [Unit Name],"[Plant-Specific Setpoint Methodology]."

B 3.3 INSTRUMENTATION

B 3.3.15 Control Room Isolation--High Radiation

BASES

BACKGROUND

The principal function of the Control Room Isolation--High Radiation is to provide an enclosed environment from which the plant can be operated following an uncontrolled release of radioactivity. The high-radiation isolation function provides assurance that under the required conditions, an isolation signal will be given. The noble-gas monitors located in the station vent stack provide isolation and shutdown of the normal control room ventilation system.

The control room isolation signal is provided by a single channel containing an iodine monitor with a scintillation detector and a gaseous monitor with a G-M detector. The iodine channel includes a particulate prefilter with the charcoal cartridge. Should a radioactivity concentration above normal background level be detected or if sampling capability is lost, the monitor will initiate a shutdown of the normal-duty supply fans and place the ventilation dampers in their recirculation mode.

Control room isolation also occurs on an Engineered Safety Feature Actuation System (ESFAS) reactor building high-pressure signal. ESFAS isolation of the control room is addressed by LCO 3.3.5, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," LCO 3.3.6, "Engineered Safety Feature Actuation System (ESFAS) Manual Initiation," and LCO 3.3.7, "Engineered Safety Feature Actuation System (ESFAS) Automatic Actuation Logic."

Trip Setpoints and ALLOWABLE VALUES

The trip setpoints are the nominal value at which the bistables are set. Any bistable is considered to be properly adjusted when the "as left" value is within the band for CHANNEL CALIBRATION accuracy, i.e., ± (rack calibration + comparator setting accuracy).

The trip setpoints used in the bistables are based on the analytical limits derived from the FSAR (Ref. 1). The selection of these trip setpoints is such that adequate

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BASES (continued)

BACKGROUND (continued)	protection is provided when all sensor and processing time delays are taken into account. To allow for calibration tolerances, instrumentation uncertainties, and instrument drift, ALLOWABLE VALUES specified in LCO 3.3.15 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 the plant-specific setpoint methodology (Ref. 2). The actual nominal trip setpoint entered into the bistable is more conservative than that specified by the ALLOWABLE VALUE, to account for changes in random measurement errors detectable by a CHANNEL FUNCTIONAL TEST. One example of such a change in measurement error is drift during the surveillance interval. If the measured setpoint does not exceed the ALLOWABLE VALUE, the bistable is considered OPERABLE.
APPLICABLE SAFETY ANALYSES	The Control Room Ventilation System is isolated upon receipt of a reactor building high-pressure ESFAS signal or a high-radiation signal. For the first 4 days following a loss-of-coolant accident (LOCA), the Control Room Emergency Ventilation System (CREVS) is operated in the total recirculation mode. Four days after the start of the accident, the CREVS is started in the intake and recirculation mode and continues to operate in this mode for 30 days. This intake slightly pressurizes the control room. In both cases, the air flows through charcoal filters that are 95% efficient for elemental, particulate, and organic materials. The high-radiation function only performs the initial isolation function to begin the recirculation mode of operation. [For this facility, the control room isolation signal provides protection for events other than LOCAs as follows:] The Control Room Isolation--High Radiation satisfies Criterion 3 of the NRC Interim Policy Statement.
LCO	One channel of Control Room Isolation--High Radiation shall be OPERABLE at all times to allow for automatic isolation and transfer to the recirculation mode. Control Room

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BASES (continued)

LCO (continued)

Isolation--High Radiation is OPERABLE when the following criteria are met:

- All channel components necessary to provide a control room isolation signal are functional and in service;
- Channel measurement uncertainties are known (via test, analysis, or design information) to be within the assumptions of the setpoint calculations; and
- Required surveillance testing is current and has demonstrated performance within each surveillance test's acceptance criteria.

Only the ALLOWABLE VALUE is specified for each Control Room Isolation--High Radiation trip function in the LCO. Nominal trip setpoints are specified in the plant-specific setpoint calculations. The nominal setpoints are selected to ensure the setpoint measured by the CHANNEL FUNCTIONAL TEST does not exceed the ALLOWABLE VALUE if the bistable 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 is consistent with the assumptions of the plant-specific setpoint calculations. Each ALLOWABLE VALUE specified is more conservative than the analytical limit assumed in the safety analysis in order to account for instrument uncertainties appropriate to the trip function. These uncertainties are defined in the plant-specific setpoint methodology (Ref. 2).

[At this facility, the basis for the ALLOWABLE VALUE is as follows:]

[For this facility, the following support systems are required OPERABLE to ensure Control Room Isolation--High Radiation instrumentation OPERABILITY:]

[For this facility, those required support systems which, upon their failure, do not result in the Control Room Isolation--High Radiation instrumentation being declared inoperable and their justification are as follows:]

[For this facility, the supported systems impacted by the inoperability of Control Room Isolation--High Radiation

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BASES (continued)

LCO (continued)	instrumentation and the justification of whether or not each supported system is declared inoperable are as follows:]

APPLICABILITY

The control room isolation capability on high radiation shall be OPERABLE whenever there is a chance for an accidental release of radioactivity. This includes MODES 1, 2, 3, and 4 and all MODES and conditions when there is movement of irradiated fuel or loads over irradiated fuel. If a radioactive release were to occur during any of these conditions, the control room would have to remain habitable to ensure reactor shutdown and cooling can be controlled from the main control room.

ACTIONS

A Control Room Isolation--High Radiation channel is inoperable when it does not satisfy the OPERABILITY criteria for the channel's function. These criteria are outlined for each function in the LCO section of the Bases. The most common cause of channel inoperability is outright failure or drift of the bistable or process module sufficient to exceed the tolerance allowed by the plant-specific setpoint 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 setpoint drift is generally made during the performance of a CHANNEL FUNCTIONAL TEST when the process instrument is set up for adjustment to bring it within specification. If the trip setpoint is less conservative than the ALLOWABLE VALUE, the channel must be declared inoperable immediately and Condition A entered immediately.

Condition A

Condition A applies to failure of the Control Room Isolation--High Radiation function in MODE 1, 2, 3, or 4.

With one channel of Control Room Isolation--High Radiation inoperable, the inoperable channel must be restored to OPERABLE status, or the Control Room Ventilation System must be placed in a condition which does not require the isolation to occur. The Required Action A.1, which places one OPERABLE train of the Control Room Emergency Air

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BASES (continued)

ACTIONS (continued)

Temperature Control System (CREHVAC) in the emergency recirculation mode of operation, ensures that the ventilation system has been placed in a state equivalent to that which occurs after the high-radiation isolation has occurred. Reactor operation can continue indefinitely in this state. The 1-hour Completion Time is a sufficient amount of time in which to take the Required Action and credits the availability of ESFAS initiation signals for control room isolation.

Condition B

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

If the inoperable channel cannot be restored and CREHVAC cannot be placed into recirculation mode while in MODE 1, 2, 3, or 4, actions must be taken to minimize the chances of an accident that could lead to radiation releases. The plant must be placed in at least MODE 3 within 6 hours, with a subsequent cooldown to MODE 5 within 36 hours. This places the reactor in a low-energy state, which allows greater time for operator action if habitation of the control room is precluded. The allowed Completion Times are reasonable, based on operating experience, to reach the required MODE from full power in an orderly manner and without challenging plant systems.

Condition C

Condition C applies to failure of the Control Room Isolation--High Radiation function during CORE ALTERATIONS and during movement of irradiated fuel or loads over irradiated fuel.

Required Action B.1 and Required Action B.2 are the same as discussed above for Condition A (except for Completion Time). If the Control Room Ventilation System cannot be placed into recirculation mode during CORE ALTERATIONS or while moving irradiated fuel or loads over irradiated fuel, Required Action B.3.1, Required Action B.3.2, and Required Action B.3.3 suspend actions that could lead to an accident which could release radioactivity resulting from a fuel-handling accident.

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BASES (continued)

ACTIONS (continued)

These actions place the core in a safe and stable configuration, in which it is less likely to experience an accident that could result in a release of radioactivity. Additionally, the movement of fuel assemblies or loads near the fuel storage area precludes the accidental dropping of a fuel assembly or a load onto irradiated fuel. By placing the reactor in a stable configuration, and stopping the movement of fuel or loads near irradiated fuel, the potential for a fuel-handling accident is minimized. The reactor must be maintained in these conditions until the automatic isolation capability is returned to operation, or manual action places one train of the CREHVAC into the emergency recirculation mode. The Completion Times of "Immediately" for these actions are consistent with the urgency of the situation and accounts for the fact that the High Radiation function provides the only automatic control room isolation function capable of responding to radiation release due to a fuel-handling accident. The Completion Time does not preclude placing any load or fuel assembly into a safe position before ceasing any such movement.

Note that in certain circumstances, such as fuel handling in the fuel building during power operation, both Conditions A and B may apply in the event of channel failure.

Required Action C.1 verifies that the Required Actions have been initiated for those supported systems declared inoperable because of the inoperability of the support channel[(s)] within a Completion Time of 1 hour. The specified Completion Time is sufficient for plant operations personnel to make this determination.

Required Action C.1 ensures that those identified Required Actions associated with supported systems impacted by the inoperability of channel[(s)] associated with the control room isolation function have been initiated. This can be accomplished by entering the supported systems' LCOs or independently as a group of Required Actions that need to be initiated every time Condition C is entered.

[For this facility, the identified supported systems' Required Actions associated with the control room isolation function are as follows:]

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BASES (continued)

ACTIONS (continued)

[Required Action C.2 verifies that all required support or supported features associated with the other redundant channel(s) are OPERABLE within a Completion Time of 1 hour.

The specified Completion Time is sufficient for plant operations personnel to make this determination. If verification determines loss of functional capability, LCO 3.0.3 must be immediately entered. However, if the support or supported feature LCO takes into consideration the loss-of-function situation, then LCO 3.0.3 may not need to be entered.]

SURVEILLANCE REQUIREMENTS

SR 3.3.15.1

SR 3.3.15.1 is the performance of the CHANNEL CHECK for the Control Room Isolation--High Radiation actuation instrumentation once every 12 hours to ensure that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. Performance of the CHANNEL CHECK helps ensure that the instrumentation continues to operate properly between each CHANNEL CALIBRATION. The high-radiation instrumentation should be compared to similar plant instruments located throughout the plant. If the radiation monitor uses keep-alive sources or check sources operated from the control room, the CHANNEL CHECK should also note the detectors response to these sources.

[For those facilities with only one radiation channel, the CHANNEL CHECK constitutes the following:]

Agreement criteria are determined by the plant staff, based on a combination of the channel instrument uncertainties, including isolation, indication, and readability. If a channel is outside the match criteria, it may be an indication that the transmitter or the signal-processing

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

equipment has drifted outside its limit. If the channels are within the match criteria, it is an indication that the channels are OPERABLE. The Surveillance Frequency, about once every shift, is based on operating experience that demonstrates the rarity of channel failure. Thus, SR 3.3.14.1 ensures that loss of function will be identified in no more than 12 hours. [At this facility the following administrative controls and design features (e.g., downscale alarms) immediately alert operators to loss of function:]

SR 3.3.15.2

SR 3.3.15.2 is the performance of a CHANNEL FUNCTIONAL TEST once every 31 days to ensure that the channels can perform their intended functions. This test verifies the capability of the instrumentation to provide the automatic control room isolation.

[For this facility, the justification of a 31-day Frequency, in view of the fact that there is only one channel, is as follows:]

SR 3.3.15.3

SR 3.3.15.3 is the performance every [18] months of a CHANNEL CALIBRATION with a setpoint ALLOWABLE VALUE of ≤ [25] mR/hr to ensure that the instrument channel remains operational with the correct setpoint.

A CHANNEL CALIBRATION is performed every [18] months, or approximately every refueling. The Surveillance Frequency is based upon the assumption of an [18]-month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis. This test is a complete check of the process control instrument loop and the transmitter. Transmitter "as found" and "as left" values are recorded and used to verify drift assumptions. The radiation monitor may be calibrated in place or on a bench using test equipment or it may be replaced by an equivalent, laboratory-calibrated unit. For OPERABLE channels, CHANNEL CALIBRATION shall find that measurement errors and bistable setpoint errors are within the assumptions of the plant-specific setpoint analysis. Recalibration restores OPERABILITY of an otherwise functional component that does not meet these criteria. However, repeated failures of the

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)	same channel over a relatively small number of test intervals must be considered as potentially indicating a deterministic failure that cannot be corrected by recalibration. Completion of this test results in the channel being properly adjusted and expected to remain within the "as found" tolerance assumed by the setpoint analysis until the next scheduled surveillance. Measurement and setpoint error determination and readjustment must be performed consistent with the assumptions of the plant-specific setpoint analysis.

REFERENCES

1. [Unit Name] FSAR, Section [15], "[Accident Analysis]."
2. [Unit Name], "[Plant-Specific Setpoint Methodology Analysis]."

B 3.3 INSTRUMENTATION

B 3.3.16 Post-Accident Monitoring (PAM) Instrumentation

BASES

BACKGROUND

Indications of plant variables are required by the control room operating personnel during accident situations to:

1. Provide information required to permit the operator to take preplanned manual actions to accomplish safe plant shutdown;

2. Determine whether the Reactor Protection System (RPS), Engineered Safety Feature Systems, manually initiated safety systems, and other systems important to safety are performing their intended functions (i.e., reactivity control, core cooling, maintaining Reactor Coolant System (RCS) integrity, and maintaining reactor building (RB) OPERABILITY);

3. Provide information to the operators that will enable them to determine the potential for causing a gross breach of the barriers to radioactivity release (i.e., fuel cladding, reactor coolant pressure boundary, and RB) and to determine if a gross breach of a barrier has occurred; and

4. 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.

At the start of an accident, it may be difficult for the operator to determine immediately what accident has occurred or is occurring and therefore to determine the appropriate response. For this reason, reactor trip and certain other safety actions (e.g., emergency core cooling actuation, RB isolation, or depressurization) have been designed to be performed automatically during the initial stages of an accident. Instrumentation is also provided to indicate information about plant variables required to enable the operation of manually initiated safety systems and other appropriate operator actions involving systems important to safety.

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BACKGROUND (continued)

Independent of the above tasks, it is important that operators be informed if the barriers to the release of radioactive materials are being challenged. Therefore, PAM instrument ranges are selected so that the instrument will always be on scale. Instruments that are not part of the PAM System may provide limited backup capability, but they may not have the necessary range to track the course of the accident; consequently, multiple instruments with overlapping ranges may be necessary. It is essential that degraded conditions and their magnitude be identified so the operators can take actions that are available to mitigate the consequences. It is not intended that operators be encouraged to prematurely circumvent systems important to safety, but that they be adequately informed so that unplanned actions can be taken when necessary.

Examples of serious events that could threaten safety if conditions degrade are loss-of-coolant accidents (LOCAs), overpressure transients, anticipated operational occurrences that become accidents such as anticipated transients without scram, and reactivity excursions that result in releases of radioactive materials. Such events require that the operators understand, within a short time period, the ability of the barriers to limit radioactivity release (i.e., that they understand the potential for breach of a barrier or whether an actual breach of a barrier has occurred because of an accident in progress).

It is essential that the required instrumentation be capable of surviving the accident environment in which it is located for the length of time its function is required. It is, therefore, either designed to withstand the accident environment or to be protected by a local environmental barrier.

Variables for accident monitoring are selected to provide the essential information needed by the operator to determine if the plant safety functions are being performed. The availability of such instrumentation is important so that responses to corrective actions can be observed and the need for, and magnitude of, further actions can be determined. These essential instruments are identified by plant-specific documents (Ref. 1) addressing the recommendations of Regulatory Guide 1.97 (RG 1.97) as required by Supplement 1 to NUREG 0737 (Ref. 3).

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BASES (continued)

BACKGROUND (continued)

The instrument channels required to be OPERABLE by this LCO equate to two classes of parameters identified during plant-specific implementation of Regulatory Guide 1.97 as Type A variables and Category 1 variables.

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 and that are required for safety systems to accomplish their safety functions for Design Basis Accidents (DBAs). Primary information is information that is essential for the direct accomplishment of the specified safety functions; it does not include those variables that are associated with contingency actions that may also be identified in written procedures. Because the list of Type A variables widely differs between plants, Table 3.3.16-1 contains no examples of Type A variables, except for those that may also be Category 1.

Category 1 variables are the key variables deemed risk significant because they are needed to:

- Determine whether 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 Regulatory Guide 1.97 analyses (Ref. 1). These analyses identified the plant-specific Type A variables and provided justification for deviating from the NRC proposed list of Category 1 variables. Table 3.3.16-1 provides a list of variables typical of those identified by plant-specific Regulatory Guide 1.97 analyses. [Table 3.3.16-1, in

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BASES (continued)

BACKGROUND (continued)

plant-specific Technical Specifications shall list all Type A and Category 1 variables identified by the plant-specific Regulatory Guide 1.97 analysis, as amended by the NRC's Safety Evaluation Report (SER).]

Type A and Category 1 variables are required to meet Regulatory Guide 1.97 Category 1 (Ref. 2) design and qualification requirements for seismic and environmental qualification, single failure criterion, utilization of emergency standby power, immediate accessible display, continuous readout, and recording of display.

Listed below are discussions of the specified instrument functions listed in Table 3.3.16-1. These discussions are intended as examples of what should be provided for each function when the plant-specific list is prepared.

1. [Wide Range] Neutron Flux

[Wide Range] neutron flux indication is provided to verify reactor shutdown. [For this facility, the wide range neutron flux channels consist of the following:]

2, 3. RCS Hot and Cold Leg Temperature

RCS hot and cold leg temperature instrumentation are Category 1 variables provided for verification of core cooling and long-term surveillance.

Reactor outlet temperature inputs to the RPS are provided by two fast-response resistance elements and associated transmitters in each loop. The channels provide indication over a range of 32°F to 700°F.

4. RCS Pressure (Wide Range)

RCS wide-range pressure instrumentation is a Category 1 variable provided for verification of core cooling and RCS integrity long-term surveillance.

Wide range RCS loop pressure is measured by pressure transmitters with a span of 0 to 3000 psig. The

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BASES (continued)

BACKGROUND (continued)

pressure transmitters are located outside the RB. Redundant monitoring capability is provided by two trains of instrumentation. Control room indications are provided through the inadequate core cooling plasma display. The inadequate core cooling plasma display is the primary indication used by the operator during an accident. Therefore, the accident monitoring specification deals specifically with this portion of the instrument string.

In some plants RCS pressure is a Type A variable because the operator uses this indication to monitor the cooldown of the RCS following a steam generator tube rupture or small-break LOCA. Operator actions to maintain a controlled cooldown, such as adjusting steam generator (SG) pressure or level, would use this indication. In addition, high pressure injection (HPI) flow is throttled based on RCS pressure and subcooled margin. For some small-break LOCAs, low pressure injection (LPI) may actuate with system pressure stabilizing above the shutoff head of the LPI pumps. If this condition exists, the operator is instructed to verify HPI flow and then terminate LPI flow prior to exceeding 30 minutes of LPI pump operation against a deadhead pressure. RCS pressure, in conjunction with LPI flow, is also used to determine if a core flood line break has occurred. If one LPI train indicates no flow, the other train indicates excessive flow, and system pressure is above the shutoff head of the LPI pumps, a core flood line break has probably occurred. The operator is instructed to isolate the LPI train with excessive flow. In addition, if a single failure has occurred in the other train, it may be necessary for the operator to cross-connect the trains.

5. Reactor Vessel Water Level

Reactor vessel water level instrumentation is provided for verification and long-term surveillance of core cooling.

The reactor vessel level monitoring system provides a direct measurement of the collapsed liquid level above the fuel alignment plate. The collapsed level

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BASES (continued)

BACKGROUND (continued)

represents the amount of liquid mass which 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 which may occur during the preceding core recovery interval.

The level range extends from the top of the vessel down to 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.

6. Containment Sump Water Level (Wide Range)

Containment Sump Water Level instrumentation is provided for verification and long-term surveillance of RCS integrity. [For this facility, the containment sump PAM instrumentation consists of the following:]

7. Containment Pressure (Wide Range)

Containment Pressure instrumentation is provided for verification of RCS and containment OPERABILITY. [For this facility, containment pressure instrumentation consists of the following:]

8. Containment Isolation Valve Position

Containment Isolation Valve Position instrumentation is provided for verification of OPERABILITY. [For this facility, the containment Isolation Valve Position consists of the following:]

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BASES (continued)

BACKGROUND (continued)

9. Containment Area Radiation (High Range)

Containment Area Radiation instrumentation is provided to monitor for the potential of significant radiation releases, and to provide release assessment for use by operators in determining the need to invoke site emergency plans. [For this facility, the Containment Area Radiation instrumentation consists of the following:]

10. Containment Hydrogen Concentration

Containment Hydrogen Concentration instrumentation is provided to detect high hydrogen concentration conditions which represent a potential for containment breach. This variable is also important in verifying the adequacy of mitigating actions. [For this facility, the containment area hydrogen instrumentation consists of the following:]

11. Pressurizer Level

Pressurizer Level instrumentation is used to determine whether to terminate safety injection (SI), if still in progress, or to reinitiate SI 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. [For this facility, the pressurizer level instrumentation consists of the following:]

12. Steam Generator Water Level

SG Water Level instrumentation is provided to monitor operation of decay heat removal via the SG. The Category 1 indication of SG level is the extended startup range level instrumentation. The extended startup range level covers a span of 6 to 394 inches above the lower tubesheet. The measured differential pressure is displayed in inches of water at 68°F. Temperature compensation of this indication is performed manually by the operator. Redundant monitoring capability is provided by two trains of instrumentation. The uncompensated level signal is

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BASES (continued)

BACKGROUND (continued)

input to the plant computer, a control room indicator, and the Emergency Feedwater (EFW) Control System.

At some plants, operator action is based on the control room indication of SG 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 SG level to establish boiler-condenser heat transfer. Operator action is initiated on a loss of subcooled margin. Feedwater flow is increased until the indicated extended startup range level reaches the boiler-condenser setpoint.

13. Condensate Storage Tank (CST) Level

CST Level instrumentation is provided to ensure a water supply for EFW. The CST provides the assured, safety-grade water supply for the EFW System. The CST consists of two identical tanks connected by a common outlet header. Inventory is monitored by a 0 to 144 inch level indication for each tank. CST level is displayed on a control room indicator, strip chart recorder, and plant computer. In addition, a control room annunciator alarms on low level.

At some plants CST 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 EFW are the loss of electric power, steam line break (SLB), and small-break LOCA. The CST is the initial source of water for the EFW System. However, as the CST is depleted, manual operator action is necessary to replenish the CST or align suction to the EFW pumps from the hotwell.

14, 15, 16, 17. Core Exit Temperature

Core Exit Temperature is provided for verification and long term surveillance of core cooling.

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BASES (continued)

BACKGROUND (continued)

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 recovery and trend the ensuing core heatup. The evaluations account for core nonuniformities 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 assured 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 56 incore instrument detector assemblies. The junction of each thermocouple is located a few inches above the fuel assembly inside a structure which 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 exists the fuel assemblies.

The core exit thermocouples have a usable temperature range from 32°F to up to 2300°F, although accuracy is reduced at temperatures above 1800°F.

18. Emergency Feedwater Flow

EFW Flow instrumentation is provided to monitor operation of decay heat removal via the SGs.

The EFW Flow to each SG is determined from a differential pressure measurement calibrated to a span of 0 to 1200 gpm. Redundant monitoring capability is provided by two independent trains of instrumentation for each SG. Each differential pressure transmitter provides an input to a control room indicator and the plant computer. Since the primary indication used by the operator during an accident is the control room indicator the accident monitoring specification deals specifically with this portion of the instrument channel.

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BASES (continued)

BACKGROUND (continued)

At some plants, EFW Flow is a Type A variable because operator action is required to throttle flow during a an SLB accident in order to prevent the EFW pumps from operating in runout conditions. EFW Flow is also used by the operator to verify that the EFW System is delivering the correct flow to each SG. However, the primary indication used by the operator to ensure an adequate inventory is SG level.

APPLICABLE SAFETY ANALYSES

Design Basis Definition

The PAM instrumentation ensures the OPERABILITY of Regulatory Guide 1.97 Type A and Category 1 variables so that the control room operating staff can:

- Perform the diagnosis required to support preplanned actions for the primary success path of Design Basis Accidents (DBAs);
- 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;
- 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 and for an estimate of the magnitude of any impending threat.

These functions support the requirements of GDC 13 and GDC 19 of 10 CFR 50, Appendix A (Ref. 4). The plant-specific Regulatory Guide 1.97 analysis documents the process that identified Type A and Category 1 variables.

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

The PAM instrumentation that satisfies the definition of Type A in Regulatory Guide 1.97 also satisfies Criterion 3 of the NRC Interim Policy Statement.

Category 1 PAM instrumentation is retained in Technical Specifications because it is intended to assist operators in minimizing the consequences of accidents. Therefore, Category 1 variables are important in reducing public risk.

LCO

The PAM instrumentation LCO provides the requirement of Type A and Category 1 monitors that provide information required by the control room operators to:

- Permit the operator to take preplanned manual actions to accomplish safe plant shutdown;
- Determine whether 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 to determine if a gross breach of a barrier has occurred; 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.

Two channels shall be OPERABLE for most functions. Two OPERABLE channels ensure no single failure within either of the PAM instrumentation, its auxiliary supporting features, or its power sources concurrent with the failures that are a condition of, or result from, a specific accident 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 channels allows a CHANNEL CHECK during the post-accident phase to confirm the validity of displayed information. More than two channels may be

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BASES (continued)

LCO (continued)

required at some plants, if the Regulatory Guide 1.97 analysis determined that failure of one accident monitoring channel results in information ambiguity (that is, the redundant displays disagree) that could lead operators to defeat or fail to accomplish a required safety function.

This might also be accomplished by providing an independent channel to monitor a different variable that bears a known relationship to the multiple channels (addition of a diverse channel).

In Table 3.3.16-1, the exceptions to the two channel requirement are core exit temperatures, loop- and SG-related variables and RB isolation valve position.

Two OPERABLE channels of core exit temperature 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. 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 ensure that a single failure will not disable the ability to determine the representative core exit temperature in each quadrant.

For loop- and SG-related variables, the required information is individual loop temperature and individual SG level. In these cases, two channels are required OPERABLE for each loop of SG to redundantly provide the necessary information.

In the case of RB isolation valve position, the important information is the status of the RB penetrations. The LCO requires one position indicator for each active RB isolation valve. 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 passive valve or system boundary status. If a normally active RB isolation valve is known to be closed and deactivated, position indication is not needed to determine

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BASES (continued)

LCO (continued)

status. Therefore, the position indication for valves in this state is not required to be OPERABLE.

A PAM channel is OPERABLE when:

- All channel components necessary to provide the required indication are functional;
- Channel measurement uncertainties are known via test, analysis, or design information to be sufficiently small such that measurement and indication errors will not mislead operators into actions that would challenge plant Safety Limits (SLs); and
- Required surveillance testing is current and has demonstrated performance within each surveillance test's acceptance criteria.

LCO Table 3.3.17-1 is for illustration purposes only. Plant-specific Technical Specifications tables will list all Type A and Category 1 variables identified by the plant's Regulatory Guide 1.97 analysis as amended by NRC's plant-specific SER.

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

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

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 which would require PAM instrumentation is low; therefore, the PAM instrumentation is not required to be OPERABLE in these MODES.

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BASES (continued)

APPLICABILITY (continued)

A Note indicates that the provisions of LCO 3.0.4 are not applicable to the functions contained in the LCO. For this LCO, a second Note added in Applicability provides clarification that each function specified in Table 3.3.16-1 is treated as an independent entity with an independent Completion Time.

ACTIONS

Condition A

When one required channel in one of more functions is inoperable, each inoperable channel must be restored to OPERABLE status within 30 days. In some channels, it may be possible to have one PAM channel inoperable, but still have all required channels OPERABLE. For example, some plants have four equivalent channels available to perform certain PAM functions. In these cases, the failure of a one or two of the channels leaves at least two channels OPERABLE to meet the LCO requirements. Therefore, for this example, Condition A need not be entered unless three channels fail. The 30-day Completion Time is based on operating experience and takes into account the remaining OPERABLE channel and the low probability of an event requiring PAM instrumentation during this interval.

Condition B

With two required channels inoperable in one or more functions, at least one channel in each 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 instrument operation and the availability of alternate means to obtain the required information. Continuous operation with two required channels inoperable is not acceptable because the alternate indications may not fully meet all performance of qualification requirements applied to the PAM instrumentation. Therefore, requiring restoration of at least one inoperable channel limits the risk that the PAM function will be in a degraded condition should an accident occurs.

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BASES (continued)

ACTIONS (continued)

Condition C

Required Action C.1 directs the operator to follow the directions given in Table 3.3.16-1 for entering either Condition D or E immediately.

Condition D

For the majority of functions in Table 3.3.16-1, if the Required Actions and associated Completion Times of Condition A or B are not met, then the plant must be placed in a MODE in which the LCO requirements are not applicable. This is done by placing the plant in MODE 3 within 6 hours and MODE 4 within 12 hours. The Completion Times are reasonable, based on operating experience, to reach the required MODES from full power in an orderly manner and without challenging plant systems.

Condition E

At this facility, alternate means of monitoring [reactor vessel water level and RB Area Radiation] have been developed and tested. These alternate means may be temporarily installed if the normal PAM channel(s) cannot be restored to OPERABLE status within the allotted time. If these alternate means are invoked, the Required Action is not to shut the plant down but rather to follow the directions of Specification 5.9.2, "Special Reports," in the Administrative Controls section of the Technical Specifications. The report provided to the NRC should discuss the alternate means invoked, 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.

[At this facility, the alternate monitoring provisions consist of the following:]

SURVEILLANCE REQUIREMENTS

The following SRs apply to each PAM Instrumentation function in Table 3.3.16-1.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.16.1

Performance of the CHANNEL CHECK once every 31 days ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or even something more serious. CHANNEL CHECK will detect gross channel failure, thus it is the key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION. The high radiation instrumentation should be compared to similar plant instruments located throughout the plant. If the radiation monitor employs keep-alive sources or check sources operable from the control room, the CHANNEL CHECK should also note the detector's response to these sources. Agreement criteria are determined by the plant staff, based on a combination of the channel instrument uncertainties, including isolation, indication, and readability. If a channel is outside the match criteria, it may be an indication that the transmitter or the signal processing equipment has drifted outside its limit. If the channels are within the match criteria, it is an indication that the channels are OPERABLE. If the channels are normally off scale during times when surveillance 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 downscale.

The Surveillance Frequency of 31 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 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 required channels.

SR 3.3.16.2

A CHANNEL CALIBRATION is performed every [18] months or approximately every refueling. CHANNEL CALIBRATION is a

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

complete check of the instrument channel including the detector. The test verifies the channel responds to measured parameter with the necessary range and accuracy. For OPERABLE channels, CHANNEL CALIBRATION shall find measurement errors are sufficiently small such that measurement and indication errors will not mislead operators into actions that would challenge plant SLs.

Transmitter "as found" and "as left" values are recorded and used to verify drift assumptions. Field transmitters may be calibrated in place, removed and calibrated in a laboratory, or replaced with an equivalent laboratory-calibrated unit.

Resistance temperature detector (RTD) and thermocouple channels may be calibrated in place using cross-calibration techniques, or in a test bath after removal from piping. For cross calibration, at least one sensor should be replaced with a newly calibrated sensor during each refueling cycle to ensure accurate cross calibration. This replacement sensor must be the same model as the remaining RTDs or thermocouples. Using a newly calibrated sensor as a reference assures signal drift continues to remain random rather than systematic and is within the limits specified in the plant setpoint analysis. The replacement interval may be extended to alternate refuelings if it is demonstrated that over the extended interval, the drift is random rather than systematic. This determination may use results of statistical analysis of operating data and calibration data from similar plants using the same model of RTD or thermocouple in the same environmental conditions.

Recalibration restores operability of an otherwise functional component that does not meet these criteria. However, repeated failures of the same channel over a relatively small number of test intervals must be considered as potentially indicating a deterministic failure that cannot be corrected by recalibration.

For the RB area radiation instrumentation, a CHANNEL CALIBRATION may consist of an electronic calibration of the channel, not including the detector, for range decades above 10 R/hr, and a one-point calibration check of the detector below 10 R/hr with a gamma source.

The Surveillance Frequency is based upon the assumption of an [18]-month calibration interval in the determination of the magnitude of equipment drift.

(continued)

BASES (continued)

REFERENCES

1. [Plant-specific document (e.g., FSAR, NRC Regulatory-Guide 1.97 SER letter).]

2. U.S. NRC Regulatory Guide 1.97, "Instrumentation for Light-Water Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident."

3. U.S. NRC NUREG-0737, Supplement 1, "Clarification of TMI Action Plan Requirements."

4. Title 10, Code of Federal Regulations, Part 50, Appendix A:
General Design Criterion 13, "Instrumentation and Control"; and
General Design Criterion 19, "Control Room."

B 3.3 INSTRUMENTATION

B 3.3.17 Remote Shutdown System

BASES

BACKGROUND

The Remote Shutdown System provides the control room operator with sufficient instrumentation and controls to place and maintain the facility in a safe shutdown condition from locations other than the control room. The capability is necessary to protect against the possibility that the control room becomes inaccessible or that a fire destroys all equipment in one facility area disabling critical control room instruments or controls. A safe shutdown condition is defined as MODE 3. With the facility in MODE 3, the Emergency Feedwater (EFW) System and the steam generator (SG) safety valves or the SG atmospheric dump valves can be used to remove core decay heat and meet all safety requirements. The long-term supply of water for the EFW and the ability to borate the Reactor Coolant System (RCS) from outside the control room allows extended operation in MODE 3.

In the event that the control room becomes inaccessible, or a fire disables critical control or display functions in the control room, the operators can establish control at the remote shutdown panel and place and maintain the facility in MODE 3. Not all controls and necessary transfer switches are located at the remote shutdown panel. Some controls and transfer switches will have to be operated locally at the switchgear, motor control panels, or other local stations. The facility automatically reaches MODE 3 following a facility shutdown and can be maintained safely in MODE 3 for an extended period of time.

The OPERABILITY of the Remote Shutdown System control and 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 or critical control room displays or controls become unavailable.

APPLICABLE SAFETY ANALYSES

The Remote Shutdown System is required to provide equipment at appropriate locations outside the control room

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

with a capability to promptly shut down and maintain the plant in a safe condition in MODE 3.

Furthermore, in the event of a fire in any one plant fire area, the Remote Shutdown System is designed to ensure one train of systems necessary to achieve and maintain MODE 3 conditions from either the control room or emergency control station(s). The criteria governing the design of the Remote Shutdown System are 10 CFR 50, Appendix A, GDC 19 (Ref. 1) and 10 CFR 50 Appendix R (Ref. 2).

Specific system requirements are presented in Reference 1 and the NRC staff-approved plant-specific fire protection topical report.

The Remote Shutdown System is considered an important contributor to the reduction of plant risk or accidents, and as such it has been retained in the Technical Specifications as indicated in the NRC Interim Policy Statement.

LCO

The Remote Shutdown System LCO provides the requirements for the OPERABILITY of the instrumentation and controls necessary to place and maintain the plant in MODE 3 from a location other than the control room. The instrumentation and controls typically required are listed in Table B 3.3.17-1 or page B 3.3-9. For Remote Shutdown System channels that support only the functions required by 10 CFR 50, Appendix R (Ref. 2), one division is required to be OPERABLE.

For channels that fulfill GDC 19 requirements, the number of OPERABLE channels required depends upon the plant licensing basis as described in the NRC plant-specific Safety Evaluation Report (SER). Generally, two divisions are required OPERABLE. However, only one channel is required if the plant has justified such a design and the NRC's SER accepted the justification. The controls, instrumentation, and transfer switches are those required for:

- Core reactivity control (initial and long term);
- RCS pressure control;
- Decay heat removal via the EFW System and the SG safety valves or SG atmospheric dump valves;

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BASES (continued)

LCO (continued)

- RCS inventory control via charging flow; and
- Safety support systems for the above functions, including service water, component cooling water, and onsite power, including the diesel generators.

A division of a Remote Shutdown System is OPERABLE if all instrument and control channels needed to support the function are OPERABLE in that division. In some cases, Table B 3.3.17-1 may indicate that the required information or control 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 or control sources is OPERABLE.

Remote Shutdown System instrumentation channels are OPERABLE when:

- All channel components necessary to provide the required indication are functional;
- Channel measurement uncertainties are known via test, analysis, or design information to be sufficiently small such that measurement and indication errors will not mislead operators into actions that would challenge plant Safety Limits or prevent prompt entry into MODE 3; and
- Required surveillance testing is current and has demonstrated performance within each Surveillance test's acceptance criteria.

Remote Shutdown System controls are OPERABLE when:

- All channel components, including transfer switches necessary to provide remote shutdown control are functional; and
- Required surveillance testing is current and has demonstrated performance within each surveillance test's acceptance criteria.

The remote shutdown equipment covered by this LCO does not need to be in operation to be considered OPERABLE. This LCO is intended to ensure the equipment will be OPERABLE if

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BASES (continued)

LCO (continued)

plant conditions require that the Remote Shutdown System be placed in operation.

For this facility, a Remote Shutdown System division is considered OPERABLE when the plant-specific instrumentation, controls, transfer switches, and support systems listed in Table B 3.3.17-1 are OPERABLE.

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

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

APPLICABILITY

The Remote Shutdown System LCO is applicable in MODES 1, 2, and 3. This is required so that the facility can be placed and maintained in MODE 3 for an extended period of time from a location other than the control room.

This LCO is not applicable in MODE 4, 5, or 6. In these MODES, the facility is already subcritical and in a condition of reduced RCS energy. Under these conditions considerable time is available to restore necessary instrument and control functions if control room instruments or control become unavailable.

A Note has been added to indicate that LCO 3.0.4 does not apply to the Remote Shutdown System LCO. This exception to LCO 3.0.4 allows normal startup during the period when the Remote Shutdown System is inoperable. Normal startup may proceed while in Condition A because the justification for Condition A action and completion time are equally applicable to startup conditions as to continued operation in MODE 1, 2, or 3. Furthermore, Remote Shutdown System equipment can generally be repaired during operation without significant risk of spurious trip.

(continued)

BASES (continued)

ACTIONS

Condition A

Condition A addresses the situation where one or more divisions of any required function(s) or channel(s) of a function of the Remote Shutdown System is inoperable. This includes any function listed in Table B 3.3.17-1 as well as the control and transfer switches.

When a division includes a function which only requires a channel to be OPERABLE, the failure of the single channel constitutes the failure of the function and as a consequence, the division becomes inoperable.

The Required Action is to restore the division to OPERABLE status within 30 days. The Completion Time is based on operating experience and takes into account the remaining OPERABLE division and the low probability of an event that would require evacuation of the control room.

For this LCO, a Note has been added in Applicability to provide clarification that each [division] is treated as an independent entity with an independent Completion Time.

Condition B

If the inoperable division cannot be restored to OPERABLE status in 30 days, the prudent action is to place the plant in a MODE in which the LCO requirements are not applicable. This is done by placing the plant in at least MODE 3 within 6 hours and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience and normal cooldown rates, to reach the required MODES from full power in an orderly manner and without challenging safety systems.

SURVEILLANCE REQUIREMENTS

The following SRs apply to each Remote Shutdown System division.

SR 3.3.17.1

Performance of the CHANNEL CHECK once every 31 days ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or even something more serious. CHANNEL CHECK will detect gross channel failure, thus it is the key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION. Remote Shutdown System instrumentation should be compared to similar plant instruments located in the control room. Agreement criteria are determined by the plant staff, based on a combination of the channel instrument uncertainties, including isolation, indication, and readability. If a channel is outside the match criteria, it may be an indication that the transmitter or the signal processing equipment has drifted outside its limit. If the channels are within the match criteria, it is an indication that the channels are OPERABLE. If the channels are normally off scale during times when surveillance 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 downscale.

The Surveillance Frequency of 31 days is based on plant operating experience with regard to channel OPERABILITY and drift which demonstrates that the probability of failure of more than one channel of a given function in any 31-day interval is a rare event. The CHANNEL supplements less formal, but more frequent checks of channel during normal operational use of the displays associated with this LCO required channels.

SR 3.3.17.2

SR 3.3.17.2 verifies that each required Remote Shutdown System transfer switch and control circuit perform their intended function. This verification is performed from the remote shutdown panel and locally, as appropriate. This will ensure that if the control room becomes inaccessible, the unit can be placed and maintained in MODE 3 from the remote shutdown panel and the local control stations. The [18]-month frequency was developed considering it was prudent that these surveillances only be performed during a facility outage. This was due to the plant conditions needed to perform the surveillance and the potential for unplanned transients if the surveillance is performed with

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

the reactor at power. Operating experience demonstrates that remote shutdown control channels seldom fail to pass the surveillance when performed on the [18]-month Frequency.

SR 3.3.17.3

CHANNEL CALIBRATION is a complete check of the instrument channel including the detector. The test verifies that the channel responds to measured parameters with the necessary range and accuracy.

Transmitter "as found" and "as left" values are recorded and used to verify drift assumptions. Field transmitters may be calibrated in place, removed and calibrated in a laboratory, or replaced with an equivalent laboratory-calibrated unit.

Resistance temperature detector (RTD) channels may be calibrated in place using cross-calibration techniques, or in a test bath after removal from piping. For cross-calibration, at least one RTD should be replaced with a newly calibrated RTD during each refueling cycle to ensure accurate RTD cross-calibration. This replacement RTD must be the same model as the remaining RTDs. Using a newly calibrated RTD as a reference assures RTD signal drift continues to remain random rather than systematic and is within the limits specified in the plant setpoint analysis. The replacement interval may be extended to alternate refueling if it is demonstrated that over the extended interval, the RTDs drift is random rather than systematic. This determination may use results of statistical analysis of operating data and calibration data from similar plants using the same model of RTD in the same environmental conditions.

The Surveillance Frequency is based upon the assumption of an 18-month calibration interval in the determination of the magnitude of equipment drift.

SR 3.3.17.4

SR 3.3.17.4 is the performance of a TRIP ACTUATING DEVICE OPERATIONAL TEST every [18] months. This test should verify the OPERABILITY of the Reactor Trip Breakers' (RTBs') open and closed indications on the remote shutdown panel, by actuating the RTBs. The Surveillance Frequency of 18 months was chosen because the RTBs cannot be exercised while the

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)	facility is at power. Operating experience has shown that these components usually pass the surveillance when performed on an [18]-month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.
REFERENCES	1. Title 10, Code of Federal Regulations, Part 50, Appendix A, General Design Criterion 19, "Control Room." 2. Title 10, Code of Federal Regulations, Part 50, Appendix R, "[Title]." 3. [Unit Name] FSAR, Section [ ], "[Title]."

Table B 3.3.17-1 (page 1 of 2)
Remote Shutdown System Instrumentation and Controls

FUNCTION/INSTRUMENT OR CONTROL PARAMETER	LOCATION	REQUIRED NUMBER OF DIVISIONS
NOTE: This table is for illustration purposes only. It does not attempt to encompass every function used at every plant, but does contain the types of functions commonly found.		
1. Reactivity Control		
a. Log Power Neutron Flux		[1]
b. Source Range Neutron Flux		[1]
c. Reactor Trip Circuit Breaker Position		[1/trip breaker]
d. Manual Reactor Trip		[4]
2. RCS Pressure Control		
a. Pressurizer Pressure or RCS Wide-Range Pressure		[1]
b. Pressurizer Power Operated Relief Valve (PORV) Control and Block Valve Control		[1, controls must be for PORV and block valves on same line]
3. Decay Heat Removal via SGs		
a. Reactor Coolant Hot Leg Temperature		[1/loop]
b. Reactor Coolant Cold Leg Temperature		[1/loop]
c. EFW Controls Condensate Storage Tank Level		[1]

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Table B 3.3.17-1 (page 2 of 2)
Remote Shutdown System Instrumentation and Controls

FUNCTION/INSTRUMENT OR CONTROL PARAMETER	LOCATION	REQUIRED NUMBER OF DIVISIONS
d. SG Pressure		[1/SG]
e. SG Level or EFW Flow		[1/SG]
4. RCS Inventory Control		
a. Pressurizer Level		[1]
b. Reactor Coolant Injection Pump Controls		[1]

APPENDIX A

Acronyms

The following acronyms are used, but not defined, in the Standard Technical Specifications:

AC	alternating current
CFR	Code of Federal Regulations
DC	direct current
FSAR	Final Safety Analysis Report
LCO	Limiting Condition for Operation
SR	Surveillance Requirement
GDC	General Design Criteria or General Design Criterion

The following acronyms are used, with definitions, in the Standard Technical Specifications:

ACOT	ANALOG CHANNEL OPERATIONAL TEST
ADS	Automatic Depressurization System
ADV	atmospheric dump valve
AFD	axial flux difference
AFW	auxiliary feedwater
AIRP	air intake, recirculation, and purification
ALARA	as low as reasonably achievable
ANS	American Nuclear Society
ANSI	American National Standards Institute
AOO	anticipated operational occurrence
AOT	allowed outage time
APD	axial power distribution
APLHGR	average planar linear heat generation rate
APRM	average power range monitor
APSR	AXIAL POWER SHAPING ROD
ARO	all rods out
ARC	auxiliary relay cabinets
ARS	Air Return System
ARTS	Anticipatory Reactor Trip System
ASGT	asymmetric steam generator transient
ASGTPTF	asymmetric steam generator transient protective trip function
ASI	AXIAL SHAPE INDEX
ASME	American Society of Mechanical Engineers

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ASTM	American Society for Testing Materials
ATWS	anticipated transient without scram
ATWS-RPT	anticipated transient without scram recirculation pump trip
AVV	atmospheric vent valve
BAST	boric acid storage tank
BAT	boric acid tank
BDPS	Boron Dilution Protection System
BIST	boron injection surge tank
BIT	boron injection tank
BOC	beginning of cycle
BOP	balance of plant
BPWS	banked position withdrawal sequence
BWST	borated water storage tank
BTP	Branch Technical Position
CAD	containment atmosphere dilution
CAOC	constant axial offset control
CAS	Chemical Addition System
CCAS	containment cooling actuation signal
CCGC	containment combustible gas control
CCW	component cooling water
CEA	control element assembly
CEAC	control element assembly calculator
CEDM	control element drive mechanism
CFT	core flood tank
CIAS	containment isolation actuation signal
COLR	CORE OPERATING LIMITS REPORT
COLSS	Core Operating Limits Supervisory System
CPC	core protection calculator
CPR	critical power ratio
CRA	CONTROL ROD assembly
CRD	CONTROL ROD drive
CRDA	CONTROL ROD drop accident
CRDM	CONTROL ROD drive mechanism
CREHVAC	Control Room Emergency Air Temperature Control System
CREFS	Control Room Emergency Filtration System
CREVS	Control Room Emergency Ventilation System
CRFAS	Control Room Fresh Air System
CS	core spray
CSAS	containment spray actuation signal

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CST	condensate storage tank
CVCS	Chemical and Volume Control System
DBA	Design Basis Accident
DBE	Design Basis Event
DF	decontamination factor
DG	diesel generator
DIV	drywell isolation valve
DNB	departure from nucleate boiling
DNBR	departure from nucleate boiling ratio
DOP	dioctyl phthalate
DPIV	drywell purge isolation valve
DRPI	digital rod position indicator
EAB	exclusion area boundary
ECCS	Emergency Core Cooling System
ECW	essential chilled water
ECP	estimated critical position
EDG	emergency diesel generator
EFAS	Emergency Feedwater Actuation System
EFIC	emergency feedwater initiation and control
EFCV	excess flow check valve
EFPDs	effective full power days
EFPYs	effective full power years
EFW	emergency feedwater
EHC	electro-hydraulic control
EOC	end of cycle
EOC-RPT	end of cycle recirculation pump trip
ESF	engineered safety feature
ESFAS	Engineered Safety Feature Actuation System
ESW	essential service water
EVS	Emergency Ventilation System
FBACS	Fuel Building Air Cleanup System
FCV	flow control valve
FHAVS	Fuel Handling Area Ventilation System
FSPVS	Fuel Storage Pool Ventilation System
FRC	fractional relief capacity
FR	Federal Register
FTC	fuel temperature coefficient
FWLB	feedwater line break

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HCS	Hydrogen Control System; Hydrazine Control System
HCU	hydraulic control unit
HIS	Hydrogen Ignition System
HELB	high energy line break
HEPA	high efficiency particulate air
HMS	Hydrogen Mixing System
HPCI	high pressure coolant injection
HPCS	high pressure core spray
HPI	high pressure injection
HPSI	high pressure safety injection
HPSP	high power setpoint
HVAC	heating, ventilation, and air conditioning
HZP	hot zero power
ICS	Integrated Control System
IEEE	Institute of Electrical and Electronic Engineers
IGSCC	intergranular stress corrosion cracking
IRM	intermediate range monitor
ISLH	inservice leak and hydrostatic
ITC	isothermal temperature coefficient
K-relay	control relay
LCS	Leakage Control System
LEFM	linear elastic fracture mechanics
LER	Licensee Event Report
LHGR	linear heat generation rate
LHR	linear heat rate
LLS	low-low set
LOCA	loss-of-coolant accident
LOCV	loss of condenser vacuum
LOMFW	loss of main feedwater
LOP	loss of power
LOPS	loss of power start
LOVS	loss of voltage start
LPCI	low pressure coolant injection
LPCS	low pressure core spray
LPD	local power density
LPI	low pressure injection
LPRM	local power range monitor
LPSI	low pressure safety injection
LPSP	low power setpoint

APPENDIX A (continued)

LPZ	low population zone
LSSS	limiting safety system settings
LTA	lead test assembly
LTOP	low temperature overpressure protection
MAPLHGR	maximum average planar linear heat generation rate
MAPFAC	MAPLHGR factor
$MAPFAC_f$	MAPLHGR factor, flow-dependent component
$MAPFAC_p$	MAPLHGR factor, power-dependent component
MCPR	minimum critical power ratio
MCR	main control room
MCREC	main control room environmental control
MFI	minimum flow interlock
MFIV	main feedwater isolation valve
MFLPD	maximum fraction of limiting power density
MFRV	main feedwater regulation valve
MFW	main feedwater
MG	motor generator
MOC	middle of cycle
MSIS	main steam isolation signal
MSIV	main steam isolation valve
MSLB	main steam line break
MSSV	main steam safety valve
MTC	moderator temperature coefficient
NDT	nil-ductility temperature
NDTT	nil-ductility transition temperature
NI	nuclear instrument
NIS	Nuclear Instrumentation System
NMS	Neutron Monitoring System
NPSH	net positive suction head
NSSS	Nuclear Steam Supply System
ODCM	Offsite Dose Calculation Manual
OPDRV	operation with a potential for draining the reactor vessel
OTSG	once-through steam generator
PAM	post-accident monitoring
PCCGC	primary containment combustible gas control
PCI	primary containment isolation

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PCIV	primary containment isolation valve
PCHRS	Primary Containment Hydrogen Recombiner System
PCP	Process Control Program
PCPV	primary containment purge valve
PCT	peak cladding temperature
PDIL	power dependent insertion limit
PDL	power distribution limit
PF	position factor
PIP	position indication probe
PIV	pressure isolation valve
PORV	power operated relief valve
PPS	Plant Protective System
PRA	probabilistic risk assessment
PREACS	Pump Room Exhaust Air Cleanup System; Penetration Room Exhaust Air Cleanup System
PSW	plant service water
P/T	pressure and temperature
PTE	PHYSICS TEST exception
PTLR	PRESSURE AND TEMPERATURE LIMITS REPORT
QA	quality assurance
QPT	QUADRANT POWER TILT
QPTR	quadrant power tilt ratio
QS	quench spray
RACS	Rod Action Control System
RAOC	relaxed axial offset control
RAS	recirculation actuation signal
RB	reactor building
RBM	rod block monitor
RCCA	rod cluster control assembly
RCIC	reactor core isolation cooling
RCIS	Rod Control and Information System
RCP	reactor coolant pump
RCPB	reactor coolant pressure boundary
RCS	Reactor Coolant System
REA	rod ejection accident
RHR	residual heat removal
RHRSW	residual heat removal service water
RMCS	Reactor Manual Control System
RPB	reactor pressure boundaries
RPC	rod pattern controller
RPCB	reactor power cutback

(continued)

RPIS	Rod Position Information System
RPS	Reactor Protection System
RPV	reactor pressure vessel
RS	recirculation spray
RT	reference temperature
RT_{NDT}	nil-ductility reference temperature
RTCB	reactor trip circuit breaker
RTD	resistance temperature detector
RTM	reactor trip module
RTP	RATED THERMAL POWER
RTS	Reactor Trip System
RWCU	reactor water cleanup
RWE	rod withdrawal error
RWL	rod withdrawal limiter
RWM	rod worth minimizer
RWP	Radiation Work Permit
RWST	refueling water storage tank
RWT	refueling water tank
SAFDL	specified acceptable fuel design limits
SBCS	Steam Bypass Control System
SBO	station blackout
SBVS	Shield Building Ventilation System
SCAT	spray chemical addition tank
SCI	secondary containment isolation
SCR	silicon controlled rectifier
SDV	scram discharge volume
SDM	SHUTDOWN MARGIN
SER	Safety Evaluation Report
SFRCS	Steam and Feedwater Rupture Control System
SG	steam generator
SGTR	steam generator tube rupture
SGTS	Standby Gas Treatment System
SI	safety injection
SIAS	safety injection actuation signal
SIS	safety injection signal
SIT	safety injection tank
SJAE	steam jet air ejector
SL	Safety Limit
SLB	steam line break
SLC	standby liquid control
SLCS	Standby Liquid Control System
SPMS	Suppression Pool Makeup System
SRM	source range monitor

(continued)

APPENDIX A (continued)

S/RV	safety/relief valve
S/RVDL	safety/relief valve discharge line
SSPS	Solid State Protection System
SSW	standby service water
SWS	Service Water System
STE	special test exception
STS	Standard Technical Specifications
TADOT	trip actuating device operational test
TCV	turbine control valve
TIP	traversing incore probe
TLD	thermoluminescent dosimeter
TM/LP	thermal margin/low pressure
TS	Technical Specifications
TSV	turbine stop valve
UHS	Ultimate Heat Sink
VCT	volume control tank
VFTP	Ventilation Filter Testing Program
VHPT	variable high power trip
v/o	volume percent
VS	vendor specific
ZPMB	zero power mode bypass

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This draft report documents the results of the NRC staff review of new Standard Technical Specifications (STS) proposed by the Babcock and Wilcox Owners Group. The new STS were developed based on the criteria in the interim Commission Policy Statement on Technical Specification Improvements for Nuclear Power Reactors, dated February 6, 1987. The new STS will be used as bases for individual nuclear power plant owners to develop improved plant-specific technical specifications. The NRC staff is issuing this draft new STS for a 30 working-day comment period. Following the comment period, the NRC staff will analyze comments received, finalize the new STS, and issue them for plant-specific implementation. This report contains three volumes. Volume 1 contains the Specifications for all sections of the new STS. Volume 2 contains the Bases for Sections 2.0 - 3.3 of the new STS and Volume 3 contains the Bases for Sections 3.4 - 3.9 of the new STS.

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