NUREG–1433
Vol. 2

Standard Technical Specifications
General Electric Plants, BWR/4

Bases (Sections 2.0–3.3)

Draft Report for Comment

Issued by the
U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

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PREFACE

This DRAFT NUREG presents the results of the Nuclear Regulatory Commission (NRC) staff review of the BWR Owners Group (BWROG) proposed new Standard Technical Specifications (STS) for the BWR/4 design. 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 BWRs designed by General Electric. 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-1433, "Standard Technical Specifications, General Electric Plants, BWR/4." 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 by specifying a MINIMUM CRITICAL POWER RATIO (MCPR) such that at least 99.9% of the fuel rods in the core would not be expected to experience onset of boiling transition. The restrictions of this SL prevent overheating of the fuel and cladding, as well as possible cladding perforation that would result in the release of fission products to the reactor coolant. Overheating of the fuel and overstress of the cladding is prevented by maintaining the steady-state peak linear heat generation rate (LHGR) below the level at which 1% plastic strain of the cladding would occur. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime, where the heat-transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature. Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of transition boiling 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. The proper functioning of the Reactor Protection System (RPS) 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 fuel design

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

criterion that an MCPR is to be established such that at least 99.9% of the fuel rods in the core would not be expected to experience the onset of transition boiling.

The RPS setpoints (Ref. 2), in combination with the LCOs, are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System temperature, pressure, and THERMAL POWER level that would result in reaching the MCPR.

Automatic enforcement of these reactor core SLs are provided by the trip setpoints for the following functions:

a. Average power range monitor trip;

b. Reactor vessel water level—low level 3 trip;

c. Main steam line isolation valve—closure trip; and

d. Scram discharge volume water level—high trip.

2.1.1.1a Fuel Cladding Integrity (General Electric Corporation (GE) Fuel)

GE critical power correlations are applicable for all critical power calculations at pressures ≥ 785 psig or core flows ≥ 10% of rated flow. For operation at low pressures and low flows another basis is used, as follows:

> Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be > 4.5 psi. Analyses (Ref. 3) show that with a bundle flow of 28 x 10^3 lb/hour bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.5 psi driving head will be greater than 28 x 10^3 lb/hour. Full-scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MW. With the design peaking factors, this corresponds to a THERMAL POWER of more than 50% RATED THERMAL POWER (RTP). Thus, a THERMAL POWER limit of 25% RTP for reactor pressure < 785 psig is conservative.

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

2.1.1.1b Fuel Cladding Integrity (Advanced Nuclear Fuel Corporation (ANF) Fuel)

The use of the XN-3 correlation is valid for critical power calculations at pressures > 580 psig and bundle mass fluxes > 0.25×10^6 lb/hour-ft^2 (Ref. 4). For operation at low pressures or low flows, the fuel cladding integrity SL is established by a limiting condition on core THERMAL POWER, with the following basis:

> Provided that the water level in the vessel downcomer is maintained above the top of the active fuel, natural circulation is sufficient to ensure a minimum bundle flow for all fuel assemblies that have a relatively high power and potentially can approach a critical heat flux condition. For the ANF 9x9 fuel design, the minimum bundle flow is > 30×10^3 lb/hour. For the ANF and GE 8x8 fuel, the minimum bundle flow is > 28×10^3 lb/hour. For all designs, the coolant minimum bundle flow and maximum flow area is such that the mass flux is always > 0.25×10^6 lb/hour-ft^2. Full-scale critical power tests taken at pressures down to 14.7 psia indicate that the fuel assembly critical power at 0.25×10^6 lb/hour-ft^2 is ≥ 3.35 MW. At 25% RTP, a bundle power of 3.35 MW corresponds to a bundle radial peaking factor of > 3.0, which is significantly higher than the expected peaking factor. Thus, a THERMAL POWER limit of 25% RTP for reactor pressures < 785 psig is conservative.

2.1.1.2a Minimum Critical Power Ratio (GE Fuel)

The fuel-cladding integrity SL is set such that no significant fuel damage is calculated to occur if the limit is not violated. Since the parameters that result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions that result in the onset of transition boiling have been used to make the beginning of the region in which fuel damage could occur. Although it is recognized that the onset of transition boiling would not result in damage to BWR fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. However, the uncertainties in monitoring the core operating state and in the procedures used to calculate the critical power result

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

in an uncertainty in the value of the critical power. Therefore, the fuel-cladding integrity SL is defined as the critical power ratio in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition, considering the power distribution within the core and all uncertainties.

The MCPR SL is determined using a statistical model that combines all the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the approved General Electric Critical Power correlations. Details of the fuel-cladding integrity SL calculation are given in Reference 3. Reference 3 also includes a tabulation of the uncertainties used in the determination of the MCPR SL and of the nominal values of the parameters used in the MCPR SL statistical analysis.

2.1.1.2b Minimum Critical Power Ratio (ANF Fuel)

The MCPR SL ensures sufficient conservatism in the operating MCPR limit that, in the event of an AOO from the LCO, at least 99.9% of the fuel rods in the core would be expected to avoid boiling transition. The margin between calculated boiling transition (MCPR = 1.00) and the MCPR SL is based on a detailed statistical procedure which considers the uncertainties in monitoring the core operating state. One specific uncertainty included in the SL is the uncertainty inherent in the XN-3 critical power correlation. Reference 4 describes the methodology used in determining the MCPR SL.

The XN-3 critical-power correlation is based on a significant body of practical test data, providing a high degree of assurance that the critical power as evaluated by the correlation is within a small percentage of the actual critical power being estimated. As long as the core pressure and flow are within the range of validity of the XN-3 correlation, the assumed reactor conditions used in defining the SL introduce conservatism into the limit because bounding high radial power factors and bounding flat local peaking distributions are used to estimate the number of rods in boiling transition. Still further conservatism is induced by the tendency of the XN-3 correlation to overpredict the number of rods in boiling transition. These

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

conservatisms and the inherent accuracy of the XN-3 correlation provide a reasonable degree of assurance that there would be no transition boiling in the core during sustained operation at the MCPR SL. If boiling transition were to occur, there is reason to believe that the integrity of the fuel would not be compromised. Significant test data accumulated by the NRC and private organizations indicate that the use of a boiling transition limitation to protect against cladding failure is a very conservative approach. Much of the data indicate that BWR fuel can survive for an extended period of time in an environment of boiling transition.

2.1.1.3 Reactor Vessel Water Level

During MODES 1 and 2 the assumption of water level above the top of the active fuel is inherent in the critical power correlations. Also, with fuel in the reactor vessel during periods when the reactor is shut down, consideration must be given to water level requirements due to the effect of decay heat. If the water level should drop below the top of the active irradiated fuel during this period, the ability to remove decay heat is reduced. This reduction in cooling capability could lead to elevated cladding temperatures and clad perforation in the event that the water level becomes < $\frac{2}{3}$ of the core height. The reactor vessel water level SL has been established at the top of the active irradiated fuel to provide a point that can be monitored and to also provide adequate margin for effective action.

The reactor core SLs represent a design requirement for establishing the Reactor Protection System setpoints identified previously. LCO 3.2.1, "Average Planar Linear Heat Generation Rate (APLHGR)"; LCO 3.2.2, "Minimum Critical Power Ratio (MCPR)"; and LCO 3.2.3, "Linear Heat Generation Rate (LHGR)," or the assumed initial conditions of the safety analyses (as indicated in the FSAR, Ref. 2), provide more restrictive limits to ensure that the reactor core SLs are not exceeded.

SAFETY LIMITS

The reactor core SLs are established to protect the integrity of the fuel clad barrier to the release of radioactive materials to the environs. SL 2.1.1.1 and SL 2.1.1.2 ensure that the core operates within the fuel

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BASES (continued)

SAFETY LIMITS (continued)

design criteria. SL 2.1.1.3 ensures that the reactor vessel water level is > top of the active irradiated fuel, thus maintaining a coolable geometry.

APPLICABILITY

SL 2.1.1.1 is applicable in MODES 1 and 2 with reactor steam dome pressure < 785 psig or core flow < 10% of rated core flow. As discussed in the Applicable Safety Analyses section, the limit of ≤ 25% RTP is sufficiently conservative to preclude boiling transition.

SL 2.1.1.2 is applicable in MODES 1 and 2 with reactor steam dome pressure ≥ 785 psig and core flow ≥ 10% of rated core flow. The MCPR SL ensures that the fuel design criteria are satisfied.

SL 2.1.1.3 is applicable in all modes.

SAFETY LIMIT VIOLATIONS

2.2.1

Exceeding any SL may cause immediate fuel damage or pressure vessel failure and create a potential for radioactive releases in excess of 10 CFR 100, "Reactor Site Criteria." Therefore, it is required to insert all insertable control rods and restore compliance with the SLs within 2 hours. The 2-hour Completion Time ensures that the operators take prompt remedial action and also ensures that the probability of an accident occurring during this period is minimal.

2.2.2

If any SL is violated, the NRC Operations Center must be notified within 1 hour. This is in accordance with 10 CFR 50.72 (Ref. 5).

2.2.3

If any SL is violated, the appropriate senior management of the nuclear plant and the utility shall be notified within 24 hours. The 24-hour period provides time for plant operators and staff to take the appropriate immediate action

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BASES (continued)

SAFETY LIMIT VIOLATIONS (continued)

and assess the condition of the plant before reporting to the senior management.

2.2.4

If any 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. 6).

2.2.5

If any 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.

REFERENCES

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

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

3. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel," (latest approved revision).

4. XN-NF524(A), "Exxon Nuclear Critical Power Methodology for Boiling Water Reactors," Revision 1, November 1983.

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

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

B 2.0 SAFETY LIMITS

B 2.1.2 Reactor Steam Dome Pressure Safety Limits (SLs)

BASES

BACKGROUND

The SL on reactor steam dome pressure protects the integrity of the reactor pressure vessel (RPV) and the recirculation piping against overpressurization. In the event of fuel cladding failure, fission products are released into the reactor coolant. The Reactor Coolant System (RCS) then serves as the primary barrier in preventing the release of fission products into the atmosphere. Establishing an upper limit on reactor steam dome pressure assures continued RPV and recirculation piping integrity. Per 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 and anticipated operational occurrences (AOOs). Also, per GDC 28, "Reactivity Limits" (Ref. 1), reactivity accidents, including rod ejection, do not result in damage to the RCPB greater than limited local yielding.

The design pressure of the RCS is 1250 psi. During normal operation and AOOs, RCS pressure is limited from exceeding the design pressure by more than 10%, in accordance with Section III of the American Society of Mechanical Engineers (ASME) Code (Ref. 2). To ensure system integrity, all RCS components are hydrostatically tested at 125% of design, per ASME Code requirements, prior to initial operation when there is no fuel in the core. Any further hydrostatic testing with fuel in the core is done under LCO 3.10.1, "Inservice Leak and Hydrostatic (ISLH) Test."

Overpressurization of the RCS could result in a breach of the RCPB. If this occurred 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."

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BASES (continued)

APPLICABLE SAFETY ANALYSES

The RCS safety valves and the reactor steam dome pressure—high trip have settings established to ensure that the RCS pressure SL will not be exceeded.

The reactor steam dome pressure SL has been selected such that it is at a pressure below which it can be shown that the integrity of the system is not endangered. The RPV is designed to Section III of the ASME, Boiler and Pressure Vessel Code, 1971 edition, including Addenda through the winter of 1972, which permits a maximum pressure transient of 110%, 1375 psig, of design pressure 1250 psig. The SL of 1325 psig, as measured by the reactor steam dome pressure indicator, is equivalent to 1375 psig at the lowest elevation of the RCS. The RCS is designed to the USAS Nuclear Power Piping Code, Section B31.1, 1969 Edition, including Addenda through July 1, 1970, for the reactor recirculation piping, which permits a maximum pressure transient of 110% of design pressures of 1250 psig for suction piping and 1500 psig for discharge piping. The reactor steam dome pressure SL is selected to be the lowest transient overpressure allowed by the applicable codes.

The Reactor Protection System (RPS) trip setpoints (Ref. 3), together with the settings of the RCS safety valves (Ref. 4), provide pressure protection for normal operation and AOOs. In particular, the reactor steam dome pressure—high setpoint is set to provide protection against overpressurization (Ref. 5). The safety analyses for both the reactor steam dome pressure—high trip and the RCS safety valves are performed using conservative assumptions relative to pressure control devices.

SAFETY LIMITS

The maximum transient pressure allowable in the RCS pressure vessel under the ASME Code, Section III, is 110% of design pressure. The maximum transient pressure allowable in the RCS piping, valves, and fittings under [USAS, Section B31.1, Ref. 6] is 110% of design pressure of 1250 psig for suction piping and 1500 psig for discharge piping. The most limiting of these two allowances is the 110% of design pressure; therefore, the SL on maximum allowable RCS pressure is established at 1375 psig.

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BASES (continued)

APPLICABILITY

SL 2.1.2 applies in MODES 1 through 4 because it is conceivable to approach or exceed this SL in these MODES due to overpressurization events. The SL is not applicable in MODE 5 because the reactor vessel head closure bolts are not fully tightened, making it impossible to pressurize the RCS.

SAFETY LIMIT VIOLATIONS

2.2.1

Exceeding any SL may cause immediate fuel damage or pressure vessel failure and create a potential for radioactive releases in excess of 10 CFR 100, "Reactor Site Criteria." Therefore, it is required to insert all insertable control rods and restore compliance with the SLs within 2 hours. The 2-hour Completion Times ensures that the operators take prompt remedial action.

2.2.2

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

2.2.3

If any SL is violated, the appropriate senior management of the nuclear plant and the utility shall be notified within 24 hours. The 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 the senior management.

2.2.4

If any 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).

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BASES (continued)

SAFETY LIMIT VIOLATIONS (continued)

2.2.5

If any 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.

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."

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. American Society of Mechanical Engineers, USAS B31.1, Standard Code for Pressure Piping, 1969, and Addenda through July 1, 1970.

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 3.01, LCO 3.02, LCO 3.03, LCO 3.04, and LCO 3.05	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, "Programs and Manuals," as well as LCOs contained in Sections 3.1 through 3.10.
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

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BASES (continued)

LCO 3.0.2 (continued)

in which the specification no longer applies. (Whether 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 stated 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 has 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 ACTIONS 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 an LCO Condition for which the Completion Times of the Required Action(s) have not expired. For example, if while in MODE 1, one of the two residual heat removal suppression pool spray subsystems is declared inoperable. The corresponding ACTIONS Condition of the LCO for one inoperable subsystem is entered and 7 days are allowed to restore the subsystem to OPERABLE status. Then, the second subsystem is declared inoperable at time 24 hours into the Completion Time. Since no ACTIONS Condition is provided for both subsystems being inoperable, LCO 3.0.3 must be entered. If one of the subsystems

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BASES (continued)

LCO 3.0.3 (continued)

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 Condition of one subsystem being inoperable. In this example, that would mean operation for another 5 days, 18 hours. If the subsystem is restored to OPERABLE status after going to MODE 2 or 3, 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 4 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 4, or other applicable MODE, is not reduced. For example, if MODE 2 is reached in 2 hours, the time allowed to reach MODE 3 is the next 11 hours, because the total time to reach MODE 3 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, and 3, 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 4 and 5 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, or 3) 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 exist:

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 3.05

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 a special operations LCO represents a Condition not necessarily in compliance with the normal requirements of the TS. Compliance with special operations LCOs is optional.

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

Some of the LCOs for special operations require that one or more of the LCOs for normal operation be met, i.e., meeting the special operations 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 special operations LCO when it is in effect. This means that, upon failure to meet a specified normal LCO, the associated ACTIONS of the special operations 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 special operations LCO.

Unless the SRs of the specified normal LCOs are suspended or changed by the special operations LCO, those SRs that are

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BASES (continued)

LCO 3.0.5 (continued)

necessary to meet the specified normal LCOs must be met prior to performing the special operation. During the conduct of the special operation, those Surveillances need not be performed unless specified by the ACTIONS or SRs of the special operations LCO.

ACTIONS for special operations LCOs provide appropriate remedial measures upon failure to meet the special operations LCO. Upon failure to meet these ACTIONS, suspend the performance of the special operations 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 special operations 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, SR 3.0.2, SR 3.0.3, and SR 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 3.10.

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 operation are only applicable when the special operation is used as an allowable exception to the requirements of a specification.

Surveillances, including Surveillances invoked by Required Actions, do not have to be performed on inoperable equipment

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BASES (continued)

SR 3.0.1 (continued)

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 applicable 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 by performing the Surveillance at its specified Frequency. This recognizes that the most probable result of any particular Surveillance being performed is the verification of conformance with the

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BASES (continued)

SR 3.0.2 (continued)

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 TS 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 (i.e., flexibility for scheduling the performance of Surveillances, etc.).

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BASES (continued)

SR 3.0.2 (continued)

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 Surveillance. The delay period is considered appropriate for balancing the risk associated with delaying completion

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BASES (continued)

SR 3.0.3 (continued)

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 provided

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BASES (continued)

SR 3.0.4 (continued)

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

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BASES (continued)

SR 3.0.4 (continued)

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 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,

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BASES (continued)

SR 3.0.4 (continued)	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 shut down 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 assure 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 control rods assuming the single rod 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.

During power operation, SDM control is ensured by operating with the control rods within the limits of LCO 3.1.6. When in the shutdown and refueling MODES, the SDM requirements are met by rods being bottomed or during Special Operations by strict administrative control and equipment interlocks.

APPLICABLE SAFETY ANALYSES

The minimum required SDM is assumed as an initial condition in safety analyses. The safety analyses establish a SDM that ensures that specified acceptable fuel design limits are not exceeded for normal operation and AOOs with the assumption of the highest worth rod stuck out on scram. Specifically, the primary safety analysis that relies on the SDM limits in MODES 1 and 2 is the control rod drop accident (CRDA).

The CRDA analysis (Refs. 2 and 3) assumes the core is subcritical with the highest worth control rod withdrawn. Typically, the first control rod withdrawn has a very high

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

reactivity worth and, should the core be critical during the withdrawal of the first control rod, the consequences of a CRDA could exceed the fuel damage limits for a CRDA (see Bases for LCO 3.1.6, "Rod Pattern Control"). Also, SDM is assumed as an initial condition for the control rod removal error during refueling (Ref. 4) and fuel assembly insertion error during refueling (Ref. 5) accidents. The analysis of these reactivity insertion events assumes the refueling interlocks are OPERABLE when the reactor is in the refueling mode of operation. These interlocks prevent the withdrawal of more than one control rod from the core during refueling. (Special consideration and requirements for multiple control rod withdrawal during refueling are covered in Special Operations LCO 3.10.6, "Multiple Control Rod Withdrawal—Refueling.") The analysis assumes this condition is acceptable since the core will be shut down with the highest worth control rod withdrawn, if adequate SDM has been demonstrated.

Prevention or mitigation of reactivity insertion events is necessary to limit energy deposition in the fuel to prevent significant fuel damage which could result in undue release of radioactivity (see Bases for LCO 3.1.7). Adequate SDM ensures inadvertent criticalities and potential CRDAs involving high worth control rods (namely the first control rod withdrawn) will not cause significant fuel damage.

SDM satisfies Criterion 2 of the NRC Interim Policy Statement.

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.

The specified SDM limit accounts for the uncertainty in the demonstration of SDM by testing. Separate SDM limits are provided for testing where the highest worth control rod is determined analytically or by measurement. This is due to the reduced uncertainty in the SDM test when the highest worth control rod is determined by measurement (Ref. 6). When SDM is demonstrated by calculations not associated with a test, additional margin must be added to the specified SDM

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BASES (continued)

LCO (continued)	limit to account for uncertainties in the calculation. To ensure adequate SDM during the design process, a design margin is included to account for uncertainties in the design calculations (Ref. 4). SDM is a core physics design condition that is evaluated during SR 3.1.1.1, and appropriate actions are taken as necessary when the SDM is not within the required limit.
APPLICABILITY	In MODES 1 and 2, SDM must be provided because subcriticality with the highest worth control rod withdrawn is assumed in the CRDA analysis (Ref. 2). Also, the capability to reach MODE 4 conditions from any initial state is required by GDC 26. In MODES 3 and 4, SDM is required to ensure the reactor will be held subcritical with margin for a single withdrawn control rod. SDM is required in MODE 5 to prevent an open vessel, inadvertent criticality during the withdrawal of a single control rod from a core cell containing one or more fuel assemblies. A Note is added to provide clarification that for this LCO, Condition A, Condition C, Condition D, or Condition E is treated as an independent entity with an independent Completion Time for each condition.
ACTIONS	A.1 With SDM not within the limits of the LCO in MODE 1 or 2, SDM must be restored within 6 hours. Failure to meet the specified SDM may be caused by a control rod that cannot be inserted. The 6-hour Completion Time is acceptable considering that the reactor can still be shut down, assuming no failures of additional control rods to insert, and the low probability of an event occurring during this interval. (continued)

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BASES (continued)

ACTIONS (continued)

B.1

If the SDM cannot be restored, the reactor must be in MODE 3 in 12 hours to prevent the potential for further reductions in available SDM (e.g., additional stuck control rods). The 12-hour Completion Time is reasonable, based on operating experience, to reach the required MODE from full power in an orderly manner and without challenging plant systems.

C.1

With SDM not within limits in MODE 3, the operator must fully insert all insertable control rods in 1 hour. This action results in the least reactive condition for the core. The 1-hour Completion Time provides sufficient time to take corrective action and is acceptable considering that the reactor can still be shut down, assuming there are no failures of additional control rods to insert.

D.1, D.2, D.3, and D.4

With SDM not within limits in MODE 4, the operator must insert all insertable control rods in 1 hour. This action results in the least reactive condition for the core. The 1-hour Completion Time provides sufficient time to take corrective action and is acceptable considering that the reactor can still be shut down assuming there are no failures of additional control rods to insert. Actions must also be initiated within 1 hour to provide means for control of potential radioactive releases. This includes ensuring secondary containment is OPERABLE (LCO 3.6.4.1), at least one Standby Gas Treatment System (SGTS) (LCO 3.6.4.5) subsystem is OPERABLE, and at least one secondary containment isolation valve (LCO 3.6.4.3) and associated instrumentation are OPERABLE in each associated penetration not isolated. This may be performed as an administrative check, by examining logs or other information, to determine if the components are out of service for maintenance or other reasons. It does not mean to perform the surveillance requirements needed to demonstrate the OPERABILITY of the components. If, however, any required component is inoperable, then it must be restored to OPERABLE status. In this case, surveillance requirements may need to be performed to restore the component to OPERABLE status.

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BASES (continued)

ACTIONS (continued)

Actions must continue until all required components are OPERABLE. The 1-hour Completion Time is sufficient time to take the Required Actions.

E.1, E.2, E.3, E.4, and E.5

With SDM not within limits in MODE 5, the operator must immediately suspend CORE ALTERATIONS that could reduce SDM. The suspensions are on inserting fuel in the core or the withdrawal of control rods. Suspension of these activities shall not preclude completion of movement of a component to a safe condition. Inserting control rods or removing fuel from the core will reduce the total reactivity and are therefore excluded from the suspension actions.

Action must also be immediately initiated to fully insert all insertable control rods in core cells containing one or more fuel assemblies. Actions must continue until all insertable control rods in core cells containing one or more fuel assemblies have been fully inserted. Control rods in core cells containing no fuel assemblies do not affect the reactivity of the core and therefore do not have to be inserted.

Actions must also be initiated within 1 hour to provide means for control of potential radioactive releases. This includes ensuring secondary containment is OPERABLE (LCO 3.6.4.1), at least one SGTS (LCO 3.6.4.5) subsystem is OPERABLE, and at least one secondary containment isolation valve (LCO 3.6.4.3) and associated instrumentation are OPERABLE in each associated penetration not isolated. This may be performed as an administrative check, by examining logs or other information, to determine if the components are out of service for maintenance or other reasons. It does not mean to perform the surveillance requirements needed to demonstrate the OPERABILITY of the components. If, however, any required component is inoperable, then it must be restored to OPERABLE status. In this case, surveillance requirements may need to be performed to restore the component to OPERABLE status. Actions must continue until all required components are OPERABLE.

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BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1

Adequate SDM is demonstrated to ensure that the reactor can be made subcritical from any initial operating condition. Adequate SDM must be demonstrated by testing before or during the first startup after fuel movement, control rod replacement, or shuffling within the reactor pressure vessel. Since core reactivity will vary during the cycle as a function of fuel depletion and poison burnup, the beginning of cycle (BOC) test must also account for changes in core reactivity during the cycle. Therefore, to obtain the SDM, the initial measured value must be increased by an adder, R, which is the difference between the calculated value of maximum core reactivity during the operating cycle and the calculated BOC core reactivity. If the value of R is negative (that is, BOC is the most reactive point in the cycle), no correction to the BOC measured value is required (Ref. 7).

The SDM may be demonstrated during an in-sequence control rod withdrawal, in which the highest worth control rod is analytically determined, or during local criticals, where the highest worth control rod is determined by testing. Local critical tests require the withdrawal of out-of-sequence control rods. This testing would therefore require bypassing of the Rod Pattern Control System to allow the out-of-sequence withdrawal and therefore additional requirements must be met (see LCO 3.10.7, "Control Rod Testing—Operating").

Up to 4 hours after reaching criticality is allowed to provide a reasonable time to perform the required calculations and have appropriate verification.

During MODE 5, adequate SDM is required to ensure that the reactor does not reach criticality during control rod withdrawals. An evaluation of each fuel movement during fuel loading (including shuffling fuel within the core) shall be performed to ensure adequate SDM is maintained during refueling. This ensures that the intermediate loading patterns are bounded by the safety analyses for the final core loading pattern. For example, bounding analyses which demonstrate adequate SDM for the most reactive configurations during the refueling may be performed to

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

demonstrate acceptability of the enti› fuel movement sequence. For these SDM demonstrations which rely solely on calculation, additional margin must be added to the calculated SDM value to account for uncertainties in the calculation. Spiral offload/reload sequences inherently satisfy the surveillance requirement provided the fuel assemblies are reloaded in the same configuration analyzed for the new cycle. Removing fuel from the core will always result in an increase in SDM.

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 [15], "[Title]."
3. NEDE-24011-P-A-9-US, "General Electric Standard Application for Reload Fuel," Supplement for United States, Section S.2.2.3 1, September 1988.
4. [Unit Name] FSAR, Section [15], "[Title]."
5. [Unit Name] FSAR, Section [15], "[Title]."
6. [Unit Name] FSAR, Amendment [24], December 1972, Question [3.6.7], "[Title]."
7. NEDE-24011-P-A-9, "General Electric Standard Application for Reload Fuel," Supplement for United States, Section 3.2.4.1, September 198.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.2 Reactivity Anomalies

BASES

BACKGROUND

Per GDCs 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 anomaly is used as a measure of the predicted versus measured core reactivity during power operation. The continual confirmation of core reactivity is necessary to ensure that safety analyses of design basis transients and accidents remain valid. A large reactivity anomaly 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 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.

In order to achieve the required fuel cycle energy output, the uranium enrichment in the new fuel loading and the fuel loaded in the previous cycle provides excess positive reactivity beyond that required to sustain steady state operation at the beginning of cycle (BOC). When the reactor is critical at RATED THERMAL POWER (RTP) and operating moderator temperature, the excess positive reactivity is

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BASES (continued)

BACKGROUND (continued)	compensated by burnable absorbers (if any), control rods, and whatever neutron poisons (mainly xenon and samarium) are present in the fuel.
APPLICABLE 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 drop accidents, are very sensitive to accurate prediction of core reactivity. These accident analysis evaluations rely on computer codes that have been qualified against available test data, operating plant data, and analytical benchmarks. Monitoring reactivity anomaly provides additional assurance that the nuclear methods provide an accurate representation of the core reactivity. Design calculations and safety analyses are performed for each fuel cycle for the purpose of predetermining the reactivity behavior and the 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 rod density for identical core conditions at BOC do not agree, then the assumptions used in the reload cycle design analysis or the calculation models used to predict rod density 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 value. Thereafter, any significant deviations in the measured rod density from the predicted rod density that develop 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. Reactivity anomalies provide an additional assurance that SDM is maintained within the limits. Thus, reactivity anomalies satisfy Criterion 2 of the NRC Interim Policy Statement.

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BASES (continued)

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.

The reactivity anomaly limit is established to ensure plant operation is maintained within the assumptions of the safety analyses. Large differences between monitored 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 method are larger than expected. A limit on the difference between the monitored core k-effective and the predicted core k-effective 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.

APPLICABILITY

In MODE 1, most of the control rods are withdrawn and steady-state operation is typically achieved. Under these conditions, the comparison between predicted and monitored core reactivity provides an effective measure of the reactivity anomaly. In MODE 2, control rods are typically being withdrawn during a startup. In MODES 3 and 4, 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 5, fuel loading results in a continually changing core reactivity. SDM requirements (LCO 3.1.1) 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 that could have altered core reactivity (e.g., fuel movement, control rod replacement, shuffling). The SDM test required by LCO 3.1.1 provides a direct comparison of the predicted and monitored core reactivity at cold conditions, and, therefore, reactivity anomaly is not required during these conditions.

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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, smaller deviations in core reactivity (greater than 0.5% Δk/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 recirculation pump trip 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 void coefficient of the Reactor Coolant System (RCS) are considered.

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

a. Reactivity worth calculations of recirculation flow, the control rods, xenon, and samarium are reviewed;

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

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

Reactivity anomalies are generally investigated when they are small, so that the evaluations are in progress before the 1% Δk/k reactivity limit for a deviation is reached, and corrective measures may be defined. The required

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BASES (continued)

ACTIONS (continued)

Completion Time of 72 hours is based on operating experience and takes into account the low probability of a Design Basis Accident occurring during this interval. 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 rod density comparison, then a recalculation 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 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% Δk/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. The allowed Completion Time is reasonable, based on operating experience related to the time required, to reach the required MODE from RTP in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1

Verifying the reactivity difference between the monitored and predicted core k-effective is within the limits of the LCO provides added assurance that plant operation is

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

maintained within the assumptions of the design basis transient and accident analyses. The core monitoring system calculates the core k-effective for the reactor conditions obtained from plant instrumentation. A comparison of the monitored core k-effective to the predicted core k-effective at the same cycle exposure is used to calculate the reactivity difference. The comparison is required when the core reactivity has potentially changed by a significant amount. This may occur following a refueling in which new fuel assemblies are loaded, fuel assemblies are shuffled within the core, or control rods are replaced or shuffled. Also, core reactivity changes during the cycle. The 24-hour interval after reaching equilibrium conditions following a startup was established based on the need for equilibrium xenon concentrations in the core such that an accurate comparison between the monitored and predicted core k-effective values can be made. The 31 effective full power days Frequency was developed considering the relatively slow change in core reactivity with exposure and operating experience related to variations in core reactivity.

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, Chapter [15], Section [ ], "Accident Analysis."

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.3 Control Rod OPERABILITY

BASES

BACKGROUND	Control rods are components of the control rod drive (CRD) system, which is the primary reactivity control system for the reactor. In conjunction with the Reactor Protection System, the CRD System provides the means for the reliable control of reactivity changes to ensure under conditions of normal operation, including anticipated operational occurrences, specified acceptable fuel design limits are not exceeded. In addition, the control rods provide the capability to hold the reactor core subcritical under all conditions and to limit the potential amount and rate of reactivity increase caused by a malfunction in the CRD System. The CRD System is designed to satisfy the requirements of GDC 26, 27, 28, and 29. The CRD System consists of [137] locking-piston control rod drive mechanisms (CRDMs) and a hydraulic control unit for each drive mechanism. The locking-piston type CRDM is a double-acting hydraulic piston which uses condensate water as the operating fluid. Accumulators provide additional energy for scram. An index tube and piston, coupled to the control rod, are locked at fixed increments by a collet mechanism. The collet fingers engage notches in the index tube to prevent unintentional withdrawal of the control rod, but without restricting insertion. This specification, along with LCO 3.1.4 and LCO 3.1.5, assures that the performance of the control rods in the event of a Design Basis Accident (DBA) or transient, meets the assumptions used in the safety analyses of References 1, 2, and 3.
APPLICABLE SAFETY ANALYSES	The analytical methods and assumptions used in the evaluations involving control rods are presented in References 1, 2, and 3. The control rods provide the primary means for rapid reactivity control (reactor scram), for maintaining the reactor subcritical and for limiting the potential effects of reactivity insertion events caused by malfunctions in the CRD System. (continued)

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

The capability to insert the control rods ensures the assumptions for scram reactivity in the design basis transient and accident analyses are not violated. Since the SHUTDOWN MARGIN (SDM) ensures the reactor will be subcritical with the strongest control rod withdrawn (assumed single failure), the additional failure of a second control rod to insert, if required, could invalidate the demonstrated SDM and potentially limit the ability of the CRD System to hold the reactor subcritical. If the control rod is stuck at an inserted position and becomes decoupled from the CRD, a control rod drop accident (CRDA) can possibly occur. Therefore, the requirement that all control rods be OPERABLE ensures the CRD System can perform its intended function.

The control rods also protect the fuel from damage which could result in release of radioactivity. The limits protected are the Safety Limit MINIMUM CRITICAL POWER RATIO (see Bases for LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)"), the 1% cladding plastic strain fuel design limit (see Bases for LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)") and the fuel damage limit (see Bases for LCO 3.1.6, "Rod Pattern Control") during reactivity insertion events.

The negative reactivity insertion (scram) provided by the CRD System provides the analytical basis for determination of plant thermal limits and provides protection against fuel damage limits during a CRDA. Bases for LCO 3.1.4, LCO 3.1.5, and LCO 3.1.6 discuss in more detail how the Safety Limits are protected by the CRD System.

Control Rod OPERABILITY satisfies Criterion 3 of the NRC Interim Policy Statement.

LCO

OPERABILITY of an individual control rod is based on a combination of factors, primarily the scram insertion times, the associated control rod accumulator status, the control rod coupling integrity, and the ability to determine the control rod position. Although not all control rods are required to be OPERABLE to satisfy the intended reactivity control requirements, strict control over the number and distribution of inoperable control rods is required to

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BASES (continued)

LCO (continued)

satisfy the assumptions of the design basis transient and accident analyses.

[For this facility, an OPERABLE control rod constitutes the following:]

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

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

[For this facility, the number of reed switch positions required to be OPERABLE for the control rod to be OPERABLE are as follows:]

APPLICABILITY

In MODES 1 and 2, the control rods are assumed to function during a DBA or transient and are therefore required to be OPERABLE in these MODES. In MODES 3 and 4, control rods are only allowed to be withdrawn under Special Operations LCO 3.10.3, "Single Control Rod Withdrawal—Hot Shutdown," and LCO 3.10.4, "Single Control Rod Withdrawal—Cold Shutdown," which provide adequate requirements for control rod OPERABILITY during these conditions. Control rod requirements in MODE 5 are located in LCO 3.9.5.

A Note is added to provide clarification that all control rods are treated as an entity for this LCO with a single Completion Time.

ACTIONS

If the required number of reed switch positions per control rod are found inoperable, the associated control rod must be declared inoperable.

A.1

With one withdrawn control rod stuck, the control rod must be restored to OPERABLE status within 1 hour. A control rod is considered stuck if it will not insert by either CRD drive water or scram pressure. The 1-hour Completion Time

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BASES (continued)

ACTIONS (continued)

is acceptable considering the reactor can still be shutdown assuming no failures of additional control rods to insert. With a fully inserted control rod stuck, no actions are required as long as the control rod remains fully inserted. The Required Action is modified by a Note which allows the rod worth minimizer (RWM) to be bypassed if required to allow continued operation. LCO 3.3.2.1 provides additional requirements when the RWM is bypassed to ensure compliance with the CRDA analysis.

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

With one withdrawn control rod stuck for more than 1 hour the control rod must be disarmed in 1 hour. The 1-hour Completion Time is acceptable considering the reactor can still be shutdown assuming no additional control rods fail to insert and provides a reasonable time to perform the Required Action in an orderly manner. Isolating the control rod from scram prevents damage to the CDRM. The control rod can be isolated from scram and normal insert or withdraw pressure, yet still maintain cooling water to the CRD.

Out-of-sequence control rods may increase the potential reactivity worth of a dropped control rod during a CRDA. Below [10]% RATED THERMAL POWER (RTP), the generic banked position withdrawal sequence (BPWS) analysis requires inserted control rods not in compliance with BPWS to be separated by at least two OPERABLE control rods in all directions, including the diagonal. The separation of inoperable control rods must be verified to comply with these requirements within 1 hour. The 1-hour Completion Time is acceptable considering the low probability of a CRDA occurring. [For this facility the reason that this action is in effect only when operating at less than [10]% RTP is as follows:]

Monitoring of the insertion capability of each withdrawn control rod must also be performed within 24 hours. SR 3.1.3.2 and SR 3.1.3.3 perform periodic tests of the control rod insertion capability of withdrawn control rods. Testing each withdrawn control rod ensures a generic problem does not exist. The 24-hour Completion Time provides a reasonable time to test the control rods considering the potential for a need to reduce power to perform the tests.

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BASES (continued)

ACTIONS (continued)

The Required Action is modified by a Note which states that the requirement is not applicable when below the actual low power setpoint (LPSP) of the RWM since the notch insertions may not be compatible with the requirements of rod pattern control (in LCO 3.1.6) and the RWM (in LCO 3.3.2.1).

To allow continued operation with a withdrawn control rod stuck, an evaluation of adequate SDM is also required within 72 hours. Should a design basis transient or accident require a shutdown, to preserve the single failure criterion, an additional control rod would have to be assumed to fail to insert when required. Therefore, the original SDM demonstration may not be valid. The SDM must therefore be evaluated (by measurement or analysis) with the stuck control rod at its stuck position and the highest worth OPERABLE control rod assumed to be fully withdrawn.

The 72-hour Completion Time to verify SDM is adequate considering that with a single control rod stuck in a withdrawn position, the remaining OPERABLE control rods are capable of providing the required scram and shutdown reactivity. Failure to reach MODE 4 is only likely if an additional control rod adjacent to the stuck control rod also fails to insert during a required scram. Even with the postulated additional single failure of an adjacent control rod to insert, sufficient reactivity control remains to reach and maintain MODE 3 conditions. Required Action B.3 of LCO 3.1.3 performs a notch test on each remaining withdrawn control rod to ensure that no additional control rods are stuck.

<u>C.1</u>

The plant must be placed in a MODE in which the LCO does not apply if the Required Actions and associated Completion Times of Condition B cannot be met. This is done by placing the plant in MODE 3 within 12 hours. Insertion of the remainder of the control rods eliminates the possibility of an additional failure of a control rod to insert. The 12-hour Completion Time is reasonable, based on operating experience, to reach the required MODE in an orderly manner from full power and without challenging plant systems.

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ACTIONS (continued)

D.1 and D.2

With two or more withdrawn control rods stuck, the stuck control rods should be isolated from scram pressure within 1 hour and the plant placed in MODE 3 within 12 hours. The 1-hour Completion Time is acceptable considering the low probability of a CRDA occurring during this interval. The occurrence of more than one control rod without insertion capability may be an indication of a generic problem in the CRD System that could potentially cause additional failures of control rods to insert. Insertion of all insertable control rods eliminates the possibility of an additional failure of a control rod to insert. The 12-hour Completion Time is reasonable, based on operating experience, to reach the required MODE in an orderly manner from full power and without challenging plant systems.

E.1

With [8 or fewer] control rods inoperable for reasons other than being stuck, the control rods must be restored to OPERABLE status within 2 hours. [For this facility, the other reasons for considering control rods inoperable, in addition to control rod scram time "slow," are as follows:] The 2-hour Completion Time provides a period of time to correct the problem commensurate with the importance of [8 or fewer] control rods relative to the available number of OPERABLE control rods capable of providing the required scram and shutdown reactivity.

F.1, F.2, and F.3

With the inoperable control rods not restored and the associated Completion Time not met, operation may continue provided the control rods are fully inserted within 1 hour and disarmed (electrically or hydraulically) within 2 hours. Inserting a control rod ensures the shutdown and scram capabilities are not adversely affected. The control rod is disarmed to prevent inadvertent withdrawal during subsequent operations. The control rods can be hydraulically disarmed by closing the drive water and exhaust water isolation valves. The control rods can be electrically disarmed by disconnecting power from all four directional control valve solenoids. Required Action F.1 is modified by a Note which allows the RWM to be bypassed if required to allow insertion

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BASES (continued)

ACTIONS (continued)

of the inoperable control rods and continued operation. LCO 3.3.2.1 provides additional requirements when the RWM is bypassed to ensure compliance with the CRDA analysis. Inserted out-of-sequence control rods may increase the potential reactivity worth of a dropped control rod during a CRDA, and, therefore, the number and distribution of inserted inoperable control rods must be verified within 2 hours. Required Action F.3 is modified by a Note which states that the Required Action is not applicable when above [10]% RTP. Below [10]% RTP, the generic BPWS analysis requires inserted control rods, not in compliance with BPWS, to be separated by at least two OPERABLE control rods in all directions including the diagonal.

The allowed Completion Times are reasonable considering the small number of allowed inoperable control rods and provide time to insert and disarm the control rods in an orderly manner and without challenging plant systems.

G.1

The plant must be placed in a MODE in which the LCO does not apply if the Required Actions and associated Completion Times of Condition F are not met or more than 8 inoperable control rods exist. This is done by placing the plant in MODE 3 within 12 hours. This ensures all insertable control rods are inserted and places the reactor in a condition that does not require the active function (i.e., scram or insertion) of the control rods. The number of control rods permitted to be inoperable when operating above [10]% RTP (e.g., no CRDA considerations) could be more than the value specified, but the occurrence of a large number of inoperable control rods could be indicative of a generic problem, and investigation and resolution of the potential problem should be undertaken. The 12-hour Completion Time is reasonable, based on operating experience, to reach the required MODE in an orderly manner from full power and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.3.1

Determining the position of each control rod is required to ensure adequate information on control rod position is

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

available to the operator for determining CRD OPERABILITY and controlling rod patterns. Control rod position may be determined by the use of OPERABLE position indicators, by moving control rods to a position with an OPERABLE indicator, or by the use of other appropriate methods. The 24-hour Frequency of this SR was developed considering operating experience related to expected changes in control rod position and the availability of control rod position indication in the control room.

SR 3.1.3.2 and SR 3.1.3.3

Control rod insertion capability is demonstrated by inserting each partially or fully withdrawn control rods at least one notch and observing that the control rod moves. The control rod may then be returned to its original position. These surveillances are not required when below the actual LPSP of the RWM, since the notch insertions may not be compatible with the requirements of rod pattern control (LCO 3.1.6) and the RWM (LCO 3.3.2.1). The 7-day Frequency of SR 3.1.3.2 was developed considering operating experience related to changes in CRD performance and the ease of performing notch testing for fully withdrawn control rods. Partially withdrawn control rods are tested with a 31-day Frequency, based on the potential power reduction required to allow the control rod movement and considering the large testing sample of SR 3.1.3.2. Furthermore, the 31-day Frequency takes into account operating experience related to changes in CRD performance.

SR 3.1.3.4

Verifying that the scram time for each control rod to notch position [06] is less than or equal to [7] seconds ensures that the control rod will insert when required during a DBA or transient, thereby completing its shutdown function ([Ref.]). This SR is performed in conjunction with the control rod scram time testing of SR 3.1.4.2, SR 3.1.4.3, and SR 3.1.4.4. The associated Frequencies are acceptable considering the more frequent testing performed to demonstrate other aspects of control rod OPERABILITY and operating experience, which shows scram times do not significantly change over an operating cycle.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.3.5

Coupling verification is performed to ensure the control rod is connected to the CRDM and will perform its intended function when necessary. The surveillance requires verifying a control rod does not go to the overtravel position when it is fully withdrawn. The overtravel position feature provides a positive check on the coupling integrity since only an uncoupled CRD can reach the overtravel position. The verification is required to be performed any time a control rod is withdrawn to the "Full Out" (notch position 48) position or prior to declaring the control rod OPERABLE when work on the control rod or CRD System could affect coupling. This includes control rods inserted one notch and then returned to the "Full Out" position during the performance of SR 3.1.3.2. This Frequency is acceptable considering the low probability that a control rod will become uncoupled when it is not being moved and operating experience related to uncoupling events.

REFERENCES

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

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.4 Control Rod Scram Times

BASES

BACKGROUND

The scram function of the Control Rod Drive (CRD) System controls reactivity changes during abnormal operational transients to ensure that specified acceptable fuel design limits are not exceeded (Ref. 1). The control rods are scrammed by positive means using hydraulic pressure exerted on the CRD piston.

When a scram signal is initiated, control air is vented from the scram valves, allowing them to open by spring action. Opening the exhaust valve reduces the pressure above the main drive piston to atmospheric pressure and opening the inlet valve applies the accumulator or reactor pressure to the bottom of the piston. Since the notches in the index tube are tapered on the lower edge, the collet fingers are forced open by cam action, allowing the index tube to move upward without restriction because of the high differential pressure across the piston. As the drive moves upward and the accumulator pressure reduces below the reactor pressure, a ball check valve opens, letting the reactor pressure complete the scram action. If the reactor pressure is low, such as during startup, the accumulator will fully insert the control rod in the required time without assistance from reactor pressure.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the control rod scram function are presented in References 2, 3, and 4. The design basis transient and accident analyses assume that all of the control rods scram at a specified insertion rate, which is defined by the time to fully insert from a given notch position as specified in the scram times in Table 3.1.4-1. The resulting negative scram reactivity forms the basis for the determination of plant thermal limits, e.g., the MINIMUM CRITICAL POWER RATIO (MCPR). Other distributions of scram times (e.g., several control rods scramming slower than the average time with several control rods scramming faster than the average time) can also provide sufficient scram reactivity. Surveillance

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

of each individual control rod's scram time ensures that the scram reactivity assumed in the design basis transient and accident analyses can be met.

The scram function of the CRD System protects the Safety Limit MCPR (see Bases for LCO 3.2.2) and the 1% cladding plastic strain fuel design limit (see Bases for LCO 3.2.1), which ensure that no fuel damage will occur if these limits are not exceeded. Above [800] psig, the scram function is designed to insert negative reactivity at a rate fast enough to prevent the actual MCPR from becoming less than the Safety Limit MCPR during the analyzed limiting power transient. Below [800] psig, the scram function is assumed to function during the control rod drop accident (Ref. 5) and, therefore, also provides protection against violating fuel damage limits during reactivity insertion accidents (see Bases for LCO 3.1.6). For the reactor vessel overpressure protection analysis, the scram function, along with the safety/relief valves, ensures that the peak vessel pressure is maintained within the applicable American Society for Mechanical Engineers Code limits.

Control rod scram times satisfies Criterion 3 of the NRC Interim Policy Statement.

LCO

The scram times specified in Table 3.1.4-1 are required to ensure that the scram reactivity assumed in the design basis transient and accident analysis is met. To account for single failures and "slow" scramming control rods, the scram times specified in Table 3.1.4-1 are faster than those assumed in the design basis analysis. The scram times have a margin that allows up to 7.5% of the control rods (e.g., [137] x 7.5% = [10]) to have scram times exceeding the specified limits (i.e., "slow" control rods) and that also assumes a single stuck control rod (as allowed by LCO 3.1.3) and an additional control rod failing to scram per the single failure criterion (Ref. 6). The scram times are specified as a function of reactor steam dome pressure to account for the pressure dependence of the scram times. The scram times are specified relative to measurements based on reed switch positions, which provide the control rod position indication. The reed switch closes ("pickup") when

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BASES (continued)

LCO (continued)

the index tube passes a specific location and then opens ("dropout") as the index tube travels upward. Verification of the specified scram times in Table 3.1.4-1 is accomplished through measurement of the "dropout" times.

To ensure that local scram reactivity rates are maintained within acceptable limits, no more than two of the allowed "slow" control rods can occupy adjacent locations.

Table 3.1.4-1 is modified by two Notes, which state that control rods with scram times not within the limits of the table are considered "slow" and that control rods with scram times greater than [7] seconds are considered inoperable as required by SR 3.1.3.4.

APPLICABILITY

In MODES 1 and 2, a scram is assumed to function during transients and accidents analyzed for these plant conditions. These events are assumed to occur during startup and power operation and, therefore, the scram function of the control rods is required during these MODES. In MODES 3 and 4, the control rods are only allowed to be withdrawn under Special Operations LCO 3.10.3, "Single Control Rod Withdrawal—Hot Shutdown," and LCO 3.10.4, "Single Control Rod Withdrawal—Cold Shutdown," which provide adequate requirements for control rod scram capability during these conditions. Scram requirements in MODE 5 are contained in LCO 3.9.5.

ACTIONS

A.1

With the requirements of this LCO not met, the plant must be placed in a MODE in which the LCO does not apply. This is done by placing the plant in MODE 3 within 12 hours. The 12-hour Completion Time is reasonable, based on operating experience related to the amount of time required, to reach MODE 3 from full power in an orderly manner and without challenging plant systems.

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BASES (continued)

SURVEILLANCE REQUIREMENTS

All four SRs of this LCO are modified by a Note that states that during a single control rod scram time test, the CRD pumps shall be isolated from the associated scram accumulator. [For this facility, the reason for this is as follows:]

SR 3.1.4.1

The scram reactivity used in design basis transient and accident analyses is based on an assumed control rod scram time. Measurement of the scram times with reactor steam dome pressure ≥ [800] psig demonstrates acceptable scram times for the transients analyzed in References 3 and 4.

Maximum scram insertion times occur at a reactor steam dome pressure of approximately [800] psig because of the competing effects of reactor steam dome pressure and stored accumulator energy. Therefore, demonstration of adequate scram times at reactor steam dome pressure > [800] psig ensures that the measured scram times will be within the specified limits at higher pressures. Limits are specified as a function of reactor pressure to account for the sensitivity of the scram insertion times with pressure and to allow a range of pressures over which scram time testing can be performed. To ensure that scram time testing is performed within a reasonable time following a refueling or after a shutdown 120 days or longer, all control rods are required to be tested before exceeding 40% RATED THERMAL POWER (RTP) following the shutdown. This Frequency is acceptable considering the additional surveillances performed for control rod OPERABILITY, the frequent verification of adequate accumulator pressure, and the required testing of control rods affected by work on control rods or the CRD System.

SR 3.1.4.2

Additional testing of a sample of control rods is required to verify the continued performance of the scram function during the cycle. A representative sample must contain at least 10% of the control rods and no more than 20% of the control rods in the sample can be "slow" (Ref. 7). With more than 20% of the sample declared to be "slow" per the criterion in Table 3.1.4-1, additional control rods must be tested until this 20% criterion is satisfied or Required Action A.1 must be taken. For planned testing, the control

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

rods selected for the sample should be different for each test. Data from inadvertent scrams should be used whenever possible to avoid unnecessary testing at power, even if the control rods with data may have been previously tested in a sample. The 120-day Frequency is based on operating experience that has shown that control rod scram times do not significantly change over an operating cycle. This Frequency is also reasonable based on the additional surveillances done on the CRDs at more frequent intervals in accordance with LCO 3.1.3 and LCO 3.1.5.

SR 3.1.4.3

When work is performed on a control rod or the CRD System, which could affect the scram insertion time, testing must be done to demonstrate that each affected control rod retains adequate scram performance over the range of applicable reactor pressures from zero to the maximum permissible pressure.

For work done while the reactor is at less than [800] psig, the scram testing must be performed once before declaring the control rod OPERABLE. The required scram time testing must demonstrate that the affected control rod is still within the limits of Table 3.1.4-1 for startup conditions.

The Frequency of once prior to declaring the affected control rod(s) OPERABLE is acceptable because of the capability to test the control rods over a range of operating conditions and the more frequent surveillances on other aspects of control rod OPERABILITY.

Specific examples of work that could affect the scram times are (but not limited to) the following: removal of any CRD for maintenance or modification; replacement of a control rod; and maintenance or modification of a scram solenoid pilot valve, scram valve, accumulator, or isolation or check valve in the piping required for scram.

SR 3.1.4.4

When work is performed on a control rod or CRD System which could affect the scram insertion time, testing must be done to demonstrate that each affected control rod is still

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

within the limits of Table 3.1.4-1 with the reactor steam dome pressure ≥ [800] psig. This testing ensures that, prior to withdrawing the control rod for continued operation, the control rod scram performance is acceptable for operating reactor pressure conditions. Where work has been performed at high reactor pressure, the requirements of SR 3.1.4.3 and SR 3.1.4.4 can be satisfied with one test. For a control rod affected by work performed while shutdown, however, a zero pressure and high-pressure test may be required. Alternatively, a control rod scram test during hydrostatic pressure testing could also satisfy both criteria.

The Frequency of once prior to exceeding 40% of RTP is acceptable because of the capability to test the control rods over a range of operating conditions and the more frequent surveillances on other aspects of control rod OPERABILITY.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix A, General Design Criterion 10, "Reactor Design."
2. [Unit Name] FSAR, Section [ ], "[Title]."
3. [Unit Name] FSAR, Section [ ], "[Title]."
4. [Unit Name] FSAR, Section [ ], "[Title]."
5. NEDE-24011-P-A-9, "General Electric Standard Application for Reload Fuel," Supplement for United States, Section 3.2.4.1, September 1988.
6. Title 10, Code of Federal Regulations, Part 50, Appendix A, General Design Criterion 21, "Protection System Reliability and Stability."
7. [Unit Name] [Reference for sampling technique related to SR 3.1.4.2].

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.5 Control Rod Scram Accumulators

BASES

BACKGROUND	The control rod scram accumulators are part of the Control Rod Drive (CRD) System and are provided to ensure that the control rods scram under varying reactor conditions. The control rod scram accumulators store sufficient energy to fully insert a control rod at any reactor vessel pressure. The accumulator is a hydraulic cylinder with a free-floating piston. The piston separates the water used to scram the control rods from the nitrogen which provides the required energy. The scram accumulators are necessary to scram the control rods within the required insertion times of LCO 3.1.4 and SR 3.1.3.4.
APPLICABLE SAFETY ANALYSES	The analytical methods and assumptions used in evaluating the control rod scram function are presented in References 1, 2, and 3. The design basis transient and accident analyses assume that all of the control rods scram at a specified insertion rate, which is defined as the time to fully insert from a given notch position as specified in the scram times in Table 3.1.4-1. OPERABILITY of each individual control rod scram accumulator, as required by LCOs 3.1.3 and 3.1.4, ensures that the scram reactivity assumed in the design basis transient and accident analyses can be met. The existence of an inoperable accumulator may invalidate prior scram time measurements for the associated control rod.
	The scram function of the CRD System, and therefore the OPERABILITY of the accumulators, protects the Safety Limit MINIMUM CRITICAL POWER RATIO (see Bases for LCO 3.2.2) and 1% cladding plastic strain fuel design limit (see Bases for LCO 3.2.1), which ensure that no fuel damage will occur if these limits are not exceeded (see Bases for LCO 3.1.4). In addition, the scram function at low reactor vessel pressure (i.e., startup conditions) provides protection against violating these limits during reactivity insertion accidents (see Bases for LCO 3.1.6). Control rod scram accumulators satisfies Criterion 3 of the NRC Interim Policy Statement.

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BASES (continued)

LCO

The OPERABILITY of the control rod scram accumulators is required to ensure that adequate scram insertion capability exists when needed over the entire range of reactor pressures. The OPERABILITY of the scram accumulators is based on maintaining adequate accumulator pressure.

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

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

APPLICABILITY

In MODES 1 and 2, the scram function is required for mitigation of Design Basis Accidents (DBAs) and transients, and, therefore, the scram accumulators must be OPERABLE to support the scram function. In MODES 3 and 4, control rods are only allowed to be withdrawn under Special Operations LCO 3.10.3, "Single Control Rod Withdrawal—Hot Shutdown," and LCO 3.10.4, "Single Control Rod Withdrawal—Cold Shutdown," which provide adequate requirements for control rod scram accumulator OPERABILITY during these conditions. Requirements for scram accumulators in MODE 5 are contained in LCO 3.9.5.

A Note is added to indicate that when the pressure in any one of the accumulators cannot be verified, this LCO must be entered and the applicable Required Actions of Conditions A, B, and C apply.

A second Note is added to provide clarification that all control rod scram accumulators are treated as an entity for this LCO with a single Completion Time.

ACTIONS

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

With one control rod scram accumulator inoperable and the reactor steam dome pressure ≥ [800] psig, the scram

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BASES (continued)

ACTIONS (continued)

accumulator must be restored to OPERABLE status within 8 hours. Alternatively, the control rod may be declared "slow," since the control rod will still scram at the reactor operating pressure but may not satisfy the required scram times in Table 3.1.4-1. In addition, the associated control rod is declared inoperable and LCO 3.1.3 is entered.

[Instead of declaring both the associated control rod "slow" and inoperable, a facility may consider selecting either alternative if the control rod status does not invalidate the NRC staff-approved licensing basis for that facility as related to control rod OPERABILITY].

The 8-hour Completion Time for both of these last two Required Actions is considered reasonable, based on the large number of control rods available to provide the scram function and the ability of the affected control rod to scram only with reactor pressure at high reactor pressures.

B.1, B.2.1, B.2.2.1, and B.2.2.2

With two or more control rod scram accumulators inoperable and reactor steam dome pressure ≥ [800] psig it must be verified within 20 minutes that adequate pressure is being supplied to the charging water header (i.e., the pressure supplied to the charging water header is ≥ [940] psig). With inadequate charging water pressure, all of the accumulators could become inoperable, resulting in a potentially severe degradation of the scram performance. The 20-minute Completion Time is considered a reasonable time to verify the pressure and place a CRD pump into service to restore the charging header pressure, if required. This Completion Time also recognizes the ability of the reactor pressure alone to fully insert all control rods. The scram accumulators must also be restored to OPERABLE status within 1 hour. Alternatively, the control rod may be declared "slow," since the control rod will still scram using only reactor pressure but may not satisfy the times in Table 3.1.4-1. In addition, the associated control rod is declared inoperable and LCO 3.1.3 is entered.

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BASES (continued)

ACTIONS (continued)

The 1-hour Completion Time is considered reasonable, based on the ability of only the reactor pressure to scram the control rods and the low probability of a DBA or severe transient occurring while the affected accumulators are inoperable.

[As discussed under the Required Actions of Condition A, a facility may consider implementing either declaring the control rod scram times "slow" or declaring the control rod inoperable, if justified in the manner described under the Required Actions of Condition A.]

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

With one or more control rod scram accumulators inoperable and the reactor steam dome pressure < [800] psig, the pressure supplied to the charging water header must be immediately verified to be adequate by checking available control room indications (i.e., the pressure supplied to the charged water header is ≥ [940] psig). With the reactor steam dome pressure < [800] psig, the function of the accumulators in providing the scram force becomes much more important since the scram function could become severely degraded during a depressurization event or at low reactor pressures. The associated control rods must also be restored to OPERABLE status or declared inoperable within 1 hour. The 1-hour Completion Time is reasonable for either of these last two Required Actions, considering the low probability of a DBA or severe transient occurring during the time that the accumulator(s) is inoperable.

D.1

Condition D is entered when the Required Actions and associated Completion Times of Condition A, B, or C are not met, OR reactor steady dome pressure is inoperable, or charging water header pressure is inoperable, AND one or more control rod scram accumulators are inoperable, or one or more control rod scram times are "slow."

When Condition D is entered the reactor mode switch must be immediately placed in the shutdown position. This ensures that all insertable control rods are inserted and that the reactor is in a condition that does not require the active

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BASES (continued)

ACTIONS (continued)

function (i.e., scram and insertion) of the control rods. This Required Action is modified by a Note that states that the ACTION is not applicable if all control rods associated with the inoperable scram accumulators are fully inserted.

SURVEILLANCE REQUIREMENTS

SR 3.1.5.1

SR 3.1.5.1 requires that the accumulator pressure be checked every 7 days to ensure that adequate accumulator pressure exists to provide sufficient scram force. The primary indicator of accumulator OPERABILITY is the accumulator pressure. A minimum accumulator pressure is specified, below which the capability of the accumulator to perform its intended function becomes degraded and the accumulator is considered inoperable. The minimum accumulator pressure of [940] psig is well below the expected pressure of [1100] psig (Ref. 1). Declaring the accumulator inoperable when the minimum pressure is not maintained ensures that significant degradation in scram times does not occur. The 7-day Frequency has been shown to be acceptable through operating experience and takes into account other indications available in the control room. [For this facility these other indications constitute the following:]

It should be noted that in this surveillance the supported control rod is not declared inoperable when the associated support scram accumulator is found inoperable. The Required Actions of LCO 3.1.5 determine when it is appropriate to declare the associated control rod scram time "slow" and control rods inoperable.

REFERENCES

1. [Unit Name] FSAR, Section [ ], "[Title]."
2. [Unit Name] FSAR, Section [5], "[Title]."
3. [Unit Name] FSAR, Section [15], "[Title]."

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.6 Rod Pattern Control

BASES

BACKGROUND

Control rod patterns during startup conditions are controlled by the operator and the rod worth minimizer (RWM) (LCO 3.3.2.1), so that only specified control rod sequences and relative positions are allowed over the operating range from all control rods inserted to [10%] RATED THERMAL POWER (RTP). The sequences limit the potential amount of reactivity addition that could occur in the event of a Control Rod Drop Accident (CRDA).

This specification assures that the control rod patterns are consistent with the assumptions of the CRDA analyses of References 1 and 2.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the CRDA are summarized in References 1 and 2. CRDA analyses assume that the reactor operator follows prescribed withdrawal sequences. These sequences define the potential initial conditions for the CRDA analysis. The RWM (LCO 3.3.2.1) provides backup to operator control of the withdrawal sequences to ensure that the initial conditions of the CRDA analysis are not violated.

Prevention or mitigation of positive reactivity insertion events is necessary to limit the energy deposition in the fuel, thereby preventing significant fuel damage which could result in the undue release of radioactivity. Since the failure consequences for UO_2 have been shown to be insignificant below fuel energy depositions of 300 cal/gm (Ref. 3), the fuel damage limit of 280 cal/gm provides a margin of safety from significant core damage which would result in release of radioactivity (Refs. 4 and 5). Generic evaluations (Refs. 1 and 6) of a design basis CRDA (i.e., a CRDA resulting in a peak fuel energy deposition of 280 cal/gm) have shown that if the peak fuel enthalpy remains below 280 cal/gm, then the maximum reactor pressure will be less than the required American Society of

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

Mechanical Engineers Code limits (Ref. 4) and the calculated offsite doses will be well within the required limits (Ref. 5).

Control rod patterns analyzed in Reference 1 follow the banked position withdrawal sequence (BPWS). The BPWS is applicable from the condition of all control rods fully inserted to [10%] RTP (Ref. 2). For the BPWS, the control rods are required to be moved in groups, with all control rods assigned to a specific group required to be within specified banked positions (e.g., between notches 08 and 12). The banked positions are established to minimize the maximum incremental control rod worth without being overly restrictive during normal plant operation. Generic analysis of the BPWS (Ref. 1) has demonstrated that the 280 cal/gm fuel damage limit will not be violated during a CRDA while following the BPWS MODE of operation. The generic BPWS analysis [Plant Specific Reference [ ]] also evaluates the effect of fully inserted, inoperable control rods not in compliance with the sequence, to allow a limited number (i.e., eight) and distribution of fully inserted, inoperable control rods.

Rod pattern control satisfies Criterion 3 of the NRC Interim Policy Statement.

LCO

Compliance with the prescribed control rod sequences minimizes the potential consequences of a CRDA by limiting the initial conditions to those consistent with the BPWS. This LCO only applies to OPERABLE control rods. For inoperable control rods required to be inserted, separate requirements are specified in LCO 3.1.2, consistent with the allowances for inoperable control rods in the BPWS.

APPLICABILITY

In MODES 1 and 2, when THERMAL POWER is ≤ [[10%]] RTP, the CRDA is a Design Basis Accident and, therefore, compliance with the assumptions of the safety analysis is required. When THERMAL POWER is greater than [10%] RTP, there is no credible control rod configuration that results in a control rod worth that could exceed the 280 cal/gm fuel damage limit during a CRDA (Ref. 2). In MODES 3, 4, and 5, since the

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BASES (continued)

APPLICABILITY (continued)

reactor is shut down and only a single control rod can be withdrawn from a core cell containing fuel assemblies, adequate SHUTDOWN MARGIN ensures that the consequences of a CRDA are acceptable, since the reactor will remain subcritical with a single control rod withdrawn.

ACTIONS

A.1 and A.2

With eight or fewer OPERABLE control rods not in compliance with the prescribed control rod sequence, ACTIONS may be taken to either correct the control rod pattern or declare the associated control rods inoperable within 8 hours. Noncompliance with the prescribed sequence may be the result of "double notching," drifting from a control rod drive cooling water transient, leaking scram valves, or a power reduction to ≤ [10%] RTP before establishing the correct control rod pattern. The number of OPERABLE control rods not in compliance with the prescribed sequence is limited to eight to prevent the operator from attempting to correct a control rod pattern that significantly deviates from the prescribed sequence. When the control rod pattern is not in compliance with the prescribed sequence, all control rod movement should be stopped except for moves needed to correct the rod pattern, or scram if warranted.

Required Action A.1 is modified by a Note which allows the RWM to be bypassed to allow the affected control rods to be returned to their correct position. LCO 3.3.1.2 requires verification of control rod movement by a qualified member of the technical staff (which includes, but is not limited to, a second licensed operator). This ensures that the control rods will be moved to the correct position. A control rod not in compliance with the prescribed sequence is not considered inoperable except as required by Required Action A.2. OPERABILITY of control rods is determined by compliance with LCO 3.1.3 through LCO 3.1.5. The 8-hour Completion Time is reasonable, considering the restrictions on the number of allowed out of sequence control rods and the low probability of a CRDA occurring during the time the control rods are out of sequence.

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BASES (continued)

ACTIONS (continued)

B.1 and B.2

If nine or more OPERABLE control rods are out of sequence, the control rod pattern significantly deviates from the prescribed sequence. Control rod withdrawal should be suspended immediately to prevent the potential for further deviation from the prescribed sequence. Control rod insertion to correct control rods withdrawn beyond their allowed position is allowed since, in general, insertion of control rods has less impact on control rod worth than withdrawals. When nine or more OPERABLE control rods are not in compliance with BPWS, the reactor mode switch must be placed in the shutdown position within 1 hour. With the mode switch in shutdown, the reactor is shut down and as such does not meet the applicability requirements of this LCO. The 1-hour Completion Time is reasonable to allow insertion of control rods to restore compliance and is short relative to the low probability of a CRDA occurring with the control rods out of sequence.

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1

The control rod pattern is verified at a 24-hour Frequency to be in compliance with the BPWS to ensure that the assumptions of the CRDA analyses are met. The 24-hour Frequency of this SR was developed considering that the primary check on compliance with the BPWS is performed by the RWM (LCO 3.3.2.1), which provides control rod blocks to enforce the required sequence and is required to be OPERABLE when operating at ≤ [10%] RTP.

SR 3.1.6.2

[For this facility the purpose of this SR is as follows:]

REFERENCES

1. NEDE-24011-P-A-9-US, "General Electric Standard Application for Reactor Fuel, Supplemental for United States," Section 2.2.3.1, September 1988.

2. [Unit Name] ["Title."]

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BASES (continued)

REFERENCES (continued)

3. NUREG-0979, "NRC Safety Evaluation Report for GESSAR II BWR/6 Nuclear Island Design, Docket No. 447," Section 4.2.1.3.2, April 1983.

4. NUREG-0800, "Standard Review Plan," Section 15.4.9, "Radiological Consequences of Control Rod Drop Accident (BWR)," Revision 2, July 1981.

5. Title 10, Code of Federal Regulations, Part 100.11, "Determination of Exclusion Area Low Population Zone and Population Center Distance."

6. NEDO-21778-A, "Transient Pressure Rises Affected Fracture Toughness Requirements for Boiling Water Reactors," December 1978.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.7 Standby Liquid Control (SLC) System

BASES

BACKGROUND

The SLC System is designed to provide the capability of bringing the reactor, at any time in a fuel cycle, from full power and minimum control rod inventory (which is at the peak of the xenon transient) to a subcritical condition with the reactor in the most reactive, xenon-free state without taking credit for control rod movement.

The SLC System consists of a boron solution storage tank, two positive displacement pumps, two explosive valves that are provided in parallel for redundancy, and associated piping and valves used to transfer borated water from the storage tank to the reactor pressure vessel (RPV). [The borated solution is discharged near the bottom of the core shroud, where it then mixes with the cooling water rising through the core. A smaller tank containing demineralized water is provided for testing purposes.]

APPLICABLE SAFETY ANALYSES

[The SLC System is manually initiated from the main control room as directed by the emergency operating procedures if the operator believes the reactor cannot be shut down, or kept shut down, with the control rods.] The SLC System is used in the event that enough control rods cannot be inserted to accomplish shutdown and cooldown in the normal manner. The SLC System injects borated water into the reactor core to add negative reactivity to compensate for all of the various reactivity effects that could occur during plant operations. To meet this objective, it is necessary to inject a quantity of boron, which produces a concentration of 660 ppm of natural boron, in the reactor coolant at 68°F. To allow for potential leakage and imperfect mixing in the reactor system, an additional amount of boron equal to 25% of the amount cited above is added (Ref. 1). The volume-versus-concentration limits in Figure 3.1.7-1 and the temperature versus concentration limits in Figure 3.1.7-2 are calculated such that the required concentration is achieved accounting for dilution in the RPV with normal water level and including the water volume in one loop of the residual heat removal shutdown

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

cooling piping and in the recirculation loop piping. This quantity of borated solution is the amount that is above the pump suction shutoff level in the boron solution storage tank. No credit is taken for the portion of the tank volume that cannot be injected.

The SLC System satisfies the requirements of the NRC Interim Policy Statement because operating experience and probabilistic risk assessments have shown SLC System to be important to public health and safety. Thus, it is retained in the Technical Specifications.

LCO

The OPERABILITY of the SLC System provides backup capability for reactivity control independent of normal reactivity control provisions provided by the control rods. The OPERABILITY of the SLC System is based on the conditions of the borated solution in the storage tank and the availability of a flow path to the RPV, including the OPERABILITY of the pumps and valves. Two SLC System subsystems are required to be OPERABLE; each contains an OPERABLE pump, an explosive valve, and associated piping, valves, and instruments and controls to ensure an OPERABLE flow path.

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

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

APPLICABILITY

Shutdown capability is required in MODES 1 and 2. In MODES 3 and 4, control rods are only allowed to be withdrawn under Special Operations LCO 3.10.3, "Single Control Rod Withdrawal—Hot Shutdown," and LCO 3.10.4, "Single Control Rod Withdrawal—Cold Shutdown," which provide adequate controls to ensure that the reactor remains subcritical. In MODE 5, only a single control rod can be withdrawn from a core cell containing fuel assemblies; demonstration of

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BASES (continued)

APPLICABILITY (continued)

adequate SHUTDOWN MARGIN (LCO 3.1.1) ensures that the reactor will not be critical. Therefore, the SLC System is not required to be OPERABLE when only a single control rod can be withdrawn.

ACTIONS

A.1

If one SLC System subsystem is inoperable, the inoperable subsystem must be restored to OPERABLE status within 7 days. In this condition, the remaining OPERABLE subsystem is adequate to perform the shutdown function. The 7-day Completion Time is based on the availability of an OPERABLE subsystem capable of performing the intended SLC System function and the low probability of a Design Basis Accident (DBA) or severe transient occurring concurrent with the failure of the Control Rod Drive (CRD) System to shut down the plant.

B.1

If both SLC System subsystems are inoperable and there are less than a total of eight control rods stuck, scram time "slow" and inoperable, at least one subsystem must be restored to OPERABLE status within 8 hours. The 8-hour Completion Time is considered acceptable given the low probability of a DBA or severe transient occurring concurrent with the failure of the control rods to shut down the reactor.

If both SLC System subsystems are inoperable and there are a total of more than eight control rods stuck, scram time "slow" and inoperable, it may be indicative of a generic problem with the CRD system. Thus, the plant must be brought to a MODE in which the LCO does not apply. This is accomplished by entering LCO 3.0.3 immediately.

C.1

The plant must be placed in a MODE in which the LCO does not apply if the inoperable SLC System subsystems cannot be restored to OPERABLE status within the associated Completion Times of Required Actions A.1 and B.1. This is done by placing the plant in MODE 3 within 12 hours. The 12-hour

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BASES (continued)

ACTIONS (continued)

Completion Time is reasonable, based on the operating experience with regard to the amount of time required, to reach MODE 3 from full power in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.7.1, SR 3.1.7.2, and SR 3.1.7.3

SRs 3.1.7.1 through 3.1.7.3 are 24-hour surveillances verifying certain characteristics of the SLC System (e.g., the volume and temperature of the borated solution in the storage tank), thereby ensuring SLC System OPERABILITY without disturbing normal plant operation. These surveillances ensure that the proper borated solution volume and temperature, including the temperature of the pump suction piping, are maintained. Maintaining a minimum specified borated solution temperature is important in ensuring that the boron remains in solution and does not precipitate out in the storage tank or in the pump suction piping. The temperature versus concentration curve of Figure 3.1.7-2 ensures that a 10°F margin will be maintained above the saturation temperature. The 24-hour Frequency of these SRs was based on operating experience that has shown that there are relatively slow variations in the measured parameters of volume and temperature.

In the event that the required instrumentation to monitor volume and temperature of the borated solution in the SLC System storage tank are found inoperable, the affected SLC System subsystems are considered inoperable.

Failure to meet SR 3.1.7.1, SR 3.1.7.2, and SR 3.1.7.3 will make both SLC System subsystems inoperable, since the boron solution storage tank and the majority of the pump suction piping is common to both subsystems.

SR 3.1.7.4 and SR 3.1.7.6

SRs 3.1.7.4 verifies the continuity of the explosive charges in the injection valves to ensure that proper operation will occur if required. Other administrative controls, such as those that limit the shelf life of the explosive charges, must be followed. The 31-day Frequency is based on

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

operating experience that has demonstrated the reliability of the explosive charge continuity.

SR 3.1.7.6 verifies that each valve in the system is in its correct position but does not apply to the squib (i.e., explosive) valves. Verifying the correct alignment for manual, power-operated, and automatic valves in the SLC System flow path provides assurance that the proper flow paths will exist for system operation. This surveillance also does not apply to valves that are locked, sealed, or otherwise secured in position since they were verified to be in the correct position prior to locking, sealing, or securing. This verification of valve alignment does not require any testing or valve manipulation and does not apply to valves that cannot be inadvertently misaligned, such as check valves. Rather, it involves verification that those valves capable of being mispositioned are in the correct position. The 31-day Frequency for SR 3.1.7.6 is appropriate because the subject valves are operated under procedural control and it was chosen to provide added assurance that the valves are in the correct positions.

SR 3.1.7.5

This surveillance requires an examination of the sodium pentaborate solution by using chemical analysis to ensure that the proper concentration of boron exists in the storage tank. SR 3.1.7.5 must be performed any time boron or water is added to the storage tank solution to determine that the boron solution concentration is within the specified limits. SR 3.1.7.5 must also be performed any time the temperature is restored to within the limits of Figure 3.1.7-2 to ensure that no significant boron precipitation occurred. The 31-day Frequency of this surveillance is appropriate because of the relatively slow variation of boron concentration between surveillances.

SR 3.1.7.7

Demonstrating that each SLC System pump develops a flow rate $\geq$ [41.2] gpm at a discharge pressure $\geq$ [1190 psig] ensures that pump performance has not degraded during the fuel cycle. This minimum pump flow rate requirement ensures that, when combined with the sodium pentaborate solution concentration requirements, the rate of negative reactivity

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

insertion from the SLC System will adequately compensate for the positive reactivity effects encountered during power reduction, cooldown of the moderator, and xenon decay. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice inspections confirm component OPERABILITY and trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of this surveillance is in accordance with the Inservice Inspection and Testing Program, but the Frequency must not exceed 92 days.

SR 3.1.7.8

This surveillance ensures that there is a functioning flow path from the boron solution storage tank to the RPV, including the firing of an explosive valve. The replacement charge for the explosive valve shall be from the same manufactured batch as the one fired or from another batch that has been certified by having one of that batch successfully fired. The pump and explosive valve tested should be alternated such that both complete flow paths are tested every 36 months. An acceptable method for verifying that the suction piping is unblocked is to pump from the storage tank to the test tank. The 18-month Frequency was developed considering that it is prudent that many surveillances be performed only during a plant outage. This reflects the plant conditions needed to perform this surveillance and the potential for a plant transient if this surveillance is performed with the reactor at power. For this facility [Unit Name], 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.

SR 3.1.7.9

Demonstrating that all heat-traced piping between the boron solution storage tank and the suction inlet to the injection pumps is unblocked provides assurance that there is a functioning flow path for injecting the sodium pentaborate solution. The 18-month Frequency is acceptable since there is a low probability that the subject piping will be blocked due to precipitation of the boron from solution in the heat-traced piping. This is especially true in light of the

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)	verification of the temperature of this piping required by SR 3.1.7.3. However, if in performing SR 3.1.7.3 it is determined that the temperature of this piping has fallen below the specified minimum, this surveillance must be performed once within 24 hours after the piping temperature is restored to within the limits of [Figure 3.1.7-2]. <u>SR 3.1.7.10</u> Enriched sodium pentaborate solution is made by mixing granular, enriched sodium pentaborate with water. Isotopic tests on the granular sodium pentaborate to verify the actual B-10 enrichment must be performed to ensure that the proper B-10 atom percent is being used. The 18-month Frequency is acceptable considering the controls on the granular sodium pentaborate specifications.
REFERENCES	1. [Unit Name] FSAR, Section [4], "[Title]."

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.8 Scram Discharge Volume (SDV) Vent and Drain Valves

BASES

BACKGROUND

The SDV vent and drain valves function to limit reactor coolant loss due to leakage past the control rod drive (CRD) seals and to maintain sufficient SDV to accommodate the water discharge during scram. The SDV is a volume of header piping that connects to each hydraulic control unit (HCU) and drains into an instrument volume. There are two SDVs (headers) and two instrument volumes, each receiving approximately one-half of the CRD discharges. The two instrument volumes are connected to a common drain line with two valves in series. Each header is connected to a vent line with two valves in series for a total of four vent valves. The header piping is sized to receive and contain all the water discharged by the CRDs during a scram. The SDV vent and drain valves are normally open and discharge any accumulated water in the SDV to ensure that sufficient volume is available at all times to allow a complete scram. During a scram, the SDV vent and drain valves close to contain reactor water. The design and functions of the SDV are described in Reference 1.

APPLICABLE SAFETY ANALYSES

The design basis transient and accident analyses assume all of the control rods are capable of scramming. The acceptance criteria for the SDV vent and drain valves is that they operate automatically to:

a. Close during scram to limit the amount of reactor coolant discharged (leakage past the CRD seals) so that adequate core cooling is maintained and offsite doses remain within the limits of 10 CFR 100; and

b. Open on scram reset to maintain the SDV vent and drain path open so that there is sufficient volume to accept the reactor coolant discharged during a scram.

Isolation of the SDV can also be accomplished by manual (control-switch) closure of the SDV valves. Additionally, the discharge of reactor coolant to the SDV can be

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

terminated by scram reset or closure of the HCU's manual isolation valves. For a bounding leakage case, the offsite doses are well within the limits of 10 CFR 100, and adequate core cooling is maintained (Ref. 2). The SDV vent and drain valves allow continuous drainage of the SDV during normal plant operation to ensure that the SDV has sufficient capacity to contain the reactor coolant discharge during a full-core scram. To automatically ensure this capacity, a reactor scram (LCO 3.3.1.1) is initiated if the SDV water level in the instrument volume exceeds a specified setpoint. The setpoint is chosen so that all control rods are inserted before the SDV has insufficient volume to accept a full scram.

SDV vent and drain valves have been assessed as potentially risk-significant, and, therefore, are retained in the Technical Specifications.

LCO

The OPERABILITY of all SDV vent and drain valves ensures that the SDV vent and drain valves will close during a scram to limit the amount of reactor water discharged to the SDV piping. Since the vent and drain lines are provided with two valves in series, the single failure of one valve in the open position will not impair the isolation function of the system. Additionally, the valves are required to open on scram reset to ensure that a path is available for the SDV piping to drain freely at other times.

For the SDV vent and drain valves to be OPERABLE requires meeting the SRs of this LCO.

[For this facility, the following support systems are required to be OPERABLE to ensure SDV vent and drain valve OPERABILITY:]

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

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BASES (continued)

APPLICABILITY

In MODES 1 and 2, scram may be required; therefore, the SDV vent and drain valves must be OPERABLE. In MODES 3 and 4, control rods are only allowed to be withdrawn under Special Operations LCO 3.10.3, "Single Control Rod Withdrawal—Hot Shutdown," and LCO 3.10.4 "Single Control Rod Withdrawal—Cold Shutdown," which provide adequate controls to ensure that only a single control rod can be withdrawn. Also, during MODE 5, only a single control rod can be withdrawn from a core cell containing fuel assemblies. Therefore, the SDV vent and drain valves are not required to be OPERABLE in these MODES since the reactor is subcritical and only one rod may be withdrawn and subject to scram. Thus, both the water discharge volume and potential leakage past the CRD seals during scram are minimal.

A Note is added to provide clarification that all SDV vent and drain valves in this LCO are treated as an entity with a single Completion Time.

ACTIONS

A.1. and A.2

When one SDV vent or drain valve in one or both lines is inoperable, the valve(s) must be restored to OPERABLE status within 7 days and the redundant valve in the associated line must immediately be verified to be OPERABLE or the line isolated in 8 hours. The Completion Times are reasonable given the level of redundancy in the lines and the low probability of a scram occurring while the valve(s) are inoperable and the lines are not isolated. The SDV is still isolable if the redundant valve in the affected line is verified OPERABLE. During these periods, the single failure criterion will not be preserved and a higher risk exists to allow reactor water out of the primary system during a scram. The redundant valve may be verified OPERABLE through an administrative check, by examining logs or other information, to determine whether the required SDV vent or drain valves are out of service for maintenance or other reasons. It does not mean performing the SRs needed to demonstrate OPERABILITY of the valves. Should a scram be required, the OPERABLE valve in the vent and drain path will close and contain the reactor water.

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BASES (continued)

ACTIONS (continued)

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

If both valves in a line are inoperable, the line must be isolated to limit the reactor coolant leakage past the CRD seals during a scram.

In addition, when a line is isolated, the potential for an inadvertent scram due to high SDV level is increased. Required Action B.1 is modified by a Note that allows periodic draining of the SDV when a line is isolated. During these periods, the valves in the line may be opened.

Also Required Action B.2 requires that the SDV Water Level—High scram instrumentation of LCO 3.3.1.1 be verified OPERABLE. With a line isolated in each SDV, the Water Level—High trip ensures that a scram is automatically initiated while there is still sufficient SDV capacity left to accept the water discharged on a scram.

The Completion Time of 8 hours to isolate the line and verify OPERABILITY of the SDV Water Level—High scram instrumentation is based on the low probability of a scram occurring while the line is not isolated and on the simultaneous occurrence of significant CRD seal leakage.

The valves must be restored to OPERABLE status in 7 days. With the affected line(s) isolated and the SDV Water Level—High scram instrumentation verified OPERABLE, both safety functions of the vent and drain lines are ensured. However, to limit the risk of inadvertent scram on the SDV Water Level—High trip function, a 7-day Completion Time is specified to restore valves to OPERABLE status.

C.1

The plant must be placed in a Condition in which the LCO does not apply if the Required Actions and associated Completion Times are not met. This is done by placing the plant in MODE 3 within 12 hours. The 12-hour Completion Time is reasonable, based on operating experience relative to the amount of time required, to reach the required MODE from full power in an orderly manner and without challenging plant systems.

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BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.1.8.1

During normal operation, the SDV vent and drain valves should be in the open position (except when performing SR 3.1.8.2) to allow for drainage of the SDV piping. Verifying that each valve is in the open position ensures that the SDV vent and drain valves will perform their intended functions during normal operation. This SR does not require any testing or valve manipulation. Rather, it involves verification that the valves are in the correct position.

The 31-day Frequency is appropriate because the valves are operated under procedural control and improper valve position (closed) would not affect the isolation function. The SDV Water Level—High scram function will still ensure that water buildup will not get to a point where scram capability is lost.

SR 3.1.8.2

During a scram, the SDV vent and drain valves should close to contain the reactor water discharged to the SDV piping. Cycling each valve through its complete range of motion (closed and open) ensures that the valve will function properly during a scram. The 92-day Frequency is based on operating experience and takes into account the level of redundancy in the system design.

SR 3.1.8.3

SR 3.1.8.3 is an integrated test of the SDV vent and drain valves to demonstrate total system performance. After receipt of a simulated or actual scram signal, the closure of the SDV vent and drain valves is verified. Similarly, after receipt of a simulated or actual scram reset signal, the opening of the SDV vent and drain valves is verified. The 18-month Frequency was developed considering the plant conditions needed to perform the SR and the potential for an unplanned plant transient if the SR 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.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)	The closure time of 60 seconds after a receipt of a scram signal is based on the bounding leakage case evaluated in the accident analysis (Ref. 2).
REFERENCES	1. [Unit Name] FSAR, Section [4], "[Title]." 2. [Unit Name] FSAR, Section [ ], "[Title]."

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

BASES

BACKGROUND

The APLHGR is a measure of the average LINEAR HEAT GENERATION RATE (LHGR) of all the fuel rods in a fuel assembly at any axial location. Limits on the APLHGR are specified to assure that the fuel design limits identified in Reference 1 will not be exceeded during anticipated operational occurrences (AOOs) and that the peak cladding temperature (PCT) during the postulated design basis loss-of-coolant accident (LOCA) will not exceed the limits specified in 10 CFR 50.46.

[For this facility, the instrumentation used for computing the APLHGR is as follows:]

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the fuel design limits are presented in References 1 and 2. The analytical methods and assumptions used in evaluating Design Basis Accidents (DBAs), anticipated operational transients, and normal operation that determine the APLHGR limits are presented in References 1, 2, 3, 4, and 5.

Fuel design evaluations are performed to demonstrate that the 1% limit on the fuel cladding plastic strain and other fuel design limits described in Reference 1 are not exceeded during AOOs for operation with LGHRs up to the operating limit LHGR. APLHGR limits are equivalent to the LHGR limit for each fuel rod divided by the local peaking factor of the fuel assembly. APLHGR limits are developed as a function of exposure and the various operating core flow and power states to ensure adherence to fuel design limits during the limiting AOOs (Refs. 2, 3, 4, and 5). Flow-dependent APLHGR limits are determined using the three-dimensional BWR simulator code (Ref. 6) to analyze slow flow runout transients. The flow-dependent multiplier, MAPLHGR FACTOR, flow dependent component ($MAPFAC_f$), is dependent on the maximum core flow runout capability. The maximum runout flow is dependent on the existing setting of the core flow limiter in the Recirculation Flow Control System.

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

Based on analyses of limiting plant transients (other than core flow increases) over a range of power and flow conditions, power-dependent multipliers, MAPLHGR FACTOR, power dependent component ($MAPFAC_p$), are also generated. Due to the sensitivity of the transient response to initial core flow levels at power levels below those at which turbine stop valve closure and turbine control valve fast closure scram trips are bypassed, both high and low core flow $MAPFAC_p$ limits are provided for operation at power levels between 25% of RATED THERMAL POWER (RTP) and the previously mentioned bypass power level. The exposure-dependent APLHGR limits are reduced by $MAPFAC_p$ and $MAPFAC_f$ at various operating conditions to ensure that all fuel design criteria are met for normal operation and AOOs. A complete discussion of the analysis code is provided in Reference 8.

LOCA analyses are then performed to ensure that the above determined APLHGR limits are adequate to meet the PCT and maximum oxidation limits of 10 CFR 50.46. The analysis is performed using calculational models which are consistent with the requirements of 10 CFR Part 50, Appendix K. A complete discussion of the analysis code is provided in Reference 7. The PCT following a postulated LOCA is a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is not strongly influenced by the rod-to-rod power distribution within an assembly. The APLHGR limits specified are equivalent to the LHGR of the highest powered fuel rod assumed in the LOCA analysis divided by its local peaking factor. A conservative multiplier is applied to the LHGR assumed in the LOCA analysis to account for the uncertainty associated with the measurement of the APLHGR.

For single recirculation loop operation, the MAPFAC multiplier is limited to a maximum of 0.75 (Ref. 2). This is due to the conservative analysis assumption of an earlier departure from nucleate boiling with one recirculation loop available resulting in a more severe cladding heatup during a LOCA.

The limits on the APLHGR satisfy Criterion 2 of the NRC Interim Policy Statement.

(continued)

BASES (continued)

LCO

The APLHGR limits specified in the CORE OPERATING LIMITS REPORT (COLR) are the result of the fuel design, DBA, and transient analyses. For two recirculation loops operating, the limit is determined by multiplying the smaller of the $MAPFAC_p$ and $MAPFAC_f$ factors times the exposure-dependent APLHGR limits. With only one recirculation loop in operation, in conformance with the requirements of LCO 3.4.1, the limit is determined by multiplying the exposure-dependent APLHGR limit times the smaller of either $MAPFAC_p$, $MAPFAC_f$, or 0.75, where 0.75 has been determined by a specific single recirculation loop analysis (Ref. 2).

[For this facility, the OPERABLE instrumentation for computing the APLHGR constitutes the following:]

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

[For this facility, those required support systems which, upon their failure, do not require declaring inoperable the instrumentation for determining the APLHGR and the justifications for not declaring them inoperable are as follows:]

APPLICABILITY

The APLHGR limits are primarily derived from fuel design evaluations, LOCA, and transient analyses that are assumed to occur at high power levels. Design calculations (Ref. 5) and operating experience have shown that as power is reduced, the margin to the required APLHGR limits increases. This trend continues down to the power range of 5% to 15% RTP when entry into MODE 2 occurs. When in MODE 2, the intermediate range monitor (IRM) scram function will provide prompt scram initiation during any significant transient, thereby effectively removing any APLHGR limit compliance concern in MODE 2. Therefore, at THERMAL POWER levels ≤ 25% RTP, the reactor will be operating with substantial margin to the APLHGR limits and this LCO's requirements are not required.

(continued)

BASES (continued)

ACTIONS

A.1

Should any APLHGR exceed the required limits, an assumption regarding an initial condition of the DBA and transient analyses may not be met. Therefore, prompt action should be taken to restore the APLHGRs to within the required limits such that the plant will be operating within analyzed conditions and within design limits of the fuel rods. [For this facility, the APLHGR is restored to within its limits by the following actions:] The 2-hour Completion Time is sufficient time to restore the APLHGR to within its limits and is acceptable based on the low probability of a transient or DBA occurring simultaneously with the APLHGR out of specification.

B.1

If the APLHGR cannot be restored to within its required limits within 2 hours or the APLHGR cannot be determined because the instrumentation for computing the APLHGR is inoperable, the THERMAL POWER must be reduced to < 25% RTP within the following 4 hours. The 4-hour Completion Time provides sufficient time to reduce THERMAL POWER to < 25% RTP in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1

APLHGRs are required to be initially calculated within 12 hours after THERMAL POWER is ≥ 25% RTP and then every 24 hours thereafter. They are compared to the specified limits to assure that the reactor is operating within the assumptions of the safety analysis. The 24-hour Frequency is based on both engineering judgment recognizing the slow changes in power distribution during normal operation and the alarms on the process computer if the APLHGR limit is exceeded. The 12-hour allowance after THERMAL POWER ≥ 25% RTP is achieved is acceptable given the large inherent margin to operating limits at low power levels.

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BASES (continued)

REFERENCES

1. NEDO-24011-P-A, "General Electric Standard Application for Reactor Fuel," (latest approved version).

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

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

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

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

6. NEDO-30130-A, "Steady State Nuclear Methods," May 1985.

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

8. [Unit Namel FSAR, Section [ ], "[.itle]."

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.2 MINIMUM CRITICAL POWER RATIO (MCPR)

BASES

BACKGROUND

The MCPR is the ratio of the fuel assembly power that would result in the onset of boiling transition to the actual fuel assembly power. The MCPR Safety Limit (SL) is set such that 99.9% of the fuel rods will avoid boiling transition if the limit is not violated (refer to the Bases for SL 2.1.2). The operating limit MCPR is established to assure that no fuel damage results during anticipated operational occurrences (AOOs). Although fuel damage would not necessarily occur if a fuel rod actually experienced boiling transition (Ref. 1), the critical power at which boiling transition is calculated to occur has been adopted as a fuel design criterion.

The onset of transition boiling is a phenomenon that is readily detected during the testing of various fuel bundle designs. Based on this experimental data, correlations have been developed that are used to predict critical bundle power (i.e., the bundle power level at the onset of transition boiling) for a given set of plant parameters (e.g., reactor coolant pressure, flow, subcooling, etc.). Since plant operating conditions and bundle power levels are monitored and determined relatively easily, monitoring the MCPR is a convenient way of ensuring that fuel failures due to inadequate cooling do not occur.

[For this facility, the instrumentation used for computing the MCPR is as follows:]

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the AOOs to establish the operating limit MCPR are presented in References 2, 3, 4, and 5. To ensure that the MCPR SL is not exceeded during any transient event which occurs with moderate frequency, limiting transients have been analyzed to determine the largest reduction in critical power ratio (CPR). The types of transients evaluated are loss of flow, increase in pressure and power, positive reactivity insertion and coolant temperature decrease. The

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

limiting transient yields the largest ΔCPR. When the largest ΔCPR is added to the MCPR SL, the required operating limit MCPR is obtained.

The MCPR operating limits derived from the transient analysis are dependent on the operating core flow and power state ($MCPR_f$ and $MCPR_p$, respectively) to ensure adherence to fuel design limits during the worst transient which occurs with moderate frequency (Refs. 3, 4, and 5). Flow-dependent MCPR limits are determined by steady-state thermal hydraulic methods with key physics response inputs benchmarked using the three-dimensional BWR simulator code (Ref. 6) to analyze slow flow runout transients. The operating limit is dependent on the maximum core flow limiter setting in the Recirculation Flow Control System.

Power-dependent MCPR limits ($MCPR_p$) are determined mainly by the one-dimensional transient code (Ref. 7). Due to the sensitivity of the transient response to initial core flow levels at power levels below those at which the turbine stop valve closure and turbine control valve fast closure scram trips are bypassed, a high and low flow operating limit $MCPR_p$ is provided for operating between 25% of RATED THERMAL POWER (RTP) and the previously mentioned bypass power level.

The limits on the MCPR satisfy Criterion 2 of the NRC Interim Policy Statement.

LCO

The MCPR operating limits specified in the CORE OPERATING LIMITS REPORT are the result of the design basis accident and transient analysis. The operating limit MCPR is determined by the larger of the $MCPR_f$ and $MCPR_p$ limits.

[For this facility, the OPERABLE instrumentation for computing MCPR consitutes the following:]

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

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BASES (continued)

LCO (continued)

[For this facility, those required support systems which, upon their failure, do not require declaring instrumentation inoperable for determining the MCPR and the justifications for not declaring them inoperable are as follows:]

APPLICABILITY

The MCPR operating limits are primarily derived from transient analyses that are assumed to occur at high power levels. Below 25% RTP, the reactor will be operating at a minimum recirculation pump speed and the moderator void ratio will be very small. Surveillance of thermal limits below 25% RTP is unnecessary due to the large inherent margin which assures that the SL MCPR will not be exceeded even if a limiting transient should occur. Statistical analyses indicate that the nominal value of the initial MCPR expected at 25% RTP is > 3.5. Studies of the variation of limiting transient behavior have been performed over the range of power and flow conditions. These studies encompass the range of key actual plant parameter values important to typically limiting transients. The results of these studies demonstrate that a margin is expected between performance and the MCPR requirements, and that margins increase as power is reduced to 25% RTP. This trend is expected to continue to the 5% to 15% power range when entry into MODE 2 occurs. When in MODE 2, the intermediate range monitor (IRM) provides rapid scram initiation for any significant power increase transient which effectively eliminates any MCPR compliance concern. Therefore, at THERMAL POWER levels < 25% RTP, the reactor will be operating with substantial margin to the MCPR limits and this LCO's requirements are not needed.

ACTIONS

A.1

Should any MCPR be outside the required limits, an assumption regarding an initial condition of the design basis transient analyses may not be met. Therefore, prompt action should be taken to restore the MCPRs to within the required limits such that the plant will be operating within analyzed conditions. [For this facility, the MCPR is restored to within its limits by the following actions:]

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BASES (continued)

ACTIONS (continued)

The 2-hour Completion Time is normally sufficient to restore the MCPR to within its limits and is acceptable based on the low probability of a transient or Design Basis Accident (DBA) occurring simultaneously with the MCPR out of specification.

B.1

If the MCPR cannot be restored to within its required limits within 2 hours or the MCPR cannot be determined because the instrumentation for computing the MCPR is inoperable, the THERMAL POWER must be reduced to < 25% RTP within the following 4 hours. The 4-hour Completion Time provides sufficient time to reduce THERMAL POWER to less than 25% RTP in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1

The MCPR is required to be initially calculated within 12 hours after THERMAL POWER is ≥ 25% RTP and then every 24 hours thereafter. It is compared to the specified limits to assure that the reactor is operating within the assumptions of the safety analysis. The 24-hour Frequency is based on both engineering judgment recognizing the slow changes in power distribution during normal operation and the alarms on the process computer if the MCPR limit is exceeded. The 12-hour allowance after THERMAL POWER ≥ 25% RTP is achieved is acceptable given the large inherent margin to operating limits at low power levels.

SR 3.2.2.2

Since the transient analysis takes credit for conservatism in the scram speed performance, it must be demonstrated that the specific scram speed distribution is consistent with that used in the transient analysis. SR 3.2.2.2 determines the value of τ, which is a measure of the actual scram speed distribution compared to the assumed distribution. The MCPR operating limit is then determined based on an interpolation between the applicable limits for Option A (scram times of LCO 3.1.4) and Option B (realistic scram times) analyses. The parameter τ must be determined once within 72 hours

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

after each set of scram time tests required by SR 3.1.4.1 and SR 3.1.4.2 since the effective scram speed distribution may change during the cycle. The 72-hour Completion Time is acceptable due to the relatively minor changes in τ expected during the fuel cycle.

The value of τ is defined in Reference 8 as follows:

τ = 1.0, before the first scram time measurements of SR 3.1.4.1 for the cycle, otherwise

$$\tau = 0 \text{ or } \frac{\tau_{ave} - \tau_B}{\tau_A - \tau_B}, \text{ whichever is greater}$$

where:

τ_A = 1.096 sec

$$\tau_B = \mu + (1.65) * \left[\frac{N_1}{\sum_{i=1}^{n} N_i}\right]^{1/2} * \sigma$$

μ = 0.822 sec (mean scram time used in the transient analysis)

σ = 0.018 sec (standard deviation of μ)

$$\tau_{ave} = \frac{\sum_{i=1}^{n} N_i \tau_i}{\sum_{i=1}^{n} N_i}$$

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

where:

n = number of surveillance tests performed to date in the cycle

N_i = number of OPERABLE control rods measured in the i_{th} surveillance test

τ_i = average scram time to notch 36 of all control rods in the i_{th} surveillance test (seconds)

N_i = total number of OPERABLE control rods measured in SR 3.1.4.1.

REFERENCES

1. NUREG-0562, "Fuel Rod Failure as a Consequence of Departure From Nucleate Boiling or Dryout," June 1979.

2. NEDO-24011-P-A, "General Electric Standard Application for Reactor Fuel," (latest approved version).

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

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

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

6. NEDO-30130-A, "Steady State Nuclear Methods," May 1985.

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

8. Letter, R. L. Tedesco (NRC) to G. G. Sherwood, (GE), "Acceptance for Referencing General Electric Licensing Topical Report NEDO-24154," February 4, 1981, Item 2 of the Supplemental Safety Evaluation, page cxvi.b.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3 LINEAR HEAT GENERATION RATE (LHGR) (Applicable to Non-GE Fuel Only)

BASES

BACKGROUND

The LHGR is a measure of the heat generation rate of a fuel rod in a fuel assembly at any axial location. Limits on the LHGR are specified to assure that fuel design limits will not be exceeded anywhere in the core during normal operation including anticipated operational occurrences (AOOs). Exceeding the LHGR limit could potentially result in fuel damage and subsequent release of radioactive materials. Fuel design limits are specified to assure that fuel system damage, fuel rod failure or inability to cool the fuel will not occur during the anticipated operating conditions identified in Reference 1.

[For this facility, the instrumentation used for computing LHGR is as follows:]

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the fuel system design are presented in References 1 and 3. The fuel assembly is designed to ensure (in conjunction with the core nuclear and thermal hydraulic design, plant equipment, instrumentation and protection system) that fuel damage will not result in the release of radioactive materials in excess of the guidelines of 10 CFR Parts 20, 50, and 100. The mechanisms that could cause fuel damage during operational transients and that are considered in fuel evaluations are:

a. Rupture of the fuel rod cladding caused by strain from the relative expansion of the UO_2 pellet; and

b. Severe overheating of the fuel rod cladding caused by inadequate cooling.

A value of 1% plastic strain of the Zircaloy cladding has been defined as the limit below which fuel damage caused by overstraining of the fuel cladding is not expected to occur (Ref. 2). The Safety Limit MINIMUM CRITICAL POWER RATIO (MCPR) ensures that fuel damage caused by severe overheating of the fuel rod cladding is avoided.

(continued)

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

Fuel design evaluations have been performed and demonstrate that the 1% fuel cladding plastic strain design limit is not exceeded during continuous operation with LHGRs up to the operating limit specified in the CORE OPERATING LIMITS REPORT (COLR). The analysis also includes allowances for short-term transient operation above the operating limit to account for AOOs, plus an allowance for densification power spiking.

The limits on the LHGR satisfy Criterion 2 of the NRC Interim Policy Statement.

LCO

The LHGR is a basic assumption in the fuel design analysis. The fuel has been designed to operate at rated core power with sufficient design margin to the LHGR calculated to cause a 1% fuel cladding plastic strain. The operating limit to accomplish this objective is specified in the COLR.

[For this facility, the OPERABLE instrumentation for computing the LHGR constitutes the following:]

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

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

APPLICABILITY

The LHGR limit is derived from fuel design analysis that is limiting at high power level conditions. At core thermal power levels < 25% RATED THERMAL POWER (RTP), the reactor will be operating with a substantial margin to the LHGR limits and, therefore, the specification is only required when operating at or above 25% RTP.

(continued)

BASES (continued)

ACTIONS

A.1

Should any LHGR exceed its required limits, an assumption regarding an initial condition of the fuel design analysis will not be met. Therefore, prompt action should be taken to restore the LHGR to within its required limits such that the plant will be operating within analyzed conditions. [For this facility, the LHGR is restored within its limits by the following actions:] The 2-hour Completion Time is normally sufficient to restore the LHGR to within its limits and is acceptable based on the low probability of a transient or Design Basis Accident (DBA) occurring simultaneously with the LHGR out of specification.

B.1

If the LHGR cannot be restored to within its required limits within 2 hours or the LHGR cannot be determined because the instrumentation for computing the LHGR is inoperable, the THERMAL POWER must be reduced to < 25% RTP within the following 4 hours. The 4-hour Completion Time provides sufficient time to reduce THERMAL POWER to < 25% RTP in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.3.1

The LHGR is required to be initially calculated within 12 hours after THERMAL POWER is ≥ 25% RTP and then every 24 hours thereafter. It is compared to the specified limits to assure that the reactor is operating within the assumptions of the safety analysis. The 24-hour Frequency is based on both engineering judgment recognizing the slow changes in power distribution during normal operation and the alarms on the process computer if the LHGR limit is exceeded. The 12-hour allowance after THERMAL POWER ≥ 25% RTP is achieved is acceptable given the large inherent margin to operating limits at lower power levels.

(continued)

BASES (continued)

REFERENCES

1. [Unit Name] FSAR, Section [ ], "[Non-GE Fuel Analysis]."
2. NUREG-0800, Standard Review Plan 4.2, "Fuel System Design," Section II.A.2(g), Revision 2, July 1981.
3. [Unit Name] FSAR, Section [ ], "[Title]."

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 Average Power Range Monitor (APRM) Setpoints

BASES

BACKGROUND

The OPERABILITY of the APRM and its setpoints is an assumption in all safety analyses which assume rod insertion upon reactor trip or rod block. Applicable GDCs are 10, Reactor Design; 13, Instrumentation and Control; 20, Protection System Functions; and 29, Protection against Anticipated Operation Occurrences (Ref. 1). This LCO is provided to required the APRM gain or APRM flow biased scram and rod block trip setpoints to be adjusted when operating under conditions of excessive power peaking so as to maintain acceptable margin to the fuel-cladding integrity Safety Limit (SL) and the fuel cladding 1% plastic strain limit.

The condition of excessive power peaking is determined by the ratio of the actual power peaking to the limiting power peaking at RATED THERMAL POWER (RTP) and is equal to the ratio of the core limiting MAXIMUM FRACTION OF LIMITING POWER DENSITY (MFLPD) to the fraction of RATED THERMAL POWER (FRTP). Excessive power peaking exists when this ratio,

$$\frac{\text{MFLPD}}{\text{FRTP}} > 1$$

indicating that MFPLD is not decreasing proportionately to the overall power reduction or conversely, that power peaking is increasing. To maintain margins similar to those at RTP conditions, the excessive power peaking is compensated for by either a gain adjustment on the APRMs, or adjustment of the APRM setpoints. Either of these adjustments effectively has the same result as maintaining MFLPD less than or equal to FRTP and thus maintains RTP margins for AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHRG) and MINIMUM CRITICAL POWER RATIO (MCPR).

The normally selected APRM setpoints position the scram trip and rod block trip above the upper bound of the normal power/flow operating region which has been considered in the design of the fuel rods. The setpoints are flow biased with a slope that approximates the upper flow control line, such that an approximately constant margin is maintained between

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(continued)

BASES (continued)

BACKGROUND (continued)

the flow biased trip level and the upper operating boundary for core flows in excess of about 45% of rated core flow. In the range of infrequent operations below 45% of rated core flow, the margin to scram or rod blocks is reduced because of the non-linear core flow versus drive flow relationship. The normally selected APRM setpoints are supported by the analyses presented in References 2 and 3 that concentrate on events initiated from rated conditions. Design experience has shown that minimum deviations occur within expected margins to operating limits (APLHGR and MCPR), at rated conditions for normal power distributions. However, at conditions other than rated, control rod patterns can be established that significantly reduce the margin to thermal limits. Therefore, the flow biased APRM scram and rod block setpoints may be reduced during operation when the combination of THERMAL POWER and MFLPD indicates an excessive power peaking distribution.

[For this facility, the instrumentation used for computing the MFLPD is as follows:]

For completeness, it is also noted that the APRM neutron flux signal is also adjusted to more closely follow the fuel cladding heat flux during power transients. The APRM neutron flux signal is a measure of the core thermal power during steady-state operation. During power transients, the APRM signal leads the actual core thermal power response because of the fuel thermal time constant. Therefore, on power increase transients, the APRM signal provides a conservatively high measure of core thermal power. By passing the APRM signal through an electronic filter with a time constant less than, but approximately equal to, that of the fuel thermal time constant, an APRM transient response that more closely follows actual fuel cladding heat flux is obtained, while maintaining a conservative margin. The delayed response of the filtered APRM signal allows the flow biased APRM scram and rod block trip levels to be positioned closer to the upper bound of the normal power and flow range, without unnecessarily causing reactor scrams or rod blocks during short duration neutron flux spikes. These spikes can be caused by insignificant transients such as main steam line valve surveillance testing or momentary flow increases of only several percent.

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BASES (continued)

APPLICABLE SAFETY ANALYSES

The acceptance criteria for the APRM gain or setpoint adjustments are that acceptable margins (to APLHGR and MCPR) be maintained to the fuel cladding integrity Safety Limit (SL) and the fuel cladding 1% plastic strain limit.

FSAR safety analyses (Refs. 2 and 3) concentrate on the rated power condition where the minimum expected margin to the operating limits (APLHGR and MCPR) exist. LCO 3.2.1 and LCO 3.2.2 limit the initial margins to these operating limits at rated conditions so that specified acceptable fuel design limits are met during transients initiated from rated conditions. At initial power levels less than rated, the margin degradation of either the APLHGR or the MCPR during a transient can be greater than at the rated condition event. This greater margin degradation during the transient is primarily offset by the larger initial margin to limits at the lower than rated power levels. However, power distributions can be hypothesized that would result in reduced margins to the pre-transient operating limit. When combined with the increased severity of certain transients at conditions other than rated, the SLs could be approached. At substantially reduced power levels, very highly peaked power distributions could be obtained which could reduce thermal margins to the minimum levels required for transient events. To prevent or mitigate such situations, either the APRM gain is adjusted upward by the ratio of the core limiting MFLPD to the fraction of RTP or the flow biased APRM scram and rod block level are required to be reduced by the ratio of the fraction of RATED THERMAL POWER (FRTP) to the core limiting MFLPD. Either of these adjustments effectively counters the increased severity of some events at conditions other than rated by proportionally increasing the APRM gain or proportionally lowering the flow biased APRM scram and rod block setpoints dependent on the increased peaking that may be encountered.

The APRM gain and its setpoints satisfy Criteria 2 and 3 of the NRC Interim Policy Statement.

LCO

When operating under conditions of excessive power peaking, either the APRM gain must be adjusted upward or the flow biased neutron flux upscale scram trip and rod block

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(continued)

BASES (continued)

LCO (continued)

setpoints must be reduced to account for the reduction in margin to the fuel cladding integrity SL and the fuel cladding 1% plastic strain limit. This is accomplished by multiplying the APRM gain or setpoints by a factor that is representative of the reduction in margin to SLs. This factor is $\frac{1}{T}$ for the gain adjustments or T for the setpoint adjustments where T is defined as the ratio of the FRTP divided by the core limiting value of MFLPD. MFLPD is the ratio of the limiting LHGR to the LHGR limit for the specific bundle type. For design power distributions, at rated power, MFLPD is equal to 1.0 and therefore, T = 1.0. As power is reduced, if the design power distribution is maintained. MFLPD will be reduced in proportion to the reduction in power and T will continue to be equal to 1.0. However, if power peaking increases above the design value, the MFLPD will not be reduced in proportion to the reduction in power and T will be less than 1.0. Under these conditions, the APRM gain would be adjusted upward or the APRM flow biased scram and rod block setpoints would be reduced accordingly. When the reactor is operating with peaking less than the design value, T will always be greater than 1.0 and it is not necessary to modify the APRM flow biased scram or rod block setpoints. Making the APRM gain or setpoint adjustments is equivalent to maintaining MFLPD less than or equal to FRTP as stated in the LCO.

[For this facility, the OPERABLE instrumentation for computing MFLPD constitutes the following:]

[For this facility, the following support systems are required to be OPERABLE to ensure instrumentation OPERABILITY for determining MFLPD:]

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

APPLICABILITY

The APRM gain adjustment or APRM flow biased scram and rod block and associated setdowns is provided to ensure that the fuel cladding integrity SL and the fuel cladding 1% plastic strain limit are not violated during design basis

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BASES (continued)

APPLICABILITY (continued)

transients. As discussed in the Bases for LCO 3.2.1 and LCO 3.2.2, sufficient margin to these limits exists below 25% of RTP and, therefore, these requirements are only necessary when operating at or above this power level.

ACTIONS

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

If the APRM gain or setpoints are not within limits the MFLPD may exceed its limit, and it is possible that the margin to the fuel cladding integrity SL and the fuel cladding 1% plastic strain limit may be reduced. Therefore, prompt action should be taken to restore the MFLPD to within its required limit such that the plant will be operating within the assumed margin of the safety analyses. A Completion Time of 6 hours to restore the MFLPD is acceptable because [ ]. [For this facility, the MFLPD is restored to within its limits by the following actions:]

An alternate action is to adjust the APRM setpoints to the relationship specified in the CORE OPERATING LIMITS REPORT (COLR). Another alternate action is to adjust the APRM gain such that the APRM readings are ≥ MFLPD, provided that the adjusted APRM reading does not exceed 100% RTP. (Posting a notice of adjustment on the reactor control panel may be appropriate.) The 6-hour Completion Time for both of these alternatives is normally sufficient to restore either the APRM gain or setpoints to within limits and is acceptable based on the low probability of a transient or Design Basis Accident (DBA) occurring simultaneously with the APRM gain or setpoints out of specification.

B.1

If the APRM gain or setpoints cannot be restored to within the required limits within 6 hours or the MFLPD cannot be restored or determined because the instrumentation for computing the MFLPD is inoperable, it is required to reduce THERMAL POWER to < 25% RTP within the following 4 hours. Reducing THERMAL POWER to < 25% RTP places the reactor outside the applicability of the LCO. The 4-hour Completion Time provides sufficient time to reduce THERMAL POWER to < 25% RTP in an orderly manner and without challenging plant systems.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1

The APLHGRs are required to be calculated every 24 hours and compared to their specified limits to assure that the reactor is operating within the assumptions of the safety analysis. When these APLHGRs are determined, The T factor should also be determined for the operating conditions. T is the ratio of FRTP to MFLPD. MFLPD is the ratio of the limiting LGHR to the LHGR limit for the specific bundle type. As long as T is ≥ 1.0, no changes to the APRM gain or flow biased neutron flux upscale scram or rod block trip setpoints are required. This SR is only required to determine the appropriate gain or setpoint and is not intended to be a CHANNEL FUNCTIONAL TEST for the APRM gain or flow biased neutron flux scram circuitry. The 24-hour Frequency is chosen to coincide with the determination of other thermal limits, specifically the APLHGR (LCO 3.2.1). The 24-hour Frequency is based on both engineering judgment recognizing the slow changes in power distribution during normal operation and the alarms on the process computer if the APRM limit is exceeded. The 12-hour allowance after THERMAL POWER ≥ 25% RTP is achieved is acceptable given the large inherent margin to operating limits at low power levels.

REFERENCES

1. Title 10, Code of Federal Regulations, Appendix A, General Design Criteria, 10, 13, 20, and 23.
2. [Unit Name] FSAR, Section [ ], "[Title]."
3. [Unit Name] FSAR, Section [ ], "[Title]."

B 3.3 INSTRUMENTATION

B 3.3.1.1 Reactor Protection System (RPS) Instrumentation

BASES

BACKGROUND

The RPS initiates a reactor shutdown, based on the values of selected plant parameters, to protect against violating the core fuel design limits and Reactor Coolant System (RCS) pressure boundary during anticipated operational occurrences (AOOs), and to assist the engineered safety feature (ESF) systems in mitigating accidents.

The protection and monitoring functions of the RPS 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 LCOs on other reactor system parameters and equipment performance.

The LSSS, defined in this specification as the ALLOWABLE VALUE, in conjunction with the LCOs, establish the threshold for protective system action to prevent exceeding acceptable limits during design basis accidents (DBAs).

Accidents are events that are analyzed (Ref. 1) even though they are not expected to occur during the plant life. The acceptable limit during accidents is that the offsite dose shall be maintained within an acceptable fraction of 10 CFR 100 limits. Different accident categories are allowed a different fraction of these limits, based on their probability of occurrence. Meeting the acceptable dose limit for an accident category is considered having acceptable consequences for that event.

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

1. The MINIMUM CRITICAL POWER RATIO (MCPR) Safety Limit (SL) value shall be met for high-pressure, high-flow conditions;

2. The THERMAL POWER SL value shall met for low-flow or low-pressure conditions; and

3. The RCS pressure SL limit of [1325] psia shall not be exceeded.

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BACKGROUND (continued)

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.

RPS instrumentation is designed as shown on Figure B 3.3.1.1-1 to include these modules:

1. Field transmitters or process sensors;

2. Signal processing and bistable modules; and

3. Trip logic.

The role of each in the RPS is discussed below.

Field Transmitters or Process Sensors

Field transmitters or process sensors provide a measurable electronic signal based on the physical characteristics of the process being measured.

For most AOOs and DBAs a wide range of dependent and independent parameters are monitored. The input parameters to the scram logic are electrical signals from instrumentation that monitor reactor vessel water level, reactor vessel pressure, neutron flux, main steam line isolation valve (MSIV) position, turbine control valve fast closure actuation, turbine stop valve position, drywell pressure, and scram discharge volume water level, as well as reactor mode switch in shutdown position and manual scram signals.

Typically four measurement channels with physical separation are provided for each parameter. Typically, these are organized into two trip systems which are physically and electrically separated. Exceptions are the reactor mode switch in shutdown position, manual scram signals used in the direct generation of trip signals, and neutron flux intermediate range monitor (IRM) and average power range monitor (APRM) channel trips. Four measurement channels are necessary to meet the redundancy and testability of GDC 21 in Appendix A to 10 CFR 50 (Ref. 2) and to implement the one-out-of-two-taken-twice logic arrangement.

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BASES (continued)

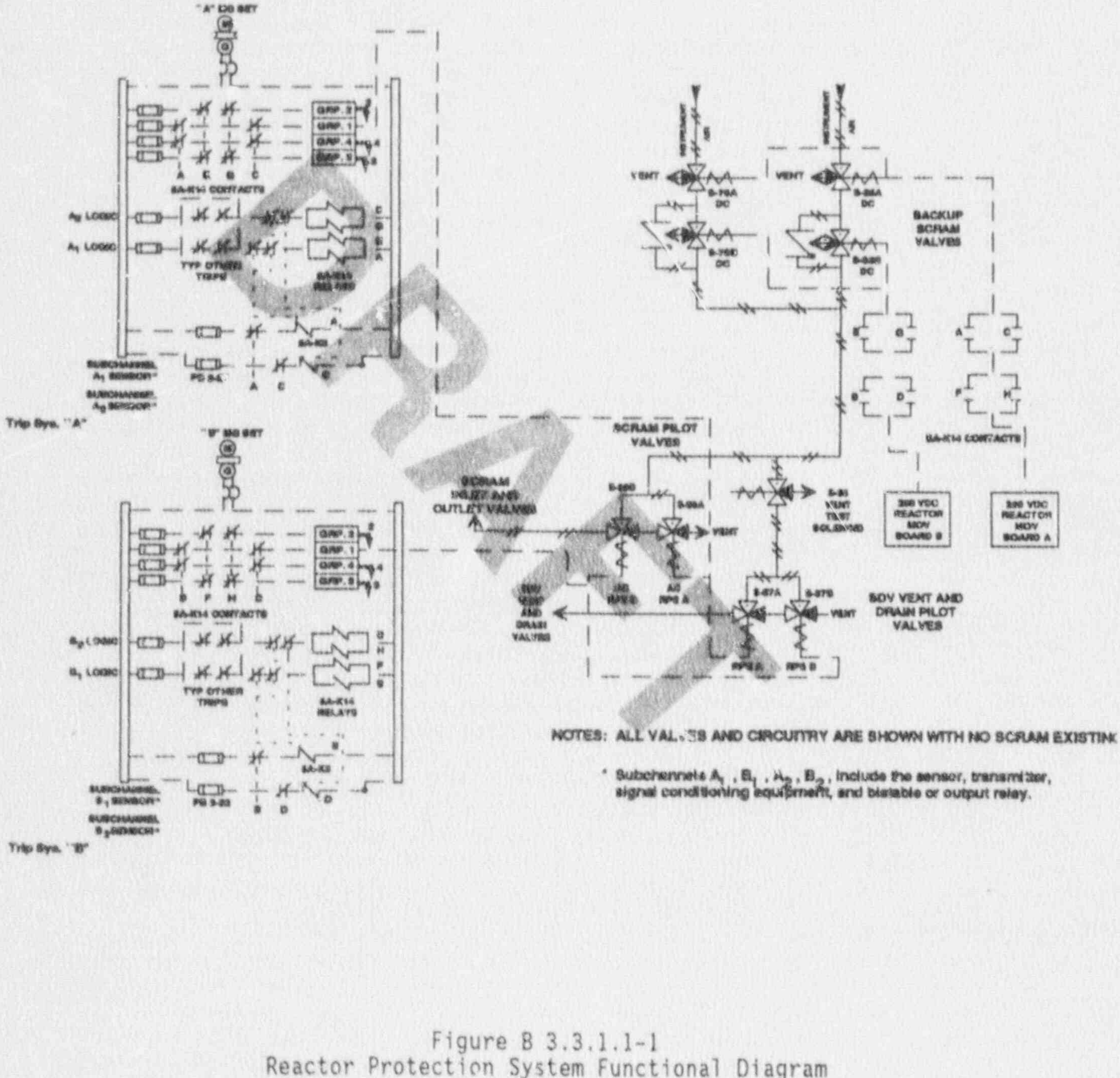

Figure B 3.3.1.1-1
Reactor Protection System Functional Diagram
Typical Figure

(continued)

(continued)

BASES (continued)

BACKGROUND (continued)

Signal Processing and Bistable Modules

Each process parameter measurement channel includes electronic equipment which provides signal conditioning, comparable output signals for main control board instruments, comparison of measured input signals with setpoints established by safety analyses, and output to the trip logic channels. This output to the trip logic channels is taken from a bistable device which can be mechanical switches that are part of the process sensors or electronic comparators that receive input from the process transmitters or sensors. In either case, the bistable output contacts are considered to be part of the trip logic channel.

Trip Logic, Trip Setpoints, and ALLOWABLE VALUES

Tip setpoints are those predetermined values of output voltage or current against which the output voltage or current related to the present value of the process parameter is compared. If the present measured output value of the process parameter exceeds the setpoint, the associated bistable changes state. The trip setpoints are the nominal values at which the bistables are set. They are derived from the limiting values of the process parameters obtained from the accident analyses (analytical limits) through a process of correction for uncertainties and errors set forth in the plant-specific setpoint methodology (Ref. 3). The analytical limits, corrected for analytic and process uncertainties, become the ALLOWABLE VALUES listed in Table 3.3.1.1-1, following this specification, which when further corrected by the methodology of Reference 3 become the calculated trip setpoint values.

The setpoints derived in this manner provide adequate protection because sensor and processing time delays are accounted for as well as calibration tolerances, instrumentation uncertainties, instrument drift and severe environment errors, for channels that must function in harsh environments as defined by 10 CFR 50.49 (Ref. 4).

The actual nominal trip setpoint entered into the bistable is usually still more conservative than that calculated by the plant-specific setpoint methodology. If the setpoint measured for the bistable by the surveillance test does not exceed the documented surveillance test acceptance criteria, the bistable is considered OPERABLE.

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BASES (continued)

BACKGROUND (continued)

Setpoints consistent with the ALLOWABLE VALUE will ensure that SLs are not violated during AOOs, and the consequences of DBAs will be acceptable, providing the plant is being operated within the LCOs at the onset of the AOO or DBA, and the equipment functions as designed.

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

The ALLOWABLE VALUES listed in Table 3.3.1.1-1 are 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.

Trip Logic

The RPS is composed of two independent, separately powered trip systems, trip system A and trip system B, each of which contains two automatic trip logic channels and one manual trip logic channel (see Figure 3.3.1.1-2). Trip system A combines the bistable outputs of automatic trip logic channels A_1 and A_2 in a one-out-of-two logic which will de-energize the solenoid on the A scram pilot valve in each of 177 hydraulic control units (HCUs) when any one of the bistables in either trip logic channel A_1 or A_2 changes to the tripped state. Placing the reactor mode switch in the shutdown position or depressing the manual scram push button in trip logic channel A_3 will also de-energize the solenoid on the A scram pilot valve. Trip of either automatic logic channel or the manual channel produces a half scram condition in the RPS. For scram to occur, similar actions must take place in trip system B which combines the bistable outputs of automatic trip logic channels B_1 and B_2 in a one-out-of-two logic exactly like that described for trip system A. Manual logic channel B_3 is exactly like trip logic channel A_3 described for trip system A. Thus, both trip system A and trip system B

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BASES (continued)

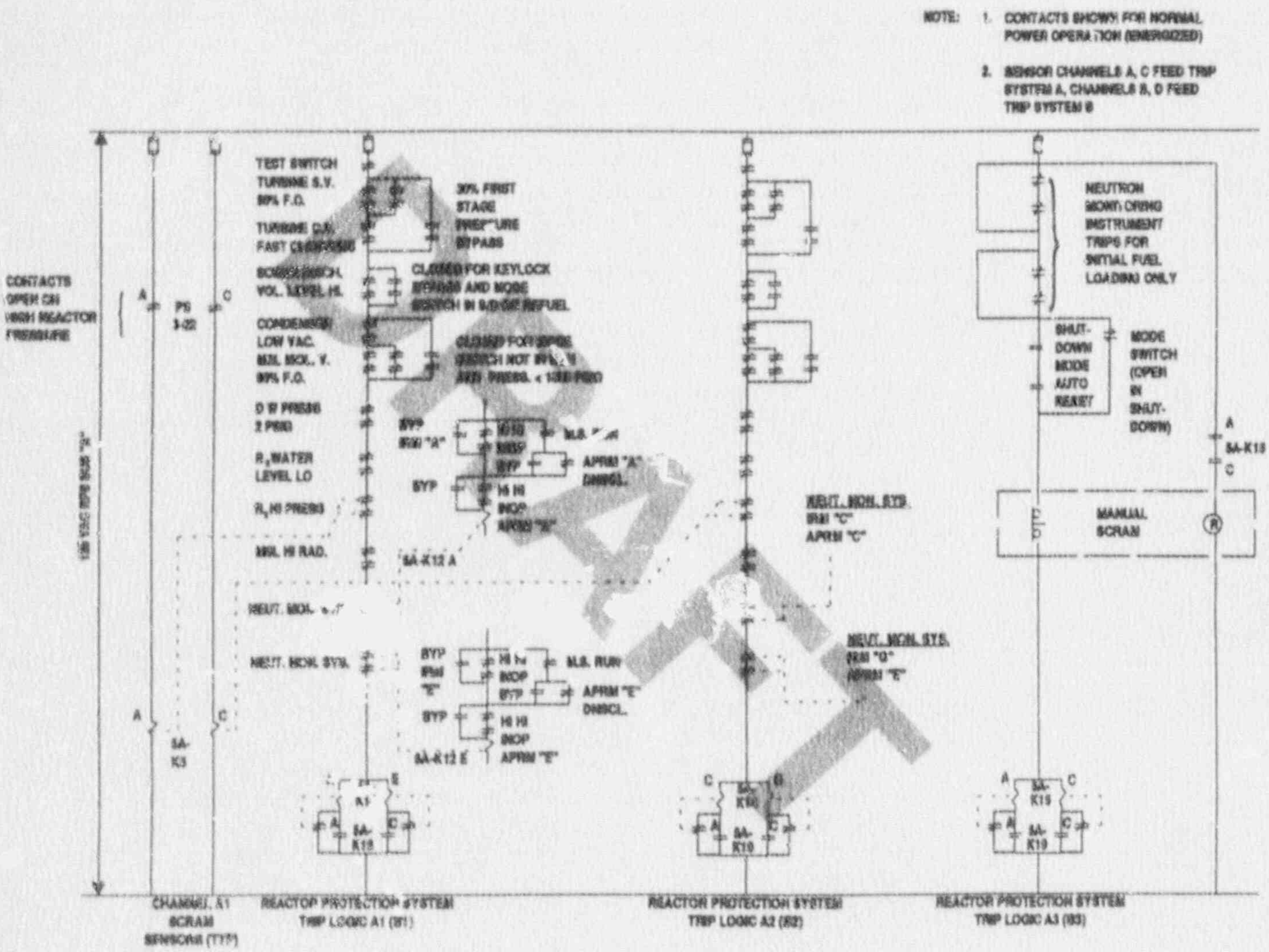

Figure B 3.3.1.1-2
RPS Trip System "A" Logic Channels A1, A2, A3
(Typical of Channels B1, B2, B3)
Typical Figure

(continued)

BASES (continued)

BACKGROUND (continued)

must trip for scram to occur. This overall logic is referred to as "one-out-of-two taken twice" logic.

For this discussion and the technical specifications, a trip logic channel is the electrical combination of all the outputs of RPS sensor channel bistables which have the same channel designator, e.g., Channel A. In a four-channel instrument system, the A sensor channels for each monitored parameter supply input to trip logic channel A_1; similarly sensor channels B, C, and D supply input to trip logic channels B_1, A_2, and B_2 respectively.

Each trip logic channel A_1, A_2, B_1, and B_2 includes operating bypasses on certain RPS trips. These bypasses are enabled automatically or manually when plant conditions do not warrant the specific trip protection. All operating bypasses are automatically removed when the enabling bypass conditions are no longer satisfied.

A trip of either trip logic channel in a trip system trips that trip system and de-energizes the solenoid of the associated scram pilot valve in each HCU. The HCU for each control rod drive (CRD) has two solenoid-operated scram pilot valves, A and B, that control the air supply to the pneumatically operated scram inlet and outlet valves of the HCU. If the solenoid of either scram pilot valves is energized, air pressure holds the scram valves closed; therefore, the solenoids of both scram pilot valves must be de-energized to scram the control rod. The scram valves control the supply and discharge of CRD water to each rod drive during a scram. Thus, a trip of trip system A in conjunction with a trip of trip system B de-energizes the solenoids of both scram pilot valves, air bleeds off, the scram inlet and outlet valves open, and the control rods scram. The backup scram valves are also energized by the RPS to depressurize the scram air header to provide additional assurance that the rods scram. In addition, the RPS scram signal causes the scram discharge volume (SDV) vent and drain valves to close to isolate the SDV.

No single failure will prevent protection system actuation, and protection channels do not interact with control channels. This arrangement meets the requirements of GDC 21 in 10 CFR 50, Appendix A (Ref. 2), and IEEE-279 (Ref. 5).

Note that in LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," the ALLOWABLE VALUE of Table 3.3.1.1-1 are the LSSS. This ALLOWABLE VALUE is established to prevent violation of the SLs during normal plant operation and AOOs.

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BASES (continued)

APPLICABLE SAFETY ANALYSES

The RPS functions to maintain the SLs during all AOOs and mitigates the consequences of DBAs in all modes except cold shutdown. Each of the analyzed accidents and transients can be detected by one or more RPS functions. Typically, this is achieved by satisfying the requirements of IEEE-279 (Ref. 5) with one-out-of-two taken twice logic. These RPS reactor trip functions are as follows:

1. Intermediate Range Monitors

1.a. Neutron Flux--High

The IRMs monitor neutron flux levels from the upper range of the source range monitor (SRM) to the lower range of the APRM subsystems. The IRMs generate trip signals to prevent fuel damage caused by neutron flux excursions from operational transients in the intermediate power range. In this power range, the most significant source of reactivity change is due to control rod withdrawal. For this event, the IRM is diverse to the rod worth minimizer (RWM), which monitors and controls the movement of the control rods at low power. The IRM mitigates neutron flux excursions by initiating a scram. Generic analyses of the consequences of control rod withdrawal events during startup that are mitigated only by the IRM have been performed to evaluate the capability of the IRM system to mitigate these events. This analysis, which assumes that one IRM channel in each trip system is bypassed, demonstrates that the IRMs provide protection against local control rod withdrawal errors and result in peak fuel energy depositions below the 170 cal/gm fuel failure threshold criterion.

For this facility, the IRM Neutron Flux--High trip provides protection during control rod withdrawal and cold water injection (AOO) events.

1.b. Inop

The IRM Inop function generates a reactor trip to prevent fuel damage resulting from significant reactivity increases by ensuring that required

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

IRM channels are OPERABLE. This function ensures that the nuclear instrument channel fails in the tripped condition upon loss of detector voltage and it must be OPERABLE to fulfill the requirements of GDC 23 in 10 CFR 50 (Ref. 2). GDC 23 requires that the RPS fail into a safe state on loss of power. For this facility, IRM Inop trip provides protection for loss of IRM neutron monitor channel (AOO) events.

2. Average Power Range Monitors

2.a. Neutron Flux--High, Setdown

The APRM Neutron Flux--High, Setdown function generates a reactor trip to prevent fuel damage during significant reactivity increases with power < 25% of RATED THERMAL POWER (RTP). It protects fuel from damage by ensuring that before the reactor mode switch is placed in the run position, reactor power does not exceed the 25% RTP SL when operating at low reactor pressure and low core flow.

For most operation at low power levels the APRM Neutron Flux--High, Setdown function will provide a secondary trip to the IRM Neutron Flux--High function because of the relative setpoints. With IRMs on Range 9 or 10; however, it is possible that APRM Neutron--High, Setdown will provide the primary trip signal for a core-wide increase in power. For this facility, APRM Neutron Flux--High, Setdown protects against fuel damage for local power transients in MODE 2 (AOO) events.

2.b. Flow Biased Simulated Thermal Power--High

The APRM Flow Biased Simulated Thermal Power--High function provides protection against transients where thermal power increases slowly such as the loss of feedwater heating event to protect the fuel integrity by ensuring LCO 3.2.2, "Minimum Critical Power Ratio," is not exceeded. During such events, the thermal power increase does not significantly lag the neutron flux response and, because of a lower trip setpoint,

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APPLICABLE SAFETY ANALYSES (continued)

will initiate a scram before the high neutron flux scram. For rapid neutron flux increase events, the thermal power lags the neutron flux and the Fixed APRM Neutron Flux--High function will provide a scram signal before the APRM Flow Biased Simulated Thermal Power function setpoint is exceeded.

The thermal power time constant of 5 to 7 seconds is based on the fuel heat transfer dynamics and provides a signal proportional to the thermal power. For this facility, the APRM Flow Biased Thermal Power--High reactor trip protects against loss of feedwater heating events.

2.c. Fixed Neutron Flux--High

The APRM Fixed Neutron Flux--High function provides the primary indication of neutron flux levels and neutron flux changes in the core and generates a trip signal to prevent fuel damage or excessive RCS pressure. For the overpressurization protection analysis described in Reference 1, the APRM Fixed Neutron Flux--High function is to terminate the MSIV closure event and together with the safety/relief valves (S/RVs), limits the peak reactor vessel pressure to less than the American Society of Mechanical Engineers (ASME) code limits. The control rod drop accident (CRDA) analysis (Ref. 1) assumes the APRM Fixed Neutron Flux--High function is OPERABLE to terminate the CRDA.

2.d. Downscale

The APRM Downscale function generates a reactor trip signal in concert with the IRM Neutron Flux--High trip to prevent fuel damage from significant reactivity increases by ensuring sufficient required neutron monitoring channels are OPERABLE when the reactor mode switch is placed in run prior to the APRMs coming on-scale. [For this facility, the APRM Downscale Function provides protection for AOO events as follows:]

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

2.e. Inop

The Average Power Range Monitor Inop function generates a reactor trip signal to prevent fuel damage from significant reactivity increases by ensuring that the required minimum number of local power range monitor (LPRM) neutron channels are OPERABLE and in operation. [For this facility, the APRM Inop provides protection for AOO events as follows:]

3. Reactor Vessel Steam Dome Pressure--High

Reactor Steam Dome Pressure--High generates reactor trip signals to prevent fuel damage by ensuring that the MCPR SLs are not exceeded and maintains the integrity of the RCS pressure boundary by ensuring that the high RCS pressure SL is not exceeded. For this facility, the Reactor Vessel Steam Dome Pressure--High function in conjunction with the S/RVs provides protection against overpressurization of the RCS during an MSIV closure event.

4. Reactor Vessel Water Level--Low, Level 3

The Reactor Vessel Water Level--Low, Level 3 function is assumed in the analysis of the recirculation line break (Ref. 1). The reactor scram reduces the amount of energy required to be absorbed and, along with the actions of the Emergency Core Cooling system (ECCS), assures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46. For this facility, Reactor Vessel Water Level--Low, Level 3 generates a reactor trip signal to ensure that the fuel cladding peak temperature limits are not exceeded due to the loss of reactor vessel water inventory for recirculation line break events.

5. Main Steam Isolation Valve--Closure

Main Steam Isolation Valve--Closure generates reactor trip signals to limit the energy that must be absorbed and together with the ECCS ensures that fuel cladding peak temperature limits are not exceeded following loss of the normal heat sink. For the overpressurization protection analysis of Reference 7,

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

however, the high neutron flux scram function along with the S/RVs limits the peak reactor vessel pressure to less than the ASME code limits. Additionally, MSIV closure is assumed in the transients (low steam line pressure, manual closure of MSIVs, high steam line flow) analyzed in Reference 1. The reactor scram reduces the amount of energy required to be absorbed and, along with the actions of the ECCS, assures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46. For this facility, the MSIV--Closure trip provides protection for turbine trip and loss of normal heat sink, MSIV closure on low steam line pressure, MSIV closure on high steam line flow, and manual closure of MSIV events.

6. Drywell Pressure--High

Drywell Pressure--High function is a diverse reactor trip signal to the Level 3 scram for loss-of-coolant accident (LOCA) events inside containment. A reactor scram is initiated to minimize the possibility of fuel damage and to reduce the amount of energy being added to the coolant and to the drywell. For this facility, the Drywell Pressure--High function provides protection for LOCA inside containment events.

7. Scram Discharge Volume Water Level--High

Scram Discharge Volume Water Level--High generates a scram anticipating the plant condition that the water level in the SDV is increasing and could, if allowed to continue, increase to near the point where there would be insufficient volume available to accept water discharged during a scram. This function prevents common-cause failure of the CRD System due to high SDV level. For this facility, the SDV Water Level--High protection is provided for SDV high water-level events.

8. Turbine Stop Valve--Closure

TSV--Closure generates a reactor trip signal to reduce the amount of energy that must be absorbed following the loss of the turbine as a heat sink. The TSV--Closure function is the primary scram signal for the turbine trip event analyzed in Reference 7. Together

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

with the ECCS TSV--Closure scram assures that the fuel cladding peak temperatures remain below the limits of 10 CFR 50.46 and provides protection for TSV closure events.

9. Turbine Control Valve Fast Closure, Trip Oil Pressure--Low

A TCV fast closure generates a reactor trip signal to prevent fuel damage from significant reactivity increases in the event of the loss of the turbine as a heat sink. The TCV Fast Closure, Trip Oil Pressure--Low trip function is the primary scram signal for the generator load rejection event analyzed in Reference 1. Together with the ECCS, the TCV fast closure trip ensures the fuel cladding SL is not exceeded by limiting the amount of energy that must be absorbed for a generator load rejection event.

10. Reactor Mode Switch--Shutdown Position

The Reactor Mode Switch--Shutdown Position provides appropriate protective functions for the condition in which the reactor is to be operated. The reactor is to be shut down with all control rods inserted when the mode switch is in shutdown. To enforce the condition defined for the shutdown position, placing the mode switch in the shutdown position initiates a reactor scram. The scram signal is removed after a 2-second time delay, permitting a scram reset, which restores the normal valve lineup in the CRD hydraulic system.

The reactor mode switch is also incorporated into other trip functions. It bypasses the main steam line isolation scram when in the shutdown, refueling, or startup position. In the refuel, startup, and run positions, it bypasses the neutron monitoring trips that would prevent operation in those modes. Therefore, the reactor mode switch is part of the neutron monitoring and main steam line isolation trip functions and has a role in mitigating all AOOs and DBAs mitigated by these functions.

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

11. Manual Scram

Manual Scram provides manual reactor trip capability. The manual reactor trip ensures that the control room operator can initiate a reactor trip at any time by depressing two of four reactor trip switches in the control room. A manual reactor trip accomplishes the same results as any one of the automatic trip functions. It is to be used by the reactor operator to shut down the reactor whenever any parameter is rapidly trending toward its trip setpoint.

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

LCO

The LCO requires all RPS instrumentation channels to be OPERABLE because all channels OPERABLE is the NRC Staff-approved design for the normal operating RPS configuration. Typically this is four channels. Exceptions are referred to in the discussion of RPS functions which follows. Furthermore, with one channel inoperable in any function and not in trip, the RPS cannot meet both the redundancy and testability requirements of GDC 21 in 10 CFR 50, Appendix A (Ref. 2).

Violation of the LCO could allow the plant to reach conditions during steady-state and transient operation beyond those evaluated for safe operation. If exceeded, these conditions could lead to fuel failures.

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 continued 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 transient and accident analysis in order to account for instrument uncertainties

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BASES (continued)

LCO (continued)

appropriate to the trip function. These uncertainties are defined in the plant-specific setpoint methodology (Ref. 3).

RPS trip functions specified in Table 3.3.1.1-1 are OPERABLE when:

(1) All channel components necessary to provide a reactor trip signal are functional and in service;

(2) Channel measurement uncertainties are known "via test," analysis, or design information to be within the assumptions of the setpoint calculations;

(3) Required surveillance testing is current and has demonstrated performance within each surveillance test's acceptance criteria: and

(4) The associated operational bypass or operational interlock(s), if any, is not enabled except under the conditions specified by the LCO applicability statement for the function.

The following Bases for each trip function will identify items (1), (2), (3) and (4) above, which are applicable to establish function OPERABILITY requirements.

It should be noted that LCO 3.3.1.1, "RPS Instrumentation," may need to be augmented with additional conditions, if it is determined that the RPS provides support to other systems included in the Standard Technical Specifications.

1. Intermediate Range Monitors

1.a. Neutron Flux--High

The IRMs ensure that fuel damage caused by neutron flux excursions from operational transients in the intermediate power range is prevented. The IRM system is divided into two groups of IRM channels, with four IRM channels inputting to each trip system. The analysis of Reference 1 assumes that one channel in each trip system is bypassed. Therefore, six channels with three channels in each trip system are required for IRM OPERABILITY. The analysis of Reference 1 has adequate conservatism to permit the IRM

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BASES (continued)

LCO (continued)

ALLOWABLE VALUE to be set at 120 divisions of a 125-division scale. [For this facility, the basis for the ALLOWABLE VALUE is as follows:] This trip is active in each of the 10 ranges of the IRM, which must be selected by the operator to maintain the neutron flux within the monitored level of an IRM range.

For this scram point to be valid, the IRM must be on the correct range. A rod block is initiated any time the IRM is both downscale and not on the most sensitive (lowest) scale. A rod block is initiated if the IRM detectors are not fully inserted in the core, unless the reactor mode switch is in the run position.

The IRM scram trips and the IRM rod block trips are automatically bypassed when the reactor mode switch is in the run position.

IRM Neutron Flux--High trip is OPERABLE when it satisfies OPERABILITY requirements items (1), (2), (3), and (4).

1.b. Inop

This signal generates a reactor trip when the minimum number of IRMs are not OPERABLE. When the voltage drops below a preset level, when one of the modules is not plugged in, or when the operate and calibrate switch is not in the "operate" position, an inoperative trip signal will be received by the RPS unless the IRM is bypassed. Since only one IRM in each RPS trip system may be bypassed, only one IRM in each RPS trip system may be taken out of "operate" without resulting in an RPS trip signal. Six channels of IRM Inop with three channels in each trip system are required to be OPERABLE. This function is required to be OPERABLE when the IRM Neutron Flux--High function is required.

This function must be OPERABLE to ensure the nuclear instrument channel fails in the tripped condition upon loss of detector voltage. IRM

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BASES (continued)

LCO (continued)

Inop trip is OPERABLE when it satisfies OPERABILITY requirement items (1), (2), (3), and (4).

2. Average Power Range Monitors

2.a. Neutron Flux--High, Setdown

For operation at low power (i.e., MODE 2), the APRM Neutron Flux--High, Setdown function protects the thermal power SL of 25% RTP for low-flow, or low-pressure conditions by generating a trip signal that prevents fuel damage resulting from operational transients in this power range.

The APRM system is divided into two groups of channels with three APRM channel inputs to each trip system. The system is designed to allow one channel in each trip system to be bypassed. Any one APRM channel in a trip system can cause the associated trip system to trip. Thus four channels of APRM Neutron Flux--High, Setdown, with two channels in each trip system, are required to be OPERABLE to ensure that no single failure will preclude a scram from this function on a valid signal. In addition, to provide adequate coverage of the entire core, at least 11 OPERABLE LPRM inputs are required for each APRM channel, with at least two OPERABLE LPRM inputs from each of the four axial levels at which the LPRMs are located.

For this facility, the ALLOWABLE VALUE is based on safety analyses that take credit for the APRM Flow Biased Thermal Power--High function for the mitigation of the loss of feedwater heating event.

[At this facility, the APRM Neutron Flux--High setdown trip is operationally bypassed as follows:]

The APRM Neutron Flux--High, Setdown trip is OPERABLE when it satisfies OPERABILITY requirements items (1), (2), (3), and (4).

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BASES (continued)

LCO (continued)

2.b. <u>Fl((Biased Simulated Thermal Power--High</u>

The APRM Flow Biased Thermal Power--High function trip level is varied as a function of recirculation drive flow (i.e., at lower core flows, the setpoint is reduced in proportion to the reduction in power experienced as core flow is reduced with a fixed control rod pattern) but is clamped at an upper limit.

The APRM system is divided into two groups of channels with three APRM inputs to each trip system. The system is designed to allow one channel in each trip system to be bypassed. Any one APRM channel in a trip system can cause the associated trip system to trip. Thus, four channels of APRM Flow Biased Simulated Thermal Power--High with two channels in each trip system are required to be OPERABLE to ensure that no single failure will preclude a scram from this function on a valid signal.

Each APRM channel receives two independent redundant total flow signals representative of total core flow. This arrangement is repeated in each trip system for a total of four flow channels. Each signal is provided by summing up the flow signals from the two recirculation loops. Each total flow summation receives input from its own pair of flow sensors, one in each recirculation loop. Therefore, no single active component failure can cause more than one of the total flow signals to read incorrectly. To obtain the most conservative reference signals, the two total flow signals in each trip system are routed to a low auction circuit associated with each APRM in that trip system. This circuit selects the lower of the two signals for use as the scram trip reference for that particular APRM. Consequently, OPERABILITY of the APRM Flow Biased Simulated Thermal Power--High channel requires OPERABILITY of the two total flow channels and four loop flow channels. [Failure of any of these channels renders both APRM Flow-Biased Simulated Thermal Power--High channels in the associated trip system inoperable.]

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BASES (continued)

LCO (continued)

[For this facility, the ALLOWABLE VALUE basis is as follows:].

To provide adequate coverage of the entire core, at least 11 LPRM inputs are required to be OPERABLE for each APRM channel with at least two LPRM inputs from each of the four axial levels at which the LPRMs are located.

The APRM Flow Biased Simulated Thermal Power--High trip is OPERABLE when it satisfies OPERABILITY requirement items (1), (2), (3), and (4).

2.c. Fixed Neutron Flux--High

The APRM channels provide the primary indication of neutron flux within the core and respond almost instantaneously to neutron flux increases. The APRM Fixed Neutron Flux--High function generates trip signals to prevent fuel damage or excessive RCS pressure.

The APRM system is divided into two groups of channels, with three APRM channels inputting to each trip system. The system is designed to allow one channel in each trip system to be bypassed. Any one APRM channel in a trip system can cause the associated trip system to trip. Four channels of APRM Fixed Neutron Flux--High with two channels in each trip system are required to be OPERABLE to ensure that no single failure will preclude a scram from this function on a valid signal. To provide adequate coverage of the entire core, at least 11 LPRM inputs are required for each APRM channel with at least 2 LPRM inputs from each of the 4 axial levels at which the LPRMs are located. [For this facility, the basis for ALLOWABLE VALUE is as follows:]

The APRM Fixed Neutron Flux--High trip is OPERABLE when it satisfies OPERABILITY requirements items (1), (2), (3), and (4).

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BASES (continued)

LCO (continued)

2.d. Downscale

This signal ensures that there is adequate Neutron Monitoring System protection if the reactor mode switch is placed in the run position prior to the APRMs coming on scale.

With the reactor mode switch in run an APRM downscale signal coincident with an associated IRM Neutron Flux--High signal generates a trip signal.

The APRM is divided into two groups of channels with three inputs into each trip system. The system is designed to allow one channel in each trip system to be bypassed. Four channels of APRM Downscale with two channels in each trip system are required to be OPERABLE to ensure that no single failure will preclude a scram from this function on a valid signal. The APRM Downscale trip is OPERABLE when it satisfies OPERABILITY requirements items (1), (2), (3), and (4).

[For this facility, the basis for the ALLOWABLE VALUE is as follows:]

2.e. Inop

This signal ensures that a minimum number of APRMs are OPERABLE. Anytime an APRM mode switch is moved to any position other than operate, an APRM module is unplugged, or the electronic operating voltage is low, an inoperative signal will be received by the RPS. Four channels of APRM Inop with two channels in each trip system are required to be OPERABLE to ensure that no single failure will preclude a scram from this function on a valid signal.

The APRM Inop trip is OPERABLE when it satisfies OPERABILITY requirements items (1), (2), (3), and (4).

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BASES (continued)

LCO (continued)

3. Reactor Vessel Steam Dome Pressure--High

The Reactor Vessel Steam Dome Pressure--High function initiates a scram for transients that results in a pressure increase, counteracting the pressure increase by rapidly reducing core power.

Four channels of the Reactor Vessel Steam Dome Pressure--High function with two channels in each trip system are required to be OPERABLE in MODES 1 and 2 when the RCS is pressurized and the potential for pressure increase exists. To provide adequate coverage, reactor pressure is measured at two locations. A pipe from each location terminates in the reactor building where two local rack-mounted pressure transmitters monitor the pressure in each pipe. Each trip unit provides a high pressure signal to one channel. Two trip units provide an input to trip system A and two trip units provide an input to trip system B. Each trip system operates on a one-out-of-two trip logic. This ensures that no single failure will preclude a scram from this function on a valid signal.

The Reactor Vessel Steam Dome Pressure--High ALLOWABLE VALUE is chosen slightly above the maximum normal operating pressure in order to accommodate expected operational transients without generating a spurious scram, yet provide a sufficient margin to the RCS ALLOWABLE VALUE pressure limits (1250 psig).

The Reactor Vessel Steam Dome Pressure--High trip is OPERABLE when it satisfies OPERABILITY requirements items (1), (2), (3), and (4).

4. Reactor Vessel Water Level--Low, Level 3

Low reactor vessel water level indicates the capability to cool the fuel may be threatened. Should reactor vessel water level decrease too far, fuel damage could result. Therefore, a reactor scram is initiated at Level 3 to substantially reduce the heat generated in the fuel from fission.

Four channels of Reactor Vessel Water Level--Low, Level 3 function with two channels in each trip system

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BASES (continued)

LCO (continued)

are required to be OPERABLE to ensure that no single failure will preclude a scram from this function on a valid signal. Each trip system operates on a one-out-of-two trip logic. The Reactor Vessel Water Level--Low, Level 3 signal is initiated from nonindicating-type differential pressure transmitters that sense the difference between the pressure due to a constant reference column of water and the pressure due to the actual water level in the vessel. Cables from these transmitters are routed to the associated differential pressure-indicating switches (trip units) located in the main control room. The transmitters are arranged on two sets of taps. Two instrument lines attached to taps on the reactor vessel, one above and one below the water level, are required for the differential pressure measurement for each transmitter. The two pairs of lines terminate outside the primary containment and inside the reactor building. They are physically separated from each other and tap off the reactor vessel at widely separated points. Other systems sense pressure and level from these same pipes. The physical separation and signal arrangement ensure that no single physical event can prevent a scram due to reactor vessel low water level.

The Reactor Vessel Water Level--Low, Level 3 ALLOWABLE VALUE is selected to ensure that, during normal operation, the separator skirts are not uncovered. This protects available recirculation pump net positive suction head from significant carryunder and, for transients involving loss of all normal feedwater flow, initiation of the low-pressure ECCS subsystems at reactor vessel water Level 1 will not be required.

The Reactor Vessel Water Level--Low, Level 3 trip is OPERABLE when it satisfies OPERABILITY requirements items (1), (2), (3), and (4).

5. <u>Main Steam Isolation Valve--Closure</u>

Main Steam Isolation Valve--Closure causes the loss of the main turbine and the condenser as a heat sink for the nuclear steam supply system and indicates a need to shut down the reactor to reduce heat generation.

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BASES (continued)

LCO (continued)

MSIV closure signals are initiated from position switches located on each of the eight MSIVs. Each position switch actuates two contacts--one inputs to RPS trip system A while the other contact inputs to RPS trip system B. Thus each RPS trip system receives an input from eight MSIV--Closure channels each consisting of one position switch. The logic for the MSIV--Closure function is arranged such that either the inboard or outboard valve on three or more of the main steam lines must close in order for a scram to occur. The MSIV--Closure ALLOWABLE VALUE is specified to ensure that a scram occurs prior to a significant reduction in steam flow, thereby reducing the severity of the subsequent pressure transient.

Sixteen channels of the MSIV--Closure function, with eight channels in each trip system, are required to be OPERABLE to ensure that no single failure will preclude the scram from this function if any three main steam lines should isolate. [At this facility, the MSIV--Closure trip is operationally bypassed as follows:]

The MSIV--Closure trip is OPERABLE when it satisfies OPERABILITY requirements items (1), (2), (3), and (4).

[For this facility, the basis for ALLOWABLE VALUE is as follows:]

6. Drywell Pressure--High

High pressure in the drywell could indicate a break in the RCS pressure boundary.

Four channels of Drywell Pressure--High function with two channels in each trip system are required to be OPERABLE to ensure that no single failure will preclude a scram from this function on a valid signal. Each trip system operates on a one-out-of-two logic configuration.

High drywell pressure signals are initiated from pressure transmitters that sense the pressure at four different locations in the drywell. The transmitters themselves are located inside the reactor building but outside the drywell. The ALLOWABLE VALUE was selected

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BASES (continued)

LCO (continued)

to be as low as possible, but not so low as to cause spurious trips. [For this facility, the basis for the ALLOWABLE VALUE is as follows:]

The Drywell Pressure--High trip is OPERABLE when it satisfies OPERABILITY requirements items (1), (2), (3), and (4).

7. Scram Discharge Volume Water Level--High

The SDV receives the water displaced by the motion of the CRD pistons during a reactor scram. A reactor scram is initiated when the water level has reached a point high enough to indicate water flow accumulation in the volume, but the remaining free volume is still sufficient to accommodate the water from a full core scram.

SDV water level is measured by two diverse methods. [For this facility, the basis for the need for diverse SDV level measurement is as follows:] The level in each of the two scram discharge volumes is measured by two float-type level switches and two thermal probes for a total of eight level signals. The outputs of these devices are arranged on separated taps, and the outputs are connected logically so that there is a signal from a level switch and thermal probe in each RPS logic channel.

Eight channels, four of each type of Scram Discharge Volume Water Level--High function, with four channels, two of each type, in each trip system are required to be OPERABLE to ensure that no single failure will preclude a scram from this function on a valid signal.

[For this facility the basis for ALLOWABLE VALUE is as follows:]

The Scram Discharge Volume Water Level--High trip is OPERABLE when it satisfies OPERABILITY requirements items (1), (2), (3), and (4).

[For this facility, the basis for the logic configuration is as follows:]

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BASES (continued)

LCO (continued)

8. Turbine Stop Valve--Closure

Closure of the TSVs results in the loss of a heat sink, which produces reactor pressure, neutron flux, and heat flux transients that must be limited.

Eight channels of TSV--Closure function with four channels in each trip system are required to be OPERABLE to ensure that no single failure will preclude a scram from this function if any three turbine stop valves should begin to close. Closure of the turbine stop valves is determined by measuring the position of each valve. There are two independent position switches associated with each turbine stop valve. One of the two switches inputs to RPS trip system A, while the other switch inputs to RPS trip system B. Thus, each RPS trip system receives an input from four TSV--Closure channels, each consisting of one position switch. The logic for the TSV--Closure function is such that three or more turbine stop valves must be closing to produce a scram. [Plant-specific Technical Specifications (TS) to include logic configuration.] The TSV--Closure ALLOWABLE VALUE is selected to be high enough to detect imminent turbine stop valve closure, thereby reducing the severity of the subsequent pressure transient.

The TSV--Closure trip is OPERABLE when it satisfies OPERABILITY requirements items (1), (2), (3), and (4).

9. Turbine Control Valve Fast Closure, Trip Oil Pressure--Low

Fast closure of the TCVs results in the loss of a heat sink that produces reactor pressure, neutron flux, and heat flux transients that must be limited.

Four channels of TCV Fast Closure, Trip Oil Pressure--Low function with two channels in each trip system are required to be OPERABLE to ensure that no single failure will preclude a scram from this function on a valid signal. Fast closure of the turbine control valves is determined by measuring the electro hydraulic control fluid pressure at each control valve. There is one pressure transmitter associated

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BASES (continued)

LCO (continued)

with each control valve, the signal from each transmitter being assigned to a separate RPS logic channel. Each trip system operates on a one-out-of-two trip logic. The TCV Fast Closure, Trip Oil Pressure--Low ALLOWABLE VALUE is selected high enough to detect imminent TCV fast closure but low enough so as to not generate spurious trips due to normal system pressure fluctuations.

The TCV Fast Closure, Trip Oil Pressure--Low trip is OPERABLE when it satisfies OPERABILITY requirements items (1), (2), (3), and (4).

10. Reactor Mode Switch--Shutdown Position

The Reactor Mode Switch--Shutdown Position provides one signal via the manual scram channels to each of the RPS trip systems, which are redundant to the automatic protective instrumentation channels, and provides additional manual reactor trip capability. Two channels of Reactor Mode Switch--Shutdown Position with one channel in each trip system, are available and required to be OPERABLE. Provision of one OPERABLE channel in each trip system is necessary to provide a reactor trip. Thus, this function is not required to meet the single-failure criterion required by GDC 21 (Ref. [2]).

The reactor mode switch also bypasses the main steam line isolation, APRM, and IRM trips in some positions. For these functions, the reactor mode switch has four channels that are considered part of the affected trip functions. For example, if the reactor mode switch fails such that the APRM trip channel is inappropriately bypassed, but the shutdown position trip is still OPERABLE, the APRM channel is considered inoperable and the conditions relevant to that function are entered.

The Reactor Mode Switch--Shutdown Position trip is OPERABLE when it satisfies OPERABILITY requirements items (1), (2), (3), and (4).

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BASES (continued)

LCO (continued)

11. Manual Scram

The Manual Scram push button switches introduce signals into the RPS trip systems that are redundant to the automatic protective instrumentation channels and provide manual reactor trip capability. There is one manual scram switch for each of the four RPS logic channels. The Manual Scram provides input to the manual RPS trip channel stand with the manual push button switches. Four channels are required to be OPERABLE with two channels in each trip system to ensure that the function is not disabled by a single random failure. In order to cause a scram it is necessary that at least one switch in each trip system be actuated.

The Manual Scram trip is OPERABLE when it satisfies OPERABILITY requirements items (1), (2), (3), and (4).

[For this facility, the following support systems are required 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 for whether or not inoperability each declared inoperable are as follows:] It should be noted that LCO 3.3.1.1 may need to be augmented with additional Conditions if it is determined that the RPS provides support to other systems included in the STS.

APPLICABILITY

In MODE 1 the following RPS trip functions are required to be OPERABLE because the reactor is critical. The trips are designed to take the reactor subcritical, which maintains the SLs during AOOs, and assists the ECCS and ESF Systems in providing acceptable consequences during DBAs.

- APRM Flow Biased Simulated Thermal Power--High

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BASES (continued)

APPLICABILITY (continued)

- APRM Fixed Neutron Flux--High
- APRM Downscale
- APRM Inop
- Reactor Vessel Steam Dome Pressure--High
- Reactor Vessel Water Level--Low, Level 3
- MSIV--Closure
- Drywell Pressure--High
- Scram Discharge Volume Water Level--High
  - Resistance Temperature Detector (RTD) Trip Unit
  - Float Switch Trip Unit
- TSV--Closure [≥ 30% RTP]
- TCV Fast Closure, Trip Oil Pressure--Low [≥ 30% RTP]
- Reactor Mode Switch--Shutdown Position
- Manual Scram

In MODE 2 the following RPS trip functions are required to be OPERABLE because the reactor is critical in this mode or control rods may be withdrawn and the potential for criticality exists. The trips are designed to take the reactor subcritical, which maintains the SLs during AOOs, and assists the ECCS and ESF Systems in providing acceptable consequences during DBAs.

- IRM Neutron Flux--High
- IRM Inop
- APRM Neutron Flux--High, Setdown
- APRM Inop
- Reactor Vessel Steam Dome Pressure--High
- Reactor Vessel Water Level--Low, Level 3

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BASES (continued)

APPLICABILITY (continued)

- Drywell Pressure--High
- Scram Discharge Volume Water Level--High
  - RTD trip unit
  - Float Switch Trip Unit
- Reactor Mode Switch--Shutdown Position
- Manual Scram

In MODE 5 with any control rod withdrawn from a core cell containing one or more fuel assemblies, the following RPS trip functions are required to be OPERABLE because in this MODE control rods may be withdrawn and the potential for criticality exists. The trips are designed to take the reactor subcritical, which maintains the SLs during AOOs, and assists the ECCS and ESF Systems in providing acceptable consequences during DBAs.

- IRM Neutron Flux--High
- IRM Inop
- APRM Neutron Flux--High, Setdown
- APRM Inop
- Scram Discharge Volume Water Level--High
  - RTD Trip Unit
  - Float Switch Trip Unit
- Reactor Mode Switch--Shutdown Position
- Manual Scram

For this facility the RPS Applicability bases for each required function are as follows:

1. <u>Intermediate Range Monitors</u>

1.a. <u>Neutron Flux--High</u>

The IRM scram is required to be OPERABLE during MODE 2 when control rods may be withdrawn and the potential for criticality exists. In MODE 5, when a cell with fuel has its control rod

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BASES (continued)

APPLICABILITY (continued)

withdrawn, the IRMs provide monitoring for and protection against unexpected reactivity excursions. In MODE 1, the APRM System and the RWM provide protection against control rod withdrawal error events thus the IRMs are not required.

1.b. Inop

This function is required to be OPERABLE when the IRM Neutron Flux--High function is required to ensure that appropriate monitoring for protection against neutron flux excursions is available.

2. Average Power Range Monitors

2.a. Neutron Flux--High, Setdown

The APRM Neutron Flux--High, Setdown function is required to be OPERABLE during MODE 2 and with control rods withdrawn from any core cell containing one or more fuel assemblies in MODE 5 because the potential for criticality exists, and therefore there is a need for functions that monitor intermediate flux ranges to be available to terminate reactivity excursions. In MODE 1, the APRM Neutron Flux--High function provides protection against reactivity transients and the RWM protects against control rod withdrawal error events and the APRM Neutron Flux--High, Setdown trip is not required.

2.b. Flow Biased Simulated Thermal Power--High

The APRM Flow Biased Simulated Thermal Power--High function is required to be OPERABLE in MODE 1 when there is the possibility of generating excessive thermal power and potentially exceeding the SL applicable to high pressure and core flow conditions in LCO 2.1.2, "Safety Limits," during MODES 2 and 5, other IRM and APRM functions provide protection for fuel-cladding integrity and the APRM Flow Biased Simulated Thermal Power trip is not required.

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BASES (continued)

APPLICABILITY (continued)

2.c. Fixed Neutron Flux--High

The APRM Fixed Neutron Flux--High function is required to be OPERABLE in MODE 1 where the potential consequences of the analyzed transients could result in the SLs (e.g., MCPR and reactor vessel pressure) being exceeded. Although the APRM Fixed Neutron Flux--High function is assumed in the CRD analysis, which is applicable in MODE 2, the APRM Neutron Flux--High, Setdown function conservatively bounds the assumed trip and together with the IRM trips provide adequate protection. Therefore, the APRM Fixed Neutron Flux--High function is not required in MODE 2.

2.d. Downscale

This function is required to be OPERABLE when the reactor mode switch is in run (MODE 1).

[For this facility the Applicability requirements are based on the following:]

2.e. Inop

This Function is required to be OPERABLE in the MODES where the APRM functions are required. These applicability requirements have been previously discussed in functions 2.a, 2.b, and 2.c.

3. Reactor Vessel Steam Dome Pressure--High

Reactor Vessel Steam Dome Pressure--High function channels are required to be OPERABLE in MODES 1 and 2 when the RCS is pressurized and the potential for pressure transients exists.

4. Reactor Vessel Water Level--Low, Level 3

The Reactor Vessel Water Level--Low, Level 3 function is required in MODES 1 and 2 because considerable energy exists in the RCS which could result in the limiting transients and accidents. ECCS initiations

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BASES (continued)

APPLICABILITY (continued)

at Low Low, Level 2, and Low, Level 3, provide sufficient protection for level transients in all other Modes.

5. Main Steam Isolation Valve--Closure

MSIV--Closure function channels are required to be OPERABLE in MODE 1. [For this facility the basis for the Applicability requirements are as follows:] In MODE 2, the heat generation is low enough that the other diverse scram functions provide sufficient protection.

6. Drywell Pressure--High

The Drywell Pressure--High function is required in MODES 1 and 2 when primary containment is required to be OPERABLE and when considerable energy exists in the RCS, which could result in the limiting transients and accidents.

7. Scram Discharge Volume Water Level--High

The Scram Discharge Volume Water Level--High function is required in MODES 1, 2, and 5 when one or more control rods are withdrawn from cells that contain fuel. [For this facility the Applicability requirements are based on the following:]

8. Turbine Stop Valve--Closure

The TSV--Closure function is required, consistent with analysis assumptions, whenever the THERMAL POWER is > 30% RTP. [For this facility, the basis for the Applicability requirements are as follows:] There is an automatic bypass of this trip function below the turbine first-stage pressure value equivalent to THERMAL POWER < 30% RTP since the Reactor Vessel Steam Dome Pressure--High and the Neutron Flux--High functions are adequate to maintain the necessary safety margins.

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BASES (continued)

APPLICABILITY (continued)

9. Turbine Control Valve Fast Closure, Trip Oil Pressure--Low

The TCV Fast Closure, Trip Oil Pressure--Low function is required consistent with the analysis, whenever the THERMAL POWER is ≥ 30% RTP. [For this facility the basis for the applicability requirements are as follows:] The basis for the setpoint of this automatic bypass is identical to that described for the TSV--Closure.

10. Reactor Mode Switch--Shutdown Position

The Reactor Mode Switch--Shutdown Position function is required to be OPERABLE in MODES 1 and 2, and in MODE 5 when control rods in cells that contain one or more fuel assemblies are withdrawn. [For this facility the Applicability requirements are based on the following:]

11. Manual Scram

Manual Scram channels are required to be OPERABLE in MODES 1 and 2, and in MODE 5 when control rods in cells that contain one or more fuel assemblies are withdrawn. [For this facility, the basis for the Applicability requirements is as follows:]

A Note has been added to provide clarification that for this LCO, each function specified in Table 3.3.1.1-1 shall be treated as an independent entity with an independent Completion Time.

ACTIONS

In order for a facility to take credit for topical reports for the basis for justifying Completion Times, topical reports should be supported by an NRC Staff Safety Evaluation Report that establishes the acceptability of each Topical Report for that facility.

A protection function 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

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BASES (continued)

ACTIONS (continued)

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 3.3.1.1-1, the channel must be declared inoperable immediately, and the appropriate Conditions from Table 3.3.1.1-1 must be entered immediately.

In the event a channel's trip setpoint is found nonconservative with respect to the ALLOWABLE VALUE, or the transmitter, instrument loop, signal processing electronics, or bistable 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 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 each of the RPS trip functions in Table 3.3.1.1-1 for a single channel failure.

A.1

This is the preferred Action because it restores full functional capability of the RPS. Reference 7 establishes an acceptable basis for the 12-hour Completion Time.

A.2

If one RPS channel is inoperable in one or more functions, startup or power operation is allowed to continue, provided the inoperable channel(s) are tripped. The provision of four trip channels allows one or both channels of a trip

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(continued)

BASES (continued)

ACTIONS (continued)

system to be tripped during operations, placing the RPS in a one-out-of-two coincidence logic in the OPERABLE trip system.

This is acceptable from an accident analysis standpoint, but it is poor operating practice. With one channel in trip, the plant is at risk that a single random failure will cause a reactor trip and unnecessarily challenge plant safety systems. The 12-hour Completion Time to trip the channel has been evaluated and the reliability of the system shown to be acceptable with a failed, but not tripped, channel for a 12-hour interval (Ref. 7).

If a channel in one trip system becomes inoperable when one or more channels in the opposite system are already in trip, placing the inoperable channel in trip will cause a scram or recirculation pump trip. A Note has been added to indicate that Required Action A.2 is not intended to force a scram or a recirculation pump trip. In this case, if Required Action A.1 cannot be met within 12 hours, then the shutdown track of Condition C is required.

[Plant-specific TS will include a discussion of the MSIV closure trip, IRM trips, TSV closure trip, and reactor mode switch trip as related to Condition A.]

Condition B

Condition B applies to multiple channel failures in all RPS functions. It ensures that trip capability is maintained in each trip system by restoring functions to OPERABLE status or it requires inoperable channels to be placed in trip. With more than one channel inoperable for any RPS function, either by itself or together with single channel failures in other RPS functions, the RPS may not be capable of performing its intended function.

B.1

In this Required Action, "ensure" allows the operator time to evaluate, repair, or trip channels within 1 hour. Unless the function is manual or reactor mode switch, "Maintains trip capability" refers to the ability to provide automatic scram for each function using combinations of OPERABLE and tripped channels without having to also account for single failure. For a four-channel function, this would only

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BASES (continued)

ACTIONS (continued)

require one OPERABLE (or tripped) channel in each trip system. [Plant-specific TS will include discussion on MSIV closure, IRM and TSV--Closure trip logic to explain. "Maintains trip capability" means not required to meet the single-failure criterion.]

The Completion Time of 1 hour is sufficient for plant operations personnel to take corrective actions.

B.2

Required Action B.2 ensures the RPS coincident logic is restored to one trip system within 6 hours. This Action limits the time the RPS scram logic for any function would not accommodate single failure (e.g., one-out-of-one and one-out-of-one arrangement for typical four-channel function). The reduced reliability of this logic arrangement was not evaluated in Reference 7 for the 12-hour Completion Time. Within the 6-hour allowance, each function will have all required channels OPERABLE or in trip (or any combination) in one trip system. This provides an equivalent level of RPS reliability to the 12-hour Completion Time for Condition A which was evaluated and found acceptable in Reference 7. For both Required Action B.1 and Required Action B.2, if compliance with these Required Actions requires channels and trip systems to be in trip, the trip system with the most inoperable channels shall be placed in trip or, alternatively, all the inoperative channels in that trip system shall be placed in trip. If this action would result in a scram then the Required Actions cannot be met. It is permissible to place the other trip system or its inoperable channels in trip.

[Plant-specific TS to include a discussion of the a discussion of the context of this sentence for MSIV, IRM, and turbine stop valve trip functions.]

Condition C

Condition C applies to all RPS instrumentation functions.

C.1

This Required Action directs entry into all remaining Conditions (i.e., except A, B, or C). When any Required Action of Condition A or B is not met and the associated

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BASES (continued)

ACTIONS (continued)

Completion Time has expired, the applicable Condition from Table 3.3.1.1-1 must be entered for each function. The applicable Condition specified in the table is function- and mode-dependent and may change as the Required Action of a previous Condition is completed. Each time an inoperable channel has not met Required Action(s) of Condition A or B, as applicable, and the associated Completion Time has expired, Condition C will be entered for that channel.

Condition D

Condition D applies to TSV--Closure and TCV Fast Closure, Trip Oil Pressure--Low functions.

D.1

If the required number of channels are not restored to OPERABLE status or placed in trip within the allowed Completion Time, the plant must be placed in a MODE or other specified condition in which the LCO does not apply. Two hours is reasonable, based on operating experience related to controlled reduction of power and the time necessary, to reach the specified Conditions from full-power operation in an orderly manner and without challenging plant systems.

Condition E

Condition E applies to APRM Flow Biased Simulated Thermal Power--High; APRM Fixed Neutron Flux--High; APRM--Downscale, and MSIV--Closure functions.

E.1

If the required number of channels are not restored to OPERABLE status or placed in trip within the allowed Completion Time, the plant must be placed in a MODE or other specified condition in which the LCO does not apply. Six hours is based on operating experience related to a controlled reduction of power and the time necessary, to reach MODE 2 in an orderly manner and without challenging plant systems.

Condition F

Condition F applies to IRM Neutron Flux--High (MODE 2); IRM Inop (MODE 2); APRM Neutron Flux--High, Setdown (MODE 2);

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BASES (continued)

ACTIONS (continued)

APRM Inop (MODES 1, 2); Reactor Vessel Steam Dome Pressure--High (MODES 1, 2); Reactor Vessel Water Level--Low, Level 3 (MODES 1, 2); Drywell Pressure--High (MODES 1, 2); Scram Discharge Volume Water Level--High (RTD Trip Unit and Float Switch Trip Unit, MODES 1, 2); Reactor Mode Switch--Shutdown Position (MODES 1, 2); and Manual Scram (MODES 1, 2) functions.

F.1

If the required number of channels are not restored to OPERABLE status or placed in trip within the allowed Completion Time, the plant must be placed in a MODE or other specified condition in which the LCO does not apply. Twelve hours is based on operating experience in controlled reduction in power and the time necessary, to reach MODE 3 in an orderly manner and without challenging plant systems.

Condition G

Condition G applies to IRM Neutron Flux--High (MODE 5[a]); IRM Inop (MODE 5[a]); APRM Neutron Flux--High, Setdown (MODE 5[a]); APRM Inop (MODE 5[a]); Scram Discharge Volume Water Level--High (RTD Trip Unit and Float Switch Trip Unit MODE 5[a]); Reactor Mode Switch--Shutdown Position (MODE 5[a]); and Manual Scram (MODE 5[a]).

G.1

If the required number of channels are not restored to OPERABLE status or placed in trip within the allowed Completion Time, the plant must be placed in a MODE or other specified condition in which the LCO does not apply. This is done by immediately inserting all insertable control rods in core cells containing one or more fuel assemblies. Action must continue until all such control rods are fully inserted.

Condition H

Condition H is applicable to each one of the RPS functions presented i. Table 3.3.1.1-1.

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(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

BASES (continued)

ACTIONS (continued)

H.1

Required Action H.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 operation personnel to make this verification. If verification determines loss of functional capability, LCO 3.0.3 must be immediately entered. However, if the support feature LCO or RPS 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 RPS function are found in the SR column of Table 3.3.1.1-1, for that function. Most functions are subject to CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION.

In order for a facility to take credit for topical reports for the basis for justifying Surveillance Frequencies, topical reports should be supported by an NRC staff Safety Evaluation Report that establishes the acceptability of each topical report for that facility.

Note 2 has been added to the SRs of this LCO to provide that a channel may be placed in an inoperable status for up to 6 hours for required surveillance without placing the trip system in trip provided at least one OPERABLE channel in the same trip system is monitoring that parameter. This Note is based on the RPS reliability analysis (Ref. 7) assumption that 6 hours is the average time required to perform channel surveillance. That analysis demonstrated that the 6-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 6 hours to perform required surveillance testing. Such a practice would be contrary to the assumptions of the reliability analysis that justified LCO Completion Times.

SR 3.3.1.1.1

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 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 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 surveillances are 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 outright channel failure is limited to 12 hours. Since the probability of two random failures in redundant channels within 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 the LCO required channels.

SR 3.3.1.1.2

To ensure that the APRMs are accurately indicating the true core average power, the APRMs are calibrated to reactor power calculated from a heat balance. [LCO 3.2.4 ensures that APRM setpoints are adjusted such that MAXIMUM FRACTION OF LIMITING POWER DENSITY (MFLPD) is less than or equal to fraction of RATED THERMAL POWER (RTP): In accordance with the Core Operating Limits Report for this plant, LCO 3.2.4

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

allows the APRMs to be reading greater than actual THERMAL POWER to compensate for localized power peaking. When this adjustment is made, the requirement for the APRMs to indicate within 2% RTP of calculated power is modified to require the APRMs to indicate within 2% RTP of calculated MFLPD.] The Frequency of once per 7 days is based on minor changes in LPRM sensitivity, which could affect the APRM reading between performances of SR 3.3.1.8, LPRM calibrations.

A Note is provided which only requires the SR at ≥ 25% RTP because it is difficult to accurately determine core THERMAL POWER from a heat balance. [For this facility, accurate measurement of core THERMAL POWER at ≥ 25% RTP is based on the following:]

[For this facility, the bases for possible calibration errors < 25% RTP and how RPS capability is maintained in spite of these errors are as follows:]

SR 3.3.1.1.3

The APRM Flow Biased Simulated Thermal Power--High function uses the recirculation loop drive flows to vary the trip setpoint. This SR ensures that the flow signal used to vary the setpoint is appropriately calibrated and therefore will accurately reflect the required setpoint as a function of flow.

[For this facility, the 7-day SR Frequency basis is as follows:]

SR 3.3.1.1.4

A CHANNEL FUNCTIONAL TEST is performed every 7 days on the IRM and APRM functions required in MODE 2 to ensure that the entire channel will perform its function when required. A test Frequency of 7 days provides an acceptable level of system average unavailability over the surveillance test interval.

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, interlocks, and alarms function when the input is beyond the trip point.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

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 VALUES 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 which cannot be corrected by recalibration.

If, during a CHANNEL FUNCTIONAL TEST, the associated trip setting is discovered to be less conservative than the ALLOWABLE VALUE specified in Table 3.3.1.1-1, the channel must be declared inoperable. The Frequency of SR 3.3.1.1.4 allows entry into MODE 2 from MODE 1 since testing of the MODE 2 required IRM and APRM functions cannot be performed in MODE 1 without utilizing jumpers, lifted leads, or moveable links. The SR must be performed within 12 hours after entering MODE 2 from MODE 1.

Twelve hours is a reasonable time, based on operating experience, to complete the SR.

SR 3.3.1.1.5

A CHANNEL FUNCTIONAL TEST is performed every 7 days on IRM and APRM functions required in MODE 5 with any control rod withdrawn from a core cell containing one or more fuel assemblies to ensure that the entire channel will perform its function when required. A Frequency of 7 days provides an acceptable level of system average availability over the surveillance test interval.

A CHANNEL FUNCTIONAL TEST verifies the functioning of the trip, interlock, and alarm functions of the channel. The CHANNEL FUNCTIONAL TEST SR test basis is discussed above in SR 3.3.1.1.4.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.1.6 and SR 3.3.1.1.7

The overlap between SRMs and IRMs is required to be demonstrated to ensure that reactor power will not be increased into a neutron flux region without adequate indication. This is required each time the transition from SRMs to IRMs is made. These surveillances are established to ensure that no gaps in flux indication from subcritical to power operation exist for monitoring core reactivity status. The overlap between IRMs and APRMs is of concern when reducing power into the IRM range. On power increases, the system design will prevent further increases (by initiating a rod block) if adequate overlap is not maintained. [For this facility, the basis for the 7-day Frequency for SR 3.3.1.1.7 is as follows:]

SR 3.3.1.1.8

LPRM gain settings are determined from the local flux profiles measured by the Transversing Incore Probe (TIP) System. This establishes the relative local flux profile for appropriate representative input to the APRM system. The 1000 MWD/T Surveillance Frequency is based on LPRM sensitivity changes.

SR 3.3.1.1.9

A CHANNEL FUNCTIONAL TEST is performed every 92 days to ensure the entire channel will perform its function when required. The Surveillance Frequency is based on a reliability analysis (Ref. 7).

A CHANNEL FUNCTIONAL TEST verifies the function of the trip, interlock, and alarm functions of the channel. The CHANNEL FUNCTIONAL TEST SR test basis is discussed above in SR 3.3.1.1.4.

The CHANNEL FUNCTIONAL TEST for the APRM Flow Biased Simulated Thermal Power--High function includes testing the turbine first-stage pressure bypass channels. [For this facility, the surveillance test acceptance criteria are as follows:]

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

[SR 3.3.1.1.10]

Calibration of trip units consists of a test to determine actual trip setpoints. Trip setpoints are adjusted if found to be outside the "as found" acceptance limits. If, during trip unit calibration, the associated trip setting is discovered to be less conservative than the specified ALLOWABLE VALUE, the channel must be declared inoperable. Measurement and setpoint error determination and readjustment must be performed consistent with the assumptions of the plant-specific setpoint analysis.

The Surveillance Frequency of 92 days is based on the assumptions of the reliability analysis and on assumptions in the methodology included in the determination of the trip setpoint SR 3.3.1.1.10 and SR 3.3.1.1.9 are often performed simultaneously using a common procedure.

SR 3.3.1.1.11

A CHANNEL CALIBRATION is performed every 6 months for APRM trip functions except for downscale and inop. The 6-month assumption is made in the determination of the magnitude of equipment drift in the setpoint analysis. This test is a complete check of the instrument channel, excluding the sensor.

The test verifies the channel responds to measured parameter values 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 setpoints 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 assumed by the setpoint analysis. However, repeated failures of the same channel over a relatively small number

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

of test intervals must be considered as potentially indicating a deterministic failure which cannot be corrected by recalibration.

The Surveillance Frequency is based upon the assumption of a 6-month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

Neutron detectors are excluded from CHANNEL CALIBRATION because they are passive devices, with minimal drift, and because of the difficulty of simulating a meaningful signal. Changes in detector sensitivity are compensated for by performing the 7-day calorimetric calibration (SR 3.3.1.1.2) and the 1000 MWD/T LPRM calibration against the TIPs (SR 3.3.1.1.8).

SR 3.3.1.1.12

A CHANNEL FUNCTIONAL TEST is performed every 18 months to ensure that the entire channel will perform its intended function when needed.

A CHANNEL FUNCTIONAL TEST verifies the function of the trip, interlock, and alarm functions of the channel. The CHANNEL FUNCTIONAL TEST SR test basis was previously discussed in SR 3.3.1.1.4.

The 18-month Frequency was developed considering that it is prudent that these surveillances be performed only during a plant outage. This is 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 that these components usually pass the Surveillance when performed at the 18-month Frequency.

SR 3.3.1.1.13

A CHANNEL CALIBRATION is performed every 18 months, or approximately at 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. CHANNEL CALIBRATION is a complete check of the instrument channel including the detector. The CHANNEL CALIBRATION SR testing basis was previously discussed in SR 3.3.1.1.11.

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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.

A Note has been added that the neutron detectors may be excluded from the CHANNEL CALIBRATION because a meaningful calibration signal cannot be readily generated. The detectors are fission chambers that are calibrated in the laboratory. The lifetime of the detector is determined by the Environmental Qualification Program. The detector will be periodically replaced based on this predetermined lifetime. Detector failure causes total loss of signal rather than drift to wrong indication. Changes in detector sensitivity are compensated for by performing the 7-day calorimetric calibration (SR 3.3.1.1.2) and the 1000 MWD/T LPRM calibration against the TIPs (SR 3.3.1.1.8).

SR 3.3.1.1.14

The APRM Flow Biased Simulated Thermal Power--High function uses an electronic filter circuit to generate a signal proportional to the core thermal power from the APRM neutron flux signal. This filter circuit is representative of the fuel heat transfer dynamics that produce the relationship between the neutron flux and the core thermal power. The filter time constant must be verified to ensure that the channel is accurately reflecting the desired parameter.

[For this facility, the Surveillance Frequency is based on the following:]

SR 3.3.1.1.15

Performance of a LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required logic for a specific channel. The LOGIC SYSTEM FUNCTIONAL TEST tests all logic components (i.e., all relays and contacts, all trip units, solid state logic elements, etc.) of a logic circuit, from sensor up to the actuated device. The functional testing of control rods, in LCO 3.1.3, and of SDV vent and drain valves, in LCO 3.1.8, overlaps this test to provide complete testing of the measurement safety function.

The 18-month Frequency was developed considering that it is prudent that these surveillances be performed only during a plant outage. This is due to the plant conditions needed

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

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 that these components usually pass the Surveillance when performed at the 18-month Frequency.

SR 3.3.1.1.16

This SR ensures that scrams initiated from the TSV Closure and TCV Fast Closure, Trip Oil Pressure--Low functions will not be inadvertently bypassed when at > 30% RTP. If bypassed, the affected functions must be declared inoperable and the appropriate Required Actions taken, or, alternatively, the bypass channel can be placed in the conservative condition (nonbypass). If placed in the nonbypass condition, this SR is met and no additional actions are required.

[For this facility, the 18-month Surveillance Frequency is as follows:]

SR 3.3.1.1.17

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. This test may be performed in one measurement or in overlapping segments, with verification that all components are tested. The RPS RESPONSE TIME limits are specified in Reference 6.

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 of generating an appropriate detector input signal. Slow changes in detector sensitivity are compensated for by performing the daily calorimetric calibration SR 3.3.1.1.2 and the LPRM calibration against TIPs (SR 3.3.1.1.8). Excluding the detectors is acceptable because the principles of detector operation together with the calibrations against heat balance calculations and incore detectors usually ensure an instantaneous response.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

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 x 18 months, where n is the number of channels per trip system in the function. Testing of the final actuation devices in a trip system, 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 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 time cannot be determined at power since equipment operation is required

REFERENCES

1. [Unit Name] FSAR, Section [ ], "[Title]."
2. Title 10, Code of Federal Regulations, Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants."
3. [Unit Name] Plant-Specific Setpoint Methodology.
4. Title 10, Code of Federal Regulations, Part 50.49, "Environmental Qualifications of Electrical Equipment Important to Safety for Nuclear Power Plants."
5. Institute of Electrical and Electronic Engineers, IEEE Standard 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations," April 5, 1972.
6. [Unit Name] FSAR, Section [7], "[Title]."
7. NEDC-30851-P-A, "Technical Specification Improvement Analyses for BWR Reactor Protection System," March 1988.

B 3.3 INSTRUMENTATION

B 3.3.1.2 Source Range Monitor (SRM) Instrumentation

BASES

BACKGROUND

The SRMs provide the operator with information relative to the neutron flux level at very low levels in the core. As such, the SRM indication is used by the operator to monitor the approach to criticality and determine when criticality is achieved. The SRMs are maintained fully inserted until the count rate is greater than a minimum-allowed count rate (a control rod block is set at this condition). After SRM and Intermediate Range Monitor (IRM) overlap is demonstrated as required by SR 3.3.1.1.6, the SRMs are normally fully withdrawn from the core.

The SRM subsystem of the Neutron Monitoring System (NMS) consists of four channels. Each of the SRM channels can be bypassed, one at a time, by the operation of a bypass switch. Each channel includes one detector that can be physically positioned in the core. Each detector assembly consists of a miniature fission chamber with associated cabling, signal-conditioning equipment, and electronics associated with the various SRM functions, including indication, alarm, and control rod blocks. This LCO specifies OPERABILITY requirements only for the monitoring and indication functions of the SRMs. The signal-conditioning equipment converts the current pulses from the fission chamber to analog DC currents that correspond to the count rate.

During refueling, shutdown, and low-power operations, the primary indication of neutron flux levels is provided by the SRMs or special movable detectors connected to the normal SRM circuits. The SRMs provide monitoring of reactivity changes during fuel or control rod movement and give the control room operator early indication of unexpected subcritical multiplication that could be indicative of an approach to criticality. When the Reactor Protection System (RPS) shorting links are removed, the SRMs also provide added protection against local criticalities by providing an initiating signal for a noncoincidence reactor scram on high neutron flux. However, the SRM instrumentation is

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BASES (continued)

BACKGROUND (continued)	electrically independent of the RPS and as such they are not provided as functions listed in Table 3.3.1.1-1, "RPS Instrumentation."
APPLICABLE SAFETY ANALYSES	The SRMs provide the only on-scale monitoring of neutron flux levels during startup and refueling, and provide added protection (scram) against reactivity excursions when the RPS shorting links are removed. Prevention and mitigation of prompt reactivity excursions during refueling and low-power operation is also provided by LCO 3.9.1, "Refueling Interlocks," LCO 3.1.1, "Shutdown Margin," LCO 3.3.1.1, "RPS Instrumentation, IRM Neutron Flux--High and APRM Neutron Flux--High Trip Functions," and LCO 3.3.2.1, "Control Rod Block Instrumentation." The SRM instrumentation satisfies Criterion 2 of the NRC Interim Policy Statement.
LCO	The LCO requires all instrumentation performing the source range neutron monitoring function to be OPERABLE. Failure of any of the required channels renders the SRM instrumentation inoperable and reduces the reliability of the affected function. The LCO on SRM instrumentation ensures that each of the following requirements are met: 1. During startup in MODE 2, [three] of the four SRM channels are required to be OPERABLE to monitor the reactor flux level prior to and during control rod withdrawal, to monitor subcritical multiplication and reactor criticality, and to monitor neutron flux level and reactor period [until the flux level is sufficient to maintain the IRM on range 2 or above]. For this facility, all but one of the channels are required in order to provide a representation of the overall core response during those periods when reactivity changes are occurring throughout the core. (continued)

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BASES (continued)

LCO (continued)

2. In MODES 3 and 4, with the reactor shut down, two SRM channels provide redundant monitoring of flux levels in the core.

3. In MODE 5, Note (b) to Table 3.3.1.2-1 states that each SRM is not required to be OPERABLE until after four fuel assemblies have been loaded adjacent to the SRM in the four locations next to the SRM dry tube and if no other fuel assemblies are in the associated core quadrant. With four fuel assemblies or fewer loaded around each SRM, even with a control rod withdrawn, the configuration will not be critical. During a complete core offload, an SRM outside the fueled region will no longer be required to be OPERABLE since it is not capable of monitoring neutron flux in the fueled region of the core. Thus, CORE ALTERATIONS are allowed in a quadrant with no OPERABLE SRM in an adjacent quadrant provided Table 3.3.1.2-1 Note (d) requirements, that the bundles being loaded or removed are all in a single fueled region containing at least one OPERABLE SRM, are met.

In non-spiral routine operations, two SRMs are required to be OPERABLE to provide redundant monitoring of reactivity changes occurring in the reactor core. Because of the local nature of reactivity changes during refueling, adequate coverage is provided by requiring one SRM to be OPERABLE in the quadrant of the reactor core where CORE ALTERATIONS are being performed and the other SRM to be OPERABLE in an adjacent quadrant. These requirements ensure that the reactivity of the core will be continuously monitored during CORE ALTERATIONS.

Special movable detectors, according to Note (c) of Table 3.3.1.2-1, may be used during CORE ALTERATIONS in place of the normal SRM nuclear detectors. These special detectors must be connected to the normal SRM circuits in the NMS such that the applicable neutron flux indication, control rod blocks, and scram signal can be generated. These special detectors provide more flexibility in monitoring reactivity changes during fuel loading since they can be positioned anywhere within the core during refueling. They must still meet the location requirements of SR 3.3.1.2 and all other surveillance requirements for SRMs.

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BASES (continued)

LCO (continued)

Continuous visual indication of the output of these detectors in the control room is implicitly required for OPERABLE monitoring capability.

[For this facility, an OPERABLE SRM, or special movable detector used in place of an SRM, constitutes the following:]

[For this facility, the following support systems are required to be OPERABLE to ensure SRM or special movable detector OPERABILITY:]

[For this facility, those required support systems which, upon their failure, do not result in the SRMs or special movable detectors being declared inoperable, and their justification are as follows:]

APPLICABILITY

In MODE 1, the average power range monitors provide adequate monitoring of reactivity changes in the core; therefore, the SRMs are not required. Furthermore, the SRMs are off-scale high except at the lower extreme of MODE 1. In MODE 2, with IRMs on range 2 or above, the IRMs provide adequate monitoring and the SRMs are not required. The SRMs are required to be OPERABLE in MODES 2, 3, 4, and 5 prior to the IRMs being on-scale on range 3 to provide for neutron monitoring.

A Note has been added to provide clarification that Condition A and Condition B shall be treated as a single entity with a single Completion Time.

ACTIONS

In the event a channel's Surveillance test results are found non-conservative with respect to the norm, or the transmitter, instrument loop, signal-processing electronics, or trip unit is found inoperable, then all affected functions provided by that channel must be declared inoperable and the LCO Condition entered for the particular function affected.

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BASES (continued)

ACTIONS (continued)

Condition A, Condition B, and Condition C apply to SRMs in MODE 2 with IRMs on range 2 or below. Condition D applies to SRMs in MODE 3 or MODE 4. Condition E applies to SRMs in MODE 5.

Condition A and Condition B

In MODE 2 with the IRMs on range 2 or below, SRMs provide the means of monitoring core reactivity and criticality. With any number of the required SRMs inoperable, the ability to monitor neutron flux is degraded. Therefore, a limited time is allowed to restore the inoperable channels to OPERABLE status.

Provided at least one SRM remains OPERABLE, Required Action A.1 allows 4 hours to restore the required SRMs to OPERABLE status. This time is reasonable because there is adequate capability remaining to monitor the core, there is limited risk of an event during this time, and there is sufficient time to take corrective actions to restore the required SRMs to OPERABLE status or to establish alternate IRM monitoring capability. During this time control rod withdrawal and power increase is not precluded by this Required Action. Having the ability to monitor the core with at least one SRM while proceeding to IRM range 3 or greater (with overlap required by SR 3.3.1.1.6), it is acceptable to exit the Applicability of this LCO and to ensure adequate core monitoring and allow continued operation.

With all required SRMs inoperable, Required Action B.1 allows no positive changes in reactivity (control rod withdrawal must be immediately suspended) due to inability to monitor the changes. Required Action B.2 allows 4 hours to restore monitoring capability prior to requiring control rod insertion. This allowance is based on the limited risk of an event during this time, provided that no control rod withdrawals are allowed, and the desire to concentrate efforts on repair rather than to immediately shut down with no SRMs OPERABLE.

Condition C

In MODE 2, if the minimum number of SRM channels is not restored to OPERABLE status within the allowed Completion

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BASES (continued)

ACTIONS (continued)

Time, the reactor shall be placed in MODE 3. With all control rods fully inserted, the core is in its least reactive state with the most margin to criticality. The 12-hour Completion Time to reach MODE 3 is reasonable, based on operating experience, to reach MODE 3 in an orderly manner and without challenging plant systems.

Condition D

With one or two required SRM channels inoperable in MODES 3 or 4, the neutron flux monitoring capability is degraded or nonexistent. The requirement to fully insert all insertable control rods ensures that the reactor will be at its minimum reactivity level while neutron monitoring capability is not available. Placing the reactor mode switch in shutdown prevents subsequent control rod withdrawal by maintaining a control rod block. The 1-hour Completion Time is sufficient to accomplish the Required Action and takes into account the low probability of an event requiring the SRM occurring during this interval.

Condition E

With fewer than the required number of OPERABLE SRM channels during MODE 5, the ability to detect local reactivity changes in the core during refueling is degraded. CORE ALTERATIONS must be suspended, and all insertable control rods in core cells containing one or more fuel assemblies must be fully inserted. Suspending CORE ALTERATIONS prevents the two most probable causes of reactivity changes, fuel loading and control rod withdrawal, from occurring. Inserting all insertable control rods ensures that the reactor will be at its minimum reactivity given that fuel is present in the core. Suspension of CORE ALTERATIONS shall not preclude completion of the movement of a component to a safe, conservative position. Action must be initiated and efforts continued to restore the required SRMs to OPERABLE status. Required Action E.3 is provided to assure that the condition of two required SRMs inoperable and the vessel head removed is not construed as an acceptable plant condition allowed even if operations were to be suspended. Entry into MODE 5 without the required SRM channels OPERABLE is not allowed per LCO 3.0.4. Immediate action is required to minimize the time that the plant is exposed to the risk of an undetected reactivity increase.

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BASES (continued)

SURVEILLANCE REQUIREMENTS

The SRs for any particular SRM function are found in the SR column of Table 3.3.1.2-1 for that function.

SR 3.3.1.2.1

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 another channel. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or even something more serious. CHANNEL CHECK will detect gross channel failure, thus it is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff 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 outright 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 the LCO required channels.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.2.2

To provide adequate coverage of potential reactivity changes in the core, one SRM is required to be OPERABLE in the quadrant where CORE ALTERATIONS are being performed; the other OPERABLE SRM must be in an adjacent quadrant. Note 1 states that the SR is required only during CORE ALTERATIONS. This surveillance is a review of plant logs to assure that SRMs required OPERABLE for given CORE ALTERATIONS are in place. In the event that only one SRM is required to be OPERABLE, per Table 3.3.1.2-1, footnote d, only the portion of this SR covering the fueled region is required. Note 2 clarifies that the three verifications do not require separate SRMs to be OPERABLE. The 12-hour Surveillance Frequency is based upon operating experience at this facility in fuel handling and that the core configuration does not significantly change within 12 hours. SR 3.3.1.2.2 supplements operational controls over refueling activities that include steps to ensure the channels required by the LCO are OPERABLE.

SR 3.3.1.2.3

SR 3.3.1.2.3 involves the performance of a CHANNEL CHECK. A CHANNEL CHECK basis is discussed in SR 3.3.1.2.1.

The surveillance interval, once every 24 hours, is based on operating experience that demonstrates the rarity of more than one channel randomly failing within the same time interval. Thus, this SR ensures that an undetected loss of function is unlikely, and that loss of redundancy will be identified within 24 hours.

[For this facility, the basis for the 24-hour CHANNEL CHECK in MODES 3 and 4 being acceptable is as follows:]

SR 3.3.1.2.4

This surveillance is a review of the plant SRM instrument logs to ensure that the SRM reading is greater than a specified minimum count rate, which ensures that the detectors are indicating count rates indicative of neutron flux levels within the core. Verification of the signal-to-noise ratio also ensures that the detectors are inserted to a normal operating level. The SRM system is designed to

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

provide a signal-to-noise ratio of at least 2:1 and a count rate of at least 3 counts per second (cps) with all control rods fully inserted prior to initial power operation (neutron sources and detectors together). For a signal-to-noise ratio of 20:1, the count rate must be at least 0.7 cps. [For this facility, the bases for count rate and signal-to-noise ratio are as follows:] In a fully withdrawn condition, the detectors are sufficiently removed from the fueled region of the core to essentially eliminate neutrons from reaching the detector. Any count rate obtained while the detectors are fully withdrawn is assumed to be noise only.

When no reactivity changes are in progress, the Frequency is relaxed from 12 hours to 24 hours. The Frequency takes into account channel redundancy and other information available in the control room.

SR 3.3.1.2.5

Performance of a CHANNEL FUNCTIONAL TEST demonstrates that the associated channel will function properly. This SR is performed in MODE 5, and the 7-day Frequency ensures that the channels are OPERABLE while core reactivity changes are in progress. This Frequency is reasonable, based on operating experience and other surveillances, such as a CHANNEL CHECK, that provide assurance of proper functioning between CHANNEL FUNCTIONAL TESTS. [At this facility, the following bistable functions are included in the CHANNEL FUNCTIONAL TEST:]

SR 3.3.1.2.6

Performance of a CHANNEL FUNCTIONAL TEST demonstrates that the associated channel will function properly. The Surveillance Frequency allows entry into the specified condition of the Applicability (THERMAL POWER decrease to IRM range 2 or below). The SR must be performed within 12 hours of entering MODE 2 with IRMs on range 2 or below. If the 31-day Frequency is not met, compliance with SR 3.0.2 is required. The 31-day Frequency is based on operating experience at this facility and on other surveillances (such as CHANNEL CHECK) that provide assurance of proper functioning between CHANNEL FUNCTIONAL TESTS. The exception to enter the Applicability with the 31-day Frequency not met

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

is reasonable, based on the limited time of 12 hours allowed after entering the Applicability and the inability to perform the surveillance while at higher power levels. Although the surveillance could be performed while on IRM range 3, the plant would not be expected to maintain steady-state operation at this power level. In this event, the 12-hour Frequency is reasonable, based on the SRMs being otherwise verified to be OPERABLE (i.e., satisfactorily performing the CHANNEL CHECK) and the time required to perform the surveillances.

SR 3.3.1.2.7

Performance of a CHANNEL CALIBRATION at a Frequency of 18 months verifies the performance of the SRM detectors and associated circuitry.

The Frequency considers the unit conditions required to perform the test, the ease of performing the test, and a likelihood of a change in the system or component status. The neutron detectors may be excluded from the CHANNEL CALIBRATION because they cannot be adjusted. The detectors are fission chambers that are calibrated in the laboratory. The lifetime of the detector is determined by the Environmental Qualification Program. The detector will be periodically replaced based on this predetermined lifetime. Detector failure causes total loss of signal rather than drift to wrong indication.

REFERENCES

None.

B 3.3 INSTRUMENTATION

B 3.3.2.1 Control Rod Block Instrumentation

BASES

BACKGROUND

Control rods provide the primary means for control of reactivity changes. Control rod block instrumentation includes channel sensors, logic circuitry, load drivers, switches and relays that are designed to ensure that specified fuel design limits are not exceeded for postulated control rod withdrawal error events. During low-power operations, control rod block instrumentation enforces specific control rod sequences designed to mitigate the consequences of the control rod drop accident (CRDA). During shutdown conditions, control rod block instrumentation inserts withdrawal blocks to ensure that all control rods remain inserted to prevent inadvertent criticalities.

The rod block monitor (RBM) supplies a trip signal to the Reactor Manual Control System (RMCS) to appropriately inhibit control rod withdrawal during power operation above the low-power setpoint. The RBM has two channels, either of which can initiate a control rod block when the channel output exceeds the control rod block setpoint. The RBM channel signal is generated by averaging a set of local power range monitor (LPRM) signals at various core heights surrounding the control rod being withdrawn. A signal from one average power range monitor (APRM) channel assigned to each Reactor Protection System (RPS) trip system supplies a reference signal for the RBM channel in the same trip system. This reference signal is used to determine which RBM range (low, intermediate, or high) is enabled. If the APRM is indicating less than the low-power range setpoint, the RBM is automatically bypassed. The RBM is also automatically bypassed if a peripheral control rod is selected (Ref. 1).

Control rod patterns during startup conditions are controlled by the operator and rod worth minimizer (RWM) so that only specified control rod sequences and relative positions are allowed over the operating range from all control rods inserted to 10% of RATED THERMAL POWER (RTP). The sequences effectively limit the potential amount and rate of reactivity increase during a CRDA. Prescribed

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BASES (continued)

BACKGROUND (continued)

control rod sequences are stored in the RWM, which will initiate control rod withdrawal and insert blocks when the actual sequence deviates beyond allowances from the stored sequence. The RWM determines the actual sequence-based position indication for each control rod. The RWM also uses [feedwater-flow and steam-flow] signals to determine reactor power to automatically bypass the RWM above a preset power level (Ref. 2). The RWM is a single channel system.

With the reactor mode switch in the shutdown position, a control rod withdrawal block is applied to all control rods to ensure that the shutdown condition is maintained. This function prevents inadvertent criticality as the result of a control rod withdrawal during MODE 3 or 4, or during MODE 5 when the reactor mode switch is required to be in the shutdown position.

The trip setpoints used in the bistables are based on the analytical limits presented in Reference 3. 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 Table 3.3.2.1-1 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 the plant-specific setpoint methodology. The actual nominal trip setpoint entered into the bistable is normally still 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 safety limits are not violated during anticipated operational occurrences (AOOs) and that the consequences of accidents will be acceptable, provided the plant is being operated within the LCOs at the onset of the AOO or accident and the equipment functions as designed.

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BASES (continued)

APPLICABLE SAFETY ANALYSES

The RBM is designed to prevent violation of the safety limit MINIMUM CRITICAL POWER RATIO (MCPR) and the cladding 1%-plastic-strain fuel design limit that may result from a single control rod withdrawal error (RWE) event. For this facility, a statistical analysis of RWE events was performed to determine the RBM response for both channels for each event. From these responses, the fuel thermal performance as a function of RBM ALLOWABLE VALUE was determined. The ALLOWABLE VALUES are chosen as a function of power level; based on the specified ALLOWABLE VALUES, operating limits are established.

The RWM enforces the banked position withdrawal sequence (BPWS) to ensure that the initial conditions of the CRDA analysis are not violated. The analytical methods and assumptions used in evaluating the CRDA are summarized in Reference 5. The BPWS requires that control rods be moved in groups; all control rods assigned to a specific group are required to be within specified banked positions. Requirements that the control rod sequence is in compliance with BPWS are specified in LCO 3.1.7.

During MODES 3 and 4, and during MODE 5 when the reactor mode switch is required to be in the shutdown position, the core is assumed to be subcritical; therefore, no positive reactivity insertion events are analyzed. The Reactor Mode Switch--Shutdown Position control rod withdrawal block ensures that the reactor remains subcritical by blocking control rod withdrawal, thereby preserving the assumptions of the safety analysis.

Control rod block instrumentation satisfies Criterion 3 of the NRC Interim Policy Statement.

LCO

The LCO requires all instrumentation performing the control rod block function to be OPERABLE. Failure of any instrument renders the affected channel inoperable and reduces the reliability of the affected function.

The THERMAL POWER values specified in the LCO Applicability for functions 1 and 2 are expressed as ALLOWABLE VALUES for the functions operational bypasses and are determined and set in accordance with the above discussion.

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BASES (continued)

LCO (continued)

A control rod block instrumentation channel is OPERABLE when the following conditions are satisfied:

1. All channel components necessary to provide a block signal are functional and in service;

2. Channel measurement uncertainties are known via test, analysis, or design information to be within the assumptions of the setpoint calculations; and

3. Required Surveillance testing is current and has demonstrated performance within each Surveillance test's acceptance criteria.

4. The associated operational bypass is not enabled except under the conditions specified in the LCO Applicability for the function.

The control rod block instrumentation functions are the Rod Block Monitor, the Rod Worth Minimizer, and the Reactor Mode Switch--Shutdown Position.

ALLOWABLE VALUES are provided as a function of power level. Nominal trip setpoints are specified in the plant-specific setpoint calculations. The nominal setpoints are selected to ensure that the setpoints measured by CHANNEL FUNCTIONAL TESTS do not exceed the ALLOWABLE VALUES. 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 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.

1. Rod Block Monitor

The two channels of the RBM are required to be OPERABLE to provide protection during an RWE event. This protection is provided assuming an additional single failure during the event. The RBM accomplished its function by averaging local neutron flux inputs

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BASES (continued)

LCO (continued)

from two LPRM channels for each RBM channel to ensure core monitoring overlap between RBM channels. The resultant RBM signals are applied to circuits for comparison with flow biased trip points. All components required for the RBM function, including associated APRM inputs, are required to be OPERABLE. [For this facility, the basis for the RBM logic configuration and ALLOWABLE VALUE are as follows:]

a. Low Power Range--Upscale
[For this facility, the bases for this instrument and its trip and bypass ALLOWABLE VALUES are as follows:]

b. Intermediate Power Range--Upscale
[For this facility, the bases for this instrument and its trip and bypass ALLOWABLE VALUES are as follows:]

c. High Power Range--Upscale
[For this facility, the bases for this instrument and its trip and bypass ALLOWABLE VALUES are as follows:]

d. Inop
[For this facility, the bases for this instrument function is as follows:]

e. Downscale
[For this facility, the bases for this instrument and its trip and bypass ALLOWABLE VALUE are as follows:]

f. Bypass Time Delay
[For this facility, the bases for this instrument and its ALLOWABLE VALUE is as follows:]

2. Rod Worth Minimizer

One channel of the RWM is required to be OPERABLE with the proper sequence loaded because the RWM is a hardwired system designed to act as a backup to operator control of the rod sequences. Special circumstances provided for in LCO 3.2.1 and LCO 3.1.7

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BASES (continued)

LCO (continued)

Required Actions may necessitate bypassing the RWM to allow continued operation with inoperable control rods, or to allow correction of a control rod pattern not in compliance with the BPWS. The RWM may be bypassed as required by these conditions, but then it must be considered inoperable and the Required Actions of this LCO followed.

[For this facility, the following constitutes an OPERABLE RWM:] [For this facility, the bases for the bypass ALLOWABLE VALUE is as follows:]

3. Reactor Mode Switch--Shutdown Position

Two channels are required to be OPERABLE to ensure that no single failure will preclude a rod block in this condition. The reactor is required to be shut down with all control rods inserted into the reactor core when the mode switch is in the shutdown position. To enforce this condition, placing the reactor mode switch in the shutdown position initiates a rod block to prevent rod withdrawal. This function of the reactor mode switch plays an important role in mitigating a rod withdrawal event.

[For this facility, those required support systems which upon their failure do not result in the control rod block instrumentation being declared inoperable and their justification, are as follows:]

[For this facility, the following support systems are required to be OPERABLE to ensure control rod block instrumentation OPERABILITY.]

APPLICABILITY

The RBM is assumed to mitigate the consequences of an RWE event when operating above 29% RTP. Below this power level, the consequences of an RWE event will not exceed the Safety Limit (SL) MCPR and therefore the RBM is not required to be OPERABLE. When operating below 90% RTP, analyses have shown that when operating with an initial MCPR > 1.70, no RWE event will result in exceeding the SL MCPR. Also, the analyses demonstrate that when operating at > 90% RTP

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BASES (continued)

APPLICABILITY (continued)

with MCPR < 1.40, no RWE event will result in exceeding the SL MCPR. Therefore, under these conditions, the RBM is also not required to be OPERABLE.

Compliance with the BPWS, and therefore OPERABILITY of the RWM, is required in MODES 1 and 2 when THERMAL POWER is ≤ 10% RTP. When THERMAL POWER is > 10% RTP, there is no possible control rod configuration that results in a control rod worth that could exceed the 280 cal/gm fuel-damage limit during a CRDA. In MODES 3 and 4, all control rods are required to be inserted in the core. In MODE 5, since only a single control rod can be withdrawn from a core cell containing fuel assemblies, adequate SHUTDOWN MARGIN ensures that the consequences of a CRDA are acceptable, since the reactor will be subcritical.

During shutdown conditions (MODE 3, 4, or 5), no positive reactivity insertion events are analyzed because assumptions are that control rod withdrawal blocks are provided to prevent criticality. Therefore, when the reactor mode switch is in the shutdown position, the control rod withdrawal block is required to be OPERABLE. During MODE 5 with the reactor mode switch in the refueling position, the one-rod-out interlock (LCO 3.9.2) also provides the required control rod withdrawal blocks.

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

ACTIONS

In order for a facility to take credit for topical reports as the basis for justifying Completion Times, topical reports should be supported by an NRC staff Safety Evaluation Report (SER) that establishes the acceptability of each topical report for that facility.

A protection function 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 causes of channel inoperability are outright failure or because the bistable or process module has drifted sufficiently so as to exceed

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BASES (continued)

ACTIONS (continued)

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 3.3.2.1-1, the channel must be declared inoperable immediately and the appropriate Conditions from Table 3.3.2.1-1 must be entered immediately.

In the event a channel's trip setpoint is found to be nonconservative with respect to the ALLOWABLE VALUE, or the transmitter, instrument loop, signal processing electronics, or bistable 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.

Condition A

Condition A applies to the rod block monitor channels. If one RBM channel is inoperable, the remaining OPERABLE channel is adequate to perform the control rod block function; however, overall reliability is reduced because a single failure in the remaining OPERABLE channel can result in no control rod block capability for the RBM. For this reason, with at least one RBM channel OPERABLE, continued operation is permitted only for a limited time.

[For this facility, the Completion Time is based upon the following:]

Condition B

If the Required Action of Condition A is not met and the associated Completion Time has expired, or if both RBM channels are inoperable, the RBM is not capable of performing its intended function. At least one channel must be placed in trip to initiate a control rod withdrawal block, thereby ensuring that the RBM function is met.

The 1-hour Completion Time is sufficient to perform the Required Actions.

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BASES (continued)

ACTIONS (continued)

Condition C

Condition C applies to the RWM channels. If the RWM is inoperable, control rod movement is immediately suspended except by scram. Alternatively, startup may continue if at least 12 control rods have already been withdrawn, and startup with an inoperable RWM was not performed in the last 12 months. Required Actions C.2.1 and C.2.2 require verification of these conditions by review of plant logs and control room indications. Once Conditions C.2.1 and C.2.2 are completed, control rod withdrawal may proceed in accordance with the restrictions imposed by Required Action C.2.3. Required Action C.2.3 allows for the RWM function to be performed manually and requires a double check of the manual monitoring by a second licensed operator or other qualified member of the technical staff. [At this facility, a non-licensed member of the technical staff must meet the following criteria to qualify for performing the double check:]

[For this facility the basis for the verification of ≥ 12 rods withdrawn is as follows:] The RWM may be bypassed under these conditions to allow continued operations. In addition, Required Actions of LCO 3.1.2 and LCO 3.1.6 may require bypassing the RWM, during which time the RWM must be considered inoperable and Condition C entered.

Condition D

Condition D applies to reactor mode switch channels. If one reactor mode switch shutdown position control rod withdrawal block channel is inoperable, the remaining OPERABLE channel is adequate to perform the control rod withdrawal block function. The Required Actions are consistent with the normal action of an OPERABLE reactor mode switch shutdown function, however, there is no distinction between having one and two channels inoperable.

In both cases, one or both channels inoperable, fully inserting all insertable control rods immediately in core cells containing one or more fuel assemblies will ensure that the core is subcritical with adequate SHUTDOWN MARGIN assured by LCO 3.1.1. Control rods in core cells containing no fuel assemblies do not affect the reactivity of the core and are therefore not required to be inserted.

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BASES (continued)

SURVEILLANCE REQUIREMENTS

The SRs for any particular control rod block function are found in the SRs column of Table 3.3.2.1, for that function. Most functions are subject to CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION.

In order for a facility to take credit for topical reports as the basis for justifying Surveillance Frequencies, topical reports should be supported by an NRC staff SER that establishes the acceptability of each topical report for that facility. The 92-day Surveillance Frequency is justified in Reference 6.

SR 3.3.2.1.1

Performance of a CHANNEL FUNCTIONAL TEST for the RBM demonstrates that the associated channel will function properly. The Surveillance Frequency is based on generic reliability analyses (Ref. 5).

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, 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 must be found within the ALLOWABLE VALUES 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 which cannot be corrected by recalibration.

SR 3.3.2.1.2

The CHANNEL FUNCTIONAL TEST for the RWM is performed by attempting to withdraw a control rod not in compliance with the prescribed sequence, and verifying that a control rod block occurs. The Surveillance Frequency allows entry into

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

MODES 1 and 2 as well as entry into the condition of THERMAL POWER ≤ 10% RTP to allow performance of the required surveillance if the 92-day Frequency is not met per SR 3.0.2. The Surveillance Frequencies are based on generic reliability analysis (Ref. 6).

[SR 3.3.2.1.3]

The RBM setpoints are automatically varied as a function of power. Three ALLOWABLE VALUES are specified, each within a specific power range. The power ALLOWABLE VALUES that automatically change the control rod block ALLOWABLE VALUES are based on the APRM signal's input to each RBM channel. Below the minimum power setpoint, the RBM is automatically bypassed. These power ALLOWABLE VALUES must be verified periodically to be less than or equal to the specified values. This SR is modified by a Note that for these tests, the neutron detectors are exempt from the Surveillance. [For this facility, the Surveillance Frequency is based on assumptions in the setpoint methodology included in the determination of the trip setpoint.]

[For this facility, the bases for exempting the neutron detectors from testing are as follows:]

SR 3.3.2.1.4

The RWM is automatically bypassed when power is above a specified value. The power level is determined from feedwater- and steam-flow signals. The automatic bypass setpoint must be verified periodically to be equal to or less than the specified value. [For this facility, the Surveillance Frequency of 18 months is based on assumptions in the setpoint methodology included in the determination of the trip setpoint.]

SR 3.3.2.1.5

The CHANNEL FUNCTIONAL TEST for the reactor mode switch shutdown position control rod withdrawal block is performed by attempting to withdraw any control rod with the reactor mode switch in the shutdown position and verifying that a control rod block occurs. The 18-month Surveillance Frequency was developed considering it was prudent that this Surveillance only be performed during a plant shutdown.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

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.

SR 3.3.2.1.6

CHANNEL CALIBRATION is a complete check of the instrument channel. The test verifies that the channel responds to measured values of the parameter within 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 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, for the determination of the magnitude of equipment drift in the setpoint analysis.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

This SR is modified by a Note indicating that the neutron detectors are excluded from the CHANNEL CALIBRATION. [For this facility, the bases for excluding neutron detectors from calibration is as follows:]

SR 3.3.2.1.7

The RWM will only enforce the proper control rod sequence if the rod sequence is properly input into the RWM computer. This SR ensures that the proper sequence is loaded into the RWM so that it can perform its intended function. [At this facility, the verification is performed as follows:] The Surveillance is performed once prior to declaring RWM OPERABLE following loading of sequence into RWM. [For this facility, the bases for this Frequency is as follows:]

REFERENCES

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

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

3. [Unit Name] FSAR, Section [15], "[Title]."

4. NEDE-24011-P-A-9-US, "General Electrical Standard Application for Reload Fuel," Supplement for United States, Section S 2.2.3.1, September 1988.

5. [Unit Name] SER accepting GE Topical Report NEDC-30851-PA Supplement 1, "Technical Specification Improvement Analysis for Control Rod Block Instrumentation," October 1988.

B 3.3 INSTRUMENTATION

B 3.3.2.2 Feedwater and Main Turbine Trip Instrumentation

BASES

BACKGROUND

The feedwater and main turbine trip instrumentation is designed to detect a potential failure of the Feedwater Level Control System that causes excessive feedwater flow. If undetected, this would lead to reactor vessel water carryover into the main steam lines and to the main turbine. The event is termed "Feedwater Controller Failure to Maximum Demand Event" and is analyzed in Reference 1 as "Excess Coolant Inventory."

In this event, the Feedwater Level Control System feedwater controller goes to its upper limit, resulting in an increase in reactor vessel water inventory. This event can be caused by failures in the feedwater flow measuring instrumentation (fails low), failures in the steam flow measuring instrumentation (fails high), or failures in the reactor vessel level measuring instrumentation. Any of these failures would cause the level controller to increase feedwater flow, causing the water level in the reactor vessel to rise toward the high-water level, Level 8 reference point, causing the trip of the three feedwater pumps and the main turbine.

Reactor Vessel Water Level--High, Level 8 signals are provided by level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level in the reactor vessel (variable leg). Pressure and temperature compensation are included in the instrumentation channel. Three channels of Reactor Vessel Water Level--High, Level 8 instrumentation are provided as input to a two-out-of-three initiation logic that trips the three feedwater pumps and the main turbine. A trip of the main turbine, in turn, initiates closure of the turbine stop valves, which scrams the reactor and causes the recirculation pumps to trip.

A trip of the feedwater pumps limits further increase in reactor vessel water level by limiting further addition of feedwater to the reactor vessel. A trip of the main turbine and closure of the stop valves protects the turbine from

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BASES (continued)

BACKGROUND (continued)

damage due to water entering the turbine. The reactor scram and the recirculation pump trip limit the peak neutron flux (power level) and limit the fuel thermal transient to within design basis by maintaining MINIMUM CRITICAL POWER RATIO (MCPR) above 1.15.

APPLICABLE SAFETY ANALYSES

The feedwater and main turbine trip instrumentation is assumed to be capable of providing a turbine trip in the design basis transient analysis for a feedwater controller failure, maximum demand event (Ref. 1). The Level 8 trip indirectly initiates a reactor scram from the main turbine trip and trips the feedwater pumps, thereby terminating the event. The reactor scram mitigates the reduction in MCPR.

Feedwater and main turbine trip instrumentation is designed to detect a potential problem with the feedwater level control system in which the feedwater controller responds to a false demand for maximum feedwater flow. With this excess feedwater flow, reactor vessel water level increases to the high-level reference point. A Reactor Vessel Water Level--High, Level 8 trip signal is provided to trip the feedwater pump turbines and the main turbine to preclude excessive moisture carryover into the main turbine.

Feedwater and main turbine trip instrumentation satisfies Criterion 3 of the NRC Interim Policy Statement.

LCO

The LCO requires three channels of the Reactor Vessel Water Level--High, Level 8 instrumentation to be OPERABLE (Ref. 1) to ensure that no single failure will prevent a feedwater pump and main turbine trip on a valid Level 8 signal. Two of the three channels are needed to provide trip signals in order for the feedwater and main turbine trips to occur. Each channel must have its setpoint set within the specified ALLOWABLE VALUE. The actual setpoint is calibrated to be consistent with the applicable setpoint methodology assumptions.

The ALLOWABLE VALUE is specified in SR 3.3.2.2.3. Nominal trip setpoints are specified in the plant-specific setpoint calculations. The nominal setpoints are selected to ensure

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BASES (continued)

LCO (continued)

the setpoints 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 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 (Ref. 2).

A channel is OPERABLE when the following conditions are met:

- All channel components necessary to provide a trip signal are functional and in service;
- Channel measurement uncertainties are known—by 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.

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

[For this facility, those required support systems which, upon their failure, do not result in the feedwater and main turbine trip instrumentation being declared inoperable and their justification are as follows:]

APPLICABILITY

The feedwater and main turbine trip instrumentation are required to be OPERABLE at ≥ 25% of RATED THERMAL POWER (RTP) to ensure that the fuel-cladding integrity safety limit and the cladding 1%-plastic-strain limit are not violated during the feedwater controller failure, maximum demand event. As discussed in the Bases for LCO 3.2.1 and LCO 3.2.2, sufficient margin to these limits exists below 25% RTP; therefore, these requirements are only necessary when operating at or above this power level.

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BASES (continued)

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 common causes of channel inoperability are 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 the appropriate Conditions must be entered immediately.

In the event a channel's trip setpoint is found nonconservative with respect to the ALLOWABLE VALUE, or the transmitter, instrument loop, signal processing electronics, or bistable 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.

<u>Condition A</u>

OPERABILITY of three channels of the feedwater and main turbine trip instrumentation is required to provide a trip signal to the main turbine. With one channel inoperable, the remaining OPERABLE channels can provide the required trip signal. Overall instrumentation reliability is reduced, however, because a single failure in one of the remaining channels concurrent with a feedwater controller failure, maximum demand, will result in the instrumentation not being able to perform its intended function. Therefore, continued operation is allowed for a limited time with one channel inoperable. If the inoperable channel cannot be restored to OPERABLE status within the Completion Time, the channel must be placed in the tripped condition. If placing the channel in trip will result in tripping the feedwater pumps or main turbine, then the Required Actions of Condition A are not met and Actions of Condition C to reduce THERMAL POWER to less than 25% RTP must be met.

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BASES (continued)

ACTIONS (continued)

[For this facility, the basis for the 7-day Completion Time is as follows:]

Condition B

With two channels inoperable, the feedwater and main turbine trip instrumentation cannot perform the designed trip function. Therefore, continued operation is only permitted for a 1-hour period, during which at least one of the two inoperable channels is required to be restored to OPERABLE status by Required Action B.1 or placed in trip to establish acceptable redundancy in the channel trip logic. If one of the inoperable channels cannot be restored to OPERABLE status within 1 hour, the channel must be placed in the tripped condition. A Note is added stating that if placing the channel in trip will result in tripping the feedwater pumps or main turbine, then the Required Actions of Condition B.2 is not met and the actions of Condition C to reduce THERMAL POWER to less than 25% RTP must be met. This performs the intended function of the channel.

The 1-hour Completion Time is sufficient for the operator to take corrective action and takes into account the likelihood of an event requiring actuation of feedwater and main turbine trip instrumentation occurring during this period.

Condition C

With three channels inoperable, or the inoperable channels not restored to OPERABLE status or trip as per Condition A or Condition B, THERMAL POWER must be reduced to < 25% RTP within 6 hours. As discussed in the Bases of Applicability, operation below 25% RTP results in sufficient margin to the required limits, and the feedwater and main turbine trip instrumentation is not required to protect fuel integrity during the feedwater controller failure, maximum demand event. The Completion Time of 6 hours is based on operating experience related to the time required to reduce power in an orderly manner and without challenging plant systems.

Condition D

Required Action D.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

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BASES (continued)

ACTIONS (continued)

Completion Time is sufficient for plant operations personnel to make this verification. If verification determines loss of functional capability, the Required Actions of Condition B are entered immediately if the equivalent of two channels are inoperable, or Required Actions of Condition C are entered immediately if the equivalent of three channels are inoperable.

SURVEILLANCE REQUIREMENTS

[SR 3.3.2.2.1]

Performance of the CHANNEL CHECK once every 24 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 even something more serious. A CHANNEL CHECK will detect gross channel failure, thus it is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff based on a 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 limits. 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 surveillances are 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 is based on operating experience that demonstrates the rarity of channel failure. Thus, performance of the CHANNEL CHECK guarantees that undetected outright channel failure is limited to 24 hours. Since the probability of two random failures in redundant channels

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

within any 24-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 the LCO required channels.

SR 3.3.2.2.2

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, 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 must be found within the ALLOWABLE VALUES 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 plant-specific setpoint analysis. Recalibrations 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 operating experience that demonstrates the rarity of more than one channel failing within the same interval, and takes into account the CHANNEL CHECK SR 3.3.2.2.1 that is performed at shorter intervals for detection of loss of channel sensors or power supply.

SR 3.3.2.2.3

CHANNEL CALIBRATION is a complete check of the instrument channel, including the detector. The test verifies that the channel responds to a measured values of the 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

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

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.

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 a 18-month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

SR 3.3.2.2.4

Performance of a LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required logic for a specific channel. The LOGIC SYSTEM FUNCTIONAL TEST tests all logic components (i.e., all relays and contacts, trip units, solid-state logic elements, etc.) of a logic circuit, from sensor up to the actuated device. The 18-month Surveillance Frequency was developed considering it is prudent that this surveillance be performed only during a plant shutdown. This is 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 that these components usually pass the surveillance when performed at the 18-month Frequency.

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BASES (continued)

REFERENCES

1. [Unit Name] FSAR, Section [15.1], "[Title]."
2. [Unit Name] FSAR, Section [ ], "[Setpoint Methodology]."

B 3.3 INSTRUMENTATION

B 3.3.3.1 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, Engineered Safety Feature Actuation System, 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 containment 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 containment) and to determine whether 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, containment 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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BASES (continued)

BACKGROUND (continued)

Independent of the above tasks, it is important that operators be informed if the barriers to prevent 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 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 that operators can take mitigating action. Operators should not prematurely circumvent systems important to safety but it is important that they be adequately informed so that planned actions can be taken when necessary.

Examples of serious events that could threaten safety if conditions degrade are loss-of-coolant accidents; 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 period of time, the barriers' ability 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, designed either to withstand the accident environment or to be protected by a local protected environment.

Variables for accident monitoring are selected to provide the essential information necessary for the operator to determine whether 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.

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BASES (continued)

BACKGROUND (continued)

These essential instruments are identified by plant-specific documents (Ref. 1), addressing the recommendations of Regulatory Guide 1.97 (Ref. 2) as required by Supplement 1 to NUREG-0737 (Ref. 3). The instrument channels required to be OPERABLE by this LCO equate to two classes of parameters identified during the implementation of Regulatory Guide 1.97 on a plant-specific basis 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. Primary information is that which 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 3.3.3.1-1 contains no examples of Type A variables, except for those that may also belong to Category 1.

Category 1 variables are the key variables deemed risk significant because they are needed to:

1. Determine whether other systems important to safety are performing their intended functions;

2. 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

3. 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 identify the plant-specific Type A variables and provide justification for deviating from the NRC proposed list of Category 1 variables.

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BASES (continued)

BACKGROUND (continued)

Table 3.3.3.1-1 provides a list of variables typical of those identified by plant-specific Regulatory Guide 1.97 analyses. [Table 3.3.3.1-1 in plant-specific Technical Specifications should 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, immediately accessible display, continuous readout, and recording of display.

Listed below is a discussion of the specified instrument functions listed in Table 3.3.3.1-1. These discussions are intended as examples of what should be provided for each function when the plant-specific bases are prepared.

1. Reactor Steam Dome Pressure

Reactor steam dome pressure is a Category 1 variable provided to support monitoring of RCS integrity and to verify operation of the Emergency Core Cooling System (ECCS). Two independent pressure transmitters with a range of 0 to 1500 psig monitor pressure. Wide-range recorders are the primary indication used by the operator during an accident. Therefore, the PAM specification deals specifically with this portion of the instrument channel.

2. Reactor Vessel Water Level

Reactor vessel water level is a Category 1 variable provided to support monitoring of core cooling and to verify operation of the ECCS. The shroud water level channels provide the PAM reactor vessel water level function. The shroud level channels measure from 17 inches below the dryer skirt down to a point just below the bottom of the active fuel. Shroud water level is measured by two independent differential pressure transmitters. The output from these channels is recorded on two independent pen recorders. These recorders are the primary indication used by the

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BASES (continued)

BACKGROUND (continued)

operator during an accident. Therefore, the PAM specification deals specifically with this portion of the instrument channel.

The shroud water level system is uncompensated for variation in reactor water density and is calibrated to be most accurate at operational pressure and temperature.

3. Suppression Pool Water Level

Suppression pool water level is a Category 1 variable provided to detect a breach in the reactor coolant pressure boundary (RCPB). This variable is also used to verify and provide long-term surveillance of ECCS function. The wide-range suppression pool water level measurement provides the operator with sufficient information to assess the status of the RCPB and to assess the status of the water supply to the ECCS. The wide-range water level indicators monitor the suppression pool level from the center line of the ECCS suction lines to the top of the pool. Two wide-range suppression pool water level signals are transmitted from separate differential pressure transmitters and are continuously recorded on two recorders in the control room. These recorders are the primary indication used by the operator during an accident. Therefore, the PAM specification deals specifically with this portion of the instrument channel.

4. Drywell Pressure

Drywell pressure is a Category 1 variable provided to detect breach of the RCPB and to verify ECCS functions that operate to maintain RCS integrity. Two wide-range drywell pressure signals are transmitted from separate pressure transmitters and are continuously recorded and displayed on two control room recorders. These recorders are the primary indication used by the operator during an accident. Therefore, the PAM specification deals specifically with this portion of the instrument channel. The total response time for

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BASES (continued)

BACKGROUND (continued)

these pressure monitoring systems is on the order of 1.2 seconds, which is adequate to detect and record any significant pressure impulses.

5. Primary Containment Area Radiation (High Range)

Primary containment area radiation 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, primary containment area radiation (high range) PAM instrumentation consists of the following:].

6. Drywell Sump Level

Drywell sump level is a Category 1 variable provided for verification of ECCS functions that operate to maintain RCS integrity. [For this facility, the drywell sump level PAM instrumentation consists of the following:].

7. Drywell Drain Sump Level

Drywell drain sump level is a Category 1 variable provided to detect breach of the RCPB and for verification and long-term surveillance of ECCS functions that operate to maintain RCS integrity. [For this facility, the drywell drain sump level PAM instrumentation consists of the following:].

8. Primary Containment Isolation Valve (PCIV) Position

PCIV position is provided for verification of containment integrity. [For this facility, the PCIV position PAM instrumentation consists of the following:].

9. Wide-Range Neutron Flux

Wide-range neutron flux is a Category 1 variable provided to verify reactor shutdown. [For this facility, the wide-range neutron flux PAM instrumentation consists of the following:].

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BASES (continued)

BACKGROUND (continued)

10., 11. Drywell and Containment Hydrogen and Oxygen Analyzers

Drywell and containment hydrogen and oxygen analyzers are Category 1 instruments provided to detect high hydrogen or oxygen concentration conditions that represent a potential for containment breach. This variable is also important in verifying the adequacy of mitigating actions. [For this facility, the drywell and containment hydrogen and oxygen analyzers PAM instrumentation consists of the following:].

12. Primary Containment Pressure

Primary containment pressure is a Category 1 variable provided to verify RCS and containment integrity and to verify the effectiveness of ECCS actions taken to prevent containment breach. Two wide-range primary containment pressure signals are transmitted from separate pressure transmitters and are continuously recorded and displayed on two control room recorders. These recorders are the primary indication used by the operator during an accident. Therefore, the PAM specification deals specifically with this portion of the instrument channel. The total response time for these pressure monitoring systems is on the order of 1.2 seconds, which is adequate to detect and record any significant pressure impulses.

13. Suppression Pool Water Temperature

Suppression pool water temperature is a Category 1 variable provided to detect a condition that could potentially lead to containment breach, and to verify the effectiveness of ECCS actions taken to prevent containment breach. The suppression pool water temperature monitoring system allows operators to detect trends in suppression pool water temperature in sufficient time to take action to prevent steam quenching vibrations in the suppression pool. Twenty-four temperature sensors are arranged in six groups of four independent and redundant channels, located such that there is a group of sensors within 30 feet line of sight of each relief valve discharge location.

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(continued)

BASES (continued)

BACKGROUND (continued)

Thus, six groups of sensors are sufficient to monitor each relief valve discharge location. Each group of four sensors includes two sensors for normal suppression pool temperature monitoring and two sensors for PAM. The outputs for the PAM sensors are recorded on four independent (channels A and C are redundant to channels B and D, respectively) recorders in the control room. All four of these recorders must be OPERABLE to furnish two channels of PAM indication for each of the relief valve discharge locations. These recorders are the primary indication used by the operator during an accident. Therefore, the PAM specification deals specifically with this portion of the instrument channels.

APPLICABLE SAFETY ANALYSES

The accident monitoring instrumentation LCO ensures the OPERABILITY of Regulatory Guide 1.97 Type A and Category 1 variables so that the control room operating staff can:

1. Perform the diagnosis required to support preplanned actions for the primary success path of Design Basis Accidents (DBAs);
2. Take the specified, preplanned, manually controlled actions for which no automatic control is provided, which are required for safety systems to accomplish their safety function;
3. Determine whether systems important to safety are performing their intended functions;
4. Determine the potential for causing a gross breach of the barriers to radioactivity release;
5. Determine whether a gross breach of a barrier has occurred; and
6. Initiate action necessary to protect the public and for an estimate of the magnitude of any impending threat.

(continued)

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

These functions support the requirements of GDC 13 and GDC 19 of Appendix A to 10 CFR 50 (Ref. 4). The plant-specific Regulatory Guide 1.97 analysis documents the process that identified Type A and Category 1 variables.

Accident monitoring instrumentation that satisfies the definition of Type A in Regulatory Guide 1.97 meets 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 for reducing public risk.

LCO

The PAM instrumentation LCO provides the requirement of Type A and Category 1 monitors, which provide information required by the control room operators to:

1. Permit the operator to take pre-planned manual actions to accomplish safe plant shutdown;
2. Determine whether other systems important to safety are performing their intended functions;
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 and to determine whether 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.

Two channels are required to be OPERABLE for most functions. Two OPERABLE channels ensure no single failure within the PAM instrumentation, its auxiliary supporting features, or its power sources concurrent with the failures that are a condition of, or that result from, a specific accident that prevents the operators from being presented with the information necessary for them to determine the safety

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(continued)

BASES (continued)

LCO (continued)

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 CHANNEL CHECKS during the post-accident phase to confirm the validity of displayed information. More than two channels may be required at some plants if the Regulatory Guide 1.97 analysis determined that failure of one PAM channel results in information ambiguity (i.e., 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 3.3.3.1-1, the exceptions to the two channel requirement are suppression pool water temperature and PCIV position.

The required number of channels for suppression pool water temperature are intended to ensure that each relief valve discharge location is monitored. One set of temperature sensors may monitor more than one relief valve. [For this facility, the relief valve arrangement and the number of sets of sensors and sensor locations are as follows:]. This ensures redundancy of sensor coverage of the relief valves. A typical BWR/4 has 6 sets of 4 groups of redundant temperature sensors to monitor the discharge locations for 22 relief valves. If any of the 12 PAM sensor channels become inoperable, one of the channels required OPERABLE by the LCO is considered inoperable.

In the case of PCIV position, the important information is the status of the containment penetration. The LCO requires one position indicator for each active PCIV. This is sufficient to redundantly verify the isolation status of each isolable penetration via indicated status of the active valve and prior knowledge of passive valve or system boundary status. If a normally active PCIV is known to be closed and deactivated, position indication is not needed to determine status. Therefore, the position indication for valves in this state is not required to be OPERABLE.

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BASES (continued)

LCO (continued)

Each PAM required channel is OPERABLE when:

1. All channel components necessary to provide the required indication are functional;

2. 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; and

3. Required surveillance testing is current and has demonstrated performance within each surveillance test's acceptance criteria.

LCO Table 3.3.3.1-1 is for illustration purposes only. In plant-specific Technical Specifications, Table 3.3.3.1-1 will list all Type A and Category 1 variables identified by the plant's Regulatory Guide 1.97 analysis, as amended by the 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 result in the PAM instrumentation being declared inoperable and their justification are as follows:]

APPLICABILITY

The PAM instrumentation LCO is applicable in MODES 1 and 2. These variables are related to the diagnosis and pre-planned actions required to mitigate DBEs. The applicable DBEs are assumed to occur in MODES 1 and 2. In MODES 3, 4, and 5, plant conditions are such that the likelihood of an event occurring that would require PAM instrumentation is extremely low; therefore, 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. A second Note provides clarification that each function specified in Table 3.3.3.1-1 shall be treated as an independent entity for this LCO with an independent Completion Time.

ACTIONS

A channel is inoperable when it does not satisfy the OPERABILITY criteria for the channel's function. These critaria are outlined for the functions in the LCO section of the bases.

Condition A

When one required channel in one or 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 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 that 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 must 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 during this interval and the availability of alternate means of obtaining the required information. Continuous operation with two required channels inoperable is not acceptable because the alternate indications may not fully meet all performance qualification requirements applied to the PAM instrumentation. Therefore, requiring restoration of at

(continued)

(continued)

BASES (continued)

ACTIONS (continued)

least one inoperable channel limits the risk that the PAM function will be in a degraded condition should an accident occur.

Condition C

If the Required Actions and associated Completion Times of Condition A or B are not met, Required Action C.1 directs the operator to follow immediately the directions for either Condition D or E, as given in Table 3.3.3.1-1.

Condition D

For the majority of functions in Table 3.3.3.1-1, if the Required Actions and associated Completion Times of Condition A or B are not met, 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 12 hours. The 12-hour Completion Time is reasonable, based on operating experience, to reach the required MODE from full power in an orderly manner and without challenging plant systems.

Condition E

At this facility alternate means of monitoring [primary containment 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 Technical Specification 5.9.2.c, Special Reports, in the Administrative Controls sections of the Technical Specifications. The report provided to the NRC must 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:]

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

These SRs apply to each PAM instrumentation function in Table 3.3.3.1-1.

SR 3.3.3.1.1

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 against 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. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION. 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 surveillances are 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 interval is based on operating experience that demonstrates the rarity of channel failure. The CHANNEL CHECK supplements less formal, but more frequent checks of channel OPERABILITY during normal operational use of the displays associated with the LCO required channels.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.3.1.2

A CHANNEL CALIBRATION is performed every 18 months, coincident with refueling outage. This test is a complete check of the process control instrument loop and the transmitter. 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. Transmitter "as found" and "as left" values are recorded and used to verify drift assumptions. Field transmitters may be calibrated in place or on a bench, using deadweight or hydraulic/pneumatic test equipment, or be replaced by an equivalent laboratory-calibrated unit.

Resistance temperature detector (RTD) and thermocouples (T/C) 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 each refueling outage to ensure accurate sensor cross-calibrations. This replacement sensor should be the same model as the remaining RTDs or T/Cs. Using a newly calibrated sensor as a reference assures that signal drift continues to remain random rather than systematic and is within the limits specified. The replacement interval may be extended to alternate refueling outages if it is demonstrated that over the extended interval, sensors drift is random rather than systematic. This determination may use results of statistical analyses of operating data and calibration data from similar plants using the same model of RTD or T/C under 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 which cannot be corrected by recalibration.

For the primary containment area radiation instrumentation, a CHANNEL CALIBRATION may consist of an electronic calibration of the channel, not including the detector, for

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(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)	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.

REFERENCES

1. [Plant-specific documents (e.g., FSAR, NRC RG 1.97, SER letter).]

2. Regulatory Guide 1.97, "Instrumentation for Light-Water Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," U.S. Nuclear Regulatory Commission, [Date].

3. NUREG-0737, Supplement 1, "Clarification of TMI Action Plan Requirements," U.S. Nuclear Regulatory Commission, [Date].

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 3.3.3.2 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 a location other than the control room. This capability is necessary to protect against the possibility of the control room becoming inaccessible or of a fire destroying all equipment in one fire area thereby disabling critical control room instruments or controls. A safe shutdown condition is defined as MODE 3. With the facility in MODE 3, the Reactor Core Isolation Cooling (RCIC) System the safety relief valves, and the Residual Heat Removal (RHR) System—Shutdown Cooling can be used to remove core decay heat and meet all safety requirements. The long-term supply of water for the RCIC and the ability to operate shutdown cooling 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 the 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.

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BASES (continued)

APPLICABLE SAFETY ANALYSES

The Remote Shutdown System is required to provide equipment at appropriate locations outside the control room with a design capability to promptly shut down the reactor to MODE 3, including the necessary instrumentation and controls to maintain the unit in a safe condition in MODE 3.

Furthermore, 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) in the event of a fire in any one fire area. The criteria governing the design of the Remote Shutdown System are 10 CFR 50, Appendix A, GDC 19, and 10 CFR 50, Appendix R. Specific system requirements are presented in Reference 1 and the plant-specific fire protection topical report.

The Remote Shutdown System is considered an important contributor to the reduction of plant risk of 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 3.3.3.2-1, following this specification. For Remote Shutdown System channels that support only the functions required by 10 CFR 50, Appendix R, one division is required to be OPERABLE.

For channels that fulfill GDC 19 requirements, the number of OPERABLE channels required depends upon the plant's licensing basis as described in the NRC plant-specific Safety Evaluation Report (SER). Generally, two divisions are required OPERABLE. However, only one channel per given function is required if the plant has justified such a design, and NRC's SER accepted the justification. The controls, instrumentation, and transfer switches are those required for:

- Core reactivity control (initial and long term);
- Reactor pressure vessel pressure control;

(continued)

(continued)

BASES (continued)

LCO (continued)

- Decay heat removal;
- Reactor pressure vessel inventory control; 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 remote shutdown function are OPERABLE in that division. In some cases, Table B 3.3.3.2-1 may indicate that the required information or control capability is available from several alternate sources. In these cases, the division is OPERABLE as long as one channel of any of the alternate information or control sources for each function is OPERABLE.

Remote Shutdown System instrumentation channels are OPERABLE when:

- All channel components necessary to provide the required information or control 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.

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BASES (continued)

LCO (continued)

The Remote Shutdown System 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 plant conditions require that the Remote Shutdown System be placed in operation.

For this facility, a Remote Shutdown System division is considered OPERABLE when all the plant-specific instrumentation, controls, transfer switches, and support systems listed in Table B 3.3.3.2-1 are OPERABLE.

[For the facility, the following support systems are required OPERABLE to ensure Remote Shutdown System OPERABILITY:]

[For this facility, those required support systems which, upon their failure, do not result in the Remote Shutdown System being declared 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 or 5. In these MODES, the facility is already subcritical and in condition of reduced Reactor Coolant System energy. Under these conditions considerable time is available to restore necessary instrument control functions if control room instruments or control become unavailable. Consequently, the Technical Specifications do not require OPERABILITY in MODE 4 or 5.

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 Required Action and Completion Time is equally applicable to startup conditions as it is 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

A Remote Shutdown System division is inoperable when each function listed in Table B 3.3.3.2-1 is not accomplished by at least one designated Remote Shutdown System channel that satisfies the OPERABILITY criteria for the channel's function. These criteria are outlined in the LCO section of the bases.

Condition A

Condition A addresses the situation where one division of any required function(s) or channel(s) of the Remote Shutdown System is inoperable. This includes any function listed in Table B 3.3.3.2-1 as well as the control and transfer switches.

When a Division includes a function that only requires one 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 divisions to OPERABLE status within 30 days. The Completion Time is based on operating experience and the low probability of an event that would require evacuation of the control room.

A Note has been added to clarify that for this LCO, each [division] is treated as an independent entity with an independent Completion Time.

Condition B

If the inoperable function cannot be restored to OPERABLE status within 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 12 hours and in MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required MODES from full power in an orderly manner and without challenging plant systems.

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SURVEILLANCE REQUIREMENTS

The following SRs apply to each Remote Shutdown System division.

SR 3.3.3.2.1

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 against 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 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 Surveillances are 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 rare.

SR 3.3.3.2.2

SR 3.3.3.2.2 verifies that each SR 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,

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

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 is prudent that these Surveillances only be performed during a facility outage. This is due to the plant conditions needed to perform the Surveillance and the potential for unplanned transients if the Surveillance is performed with the reactor at power. Operating experience demonstrates that Remote Shutdown System control channels usually pass the Surveillance when performed at the 18-month Frequency.

SR 3.3.3.2.3

CHANNEL CALIBRATION is a complete check of the instrument channel including the detector. The test verifies that the channel responds to measured parameter values 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 that 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 outages if it is demonstrated that over the extended interval, the RTD's 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.

(continued)

BASES (continued)

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix R, and Appendix A, General Design Criterion 19.

Table B 3.3.3.2-1 (page 1 of 1)
Remote Shutdown System Instrumentation

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. Source Range Neutron Flux		[1]
2. Reactor Pressure Vessel Pressure		
a. Reactor Pressure		[1]
3. Decay Heat Removal		
a. RCIC Flow		[1]
b. RCIC Controls		[1]
c. RHR Flow		[1]
d. RHR Controls		[1]
4. Reactor Pressure Vessel Inventory Control		
a. RCIC Flow		[1]
b. RCIC Controls		[1]
c. RHR Flow		[1]
d. RHR Controls		[1]

Table B 3.3.3.2-1 (page 1 of 1)
Remote Shutdown System Instrumentation

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.

FUNCTION/INSTRUMENT OR CONTROL PARAMETER	LOCATION	REQUIRED NUMBER OF DIVISIONS
1. Reactivity Control		
a. Source Range Neutron Flux		[1]
2. Reactor Pressure Vessel Pressure		
a. Reactor Pressure		[1]
3. Decay Heat Removal		
a. RCIC Flow		[1]
b. RCIC Controls		[1]
c. RHR Flow		[1]
d. RHR Controls		[1]
4. Reactor Pressure Vessel Inventory Control		
a. RCIC Flow		[1]
b. RCIC Controls		[1]
c. RHR Flow		[1]
d. RHR Controls		[1]

B 3.3 INSTRUMENTATION

B 3.3.4.1 End-of-Cycle Recirculation Pump Trip (EOC-RPT) Instrumentation

BASES

BACKGROUND

The EOC-RPT instrumentation initiates a recirculation pump trip (RPT) to reduce the peak reactor pressure and power resulting from turbine trip or generator load rejection transients coincident with failure of the Turbine Bypass System, to provide additional margin to core thermal Safety Limits (SLs) (MINIMUM CRITICAL POWER RATIO (MCPR)).

To mitigate pressurization transient effects, the EOC-RPT must trip the recirculation pumps within approximately 175 ms after initiation of closure movement of either the turbine stop valves (TSVs) or the turbine control valves (TCVs). The combined effects of this trip and a scram reduce fuel bundle power more rapidly than a scram alone, resulting in an increased margin to the MCPR SL. The EOC-RPT function is automatically disabled when turbine first-stage pressure is < [40%] RATED THERMAL POWER (RTP).

The need for the additional negative reactivity in excess of that normally inserted on a scram reflects end-of-cycle reactivity considerations. Flux shapes at the end of cycle are such that the control rods may not be able to ensure that thermal limits are maintained by inserting sufficient negative reactivity during the first few feet of rod travel upon a scram caused by TCV Fast Closure or TSV Closure.

The EOC-RPT instrumentation is comprised of sensors that detect initiation of closure of the TSVs or fast closure of the TCVs combined with relays, logic circuits, and fast-acting circuit breakers that interrupt power from the recirculation pump MG-set generators to each of the recirculation pump motors. When the RPT breakers trip open, the recirculation pumps coast down under their own inertia. The EOC-RPT has two identical trip systems which can actuate an RPT based on a one-out-of-two trip by system configuration. Each EOC-RPT trip system is a two-out-of-two logic for each function, thus either two TSV or two TCV signals are required for a trip system to actuate. If either trip system actuates, both recirculation pumps will trip. There are two EOC-RPT breakers in series per

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BASES (continued)

BACKGROUND (continued)	recirculation pump; each trip system operates independent breakers in the supply circuits of both recirculation pump motors. See the EOC-RPT Logic Diagram, Figure B 3.3.4.1-1.

APPLICABLE SAFETY ANALYSES

The TSV Closure and the TCV Fast Closure functions are designed to trip the recirculation pumps in the event of a turbine trip or generator load rejection to mitigate the pressurization transient and to increase the margin to the MCPR SL. The analytical methods and assumptions used in evaluating the turbine trip and generator load rejection without bypass are summarized in Reference 1.

TSV Closure or TCV Fast Closure and a main turbine trip result in reactor pressure, neutron flux, and heat flux transients that must be limited. Therefore, a recirculation pump trip is initiated at the start of TSV Closure and a TCV Fast Closure in anticipation of the transients that would result from the closure of these valves. The EOC-RPT decreases reactor power and aids the reactor scram in ensuring that reactor pressure and MCPR SLs are not exceeded during the worst-case transient.

EOC-RPT instrumentation satisfies Criteria 1, 2, and 3 of the NRC Interim Policy Statement.

LCO

The LCO requires [two] channels per trip system to be OPERABLE for each EOC-RPT function. The OPERABILITY of the individual instrumentation functions is confirmed through successful completion of required Surveillance testing; i.e., CHANNEL CHECKS, CHANNEL FUNCTIONAL TESTS, LOGIC SYSTEM FUNCTIONAL TESTS, EOC-RPT RESPONSE TIME TESTS and CHANNEL CALIBRATIONS. Individual measurement channels and the associated bistable trip units are considered operable when the following conditions are satisfied:

1. All channel components necessary to provide a trip signal are functional and in service;

2. Channel measurement uncertainties are know via test, analysis, or design information to be within the assumptions of the setpoint calculations;

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(continued)

BASES (continued)

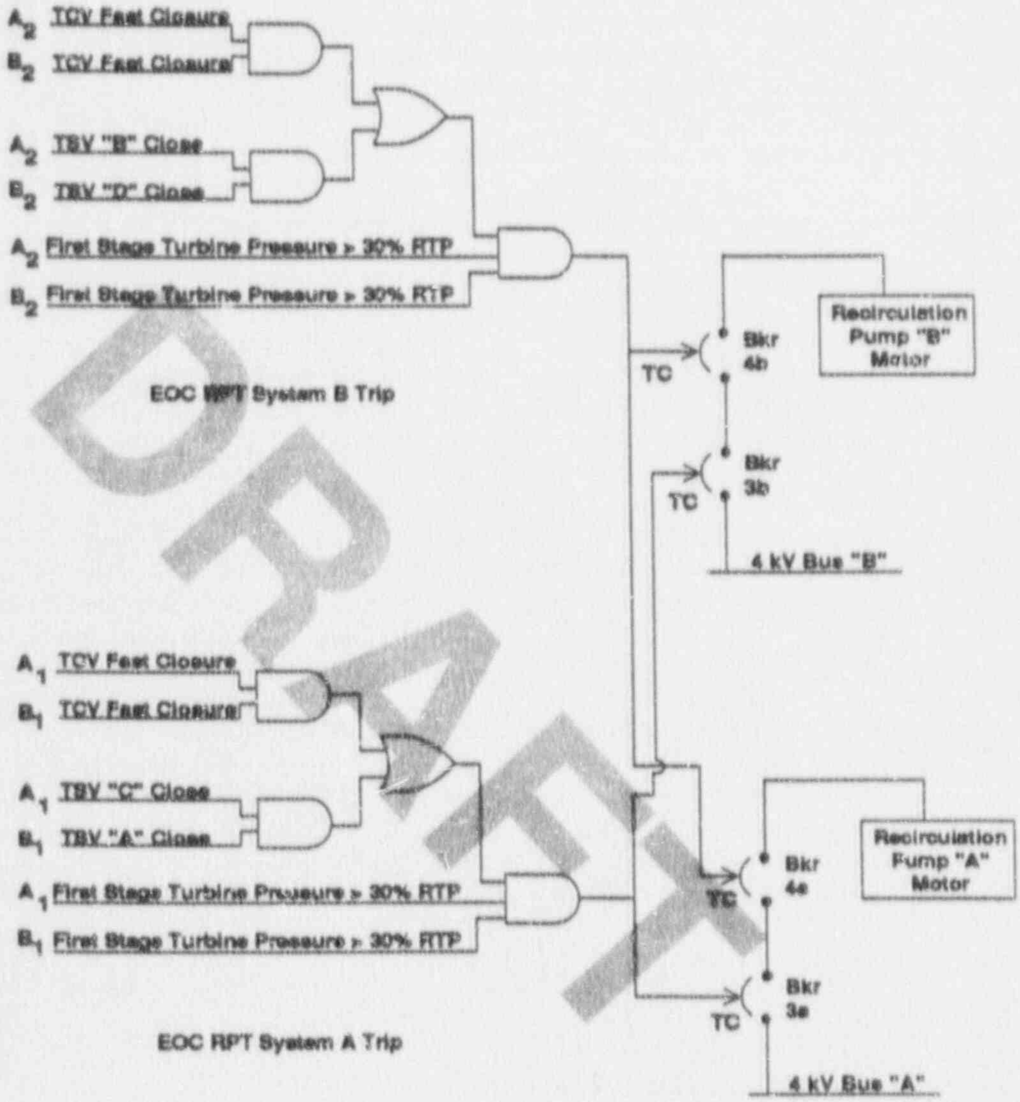

Figure B 3.3.4.1-1
End-of-Cycle Recirculation Pump Trip Logic Diagram
Typical Figure

1. Either both TCV Fast Closure signals must be present or both TSV Closure signals must be present in one trip system for trip of the recirculation pumps to occur. The RPT will not occur if one TCV and one TSV signal are present in a trip system, nor will it occur if one TCV (or TSV) signal is present in both trip systems.

2. Because the two first-stage turbine pressure signals are combined into an "and" logic configuration in each trip system, loss of either will not permit the RPT to occur. For the RPT to occur, both A_1 and B_1 must be present for Trip System A, and likewise A_2 and B_2 must be present for Trip System B.

(continued)

BASES (continued)

LCO (continued)

3. Required surveillance testing is current and has demonstrated performance within each surveillance test's acceptance criteria; and

4. The associated operational bypass is not enabled except under the conditions specified by the LCO Applicability statement for the function.

The ALLOWABLE VALUES are specified for each EOC-RPT function in the LCO. Nominal trip setpoints are specified in the plant-specific setpoint calculations. The nominal setpoints are selected to ensure the setpoint measurements 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 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 (Ref. 2). Each channel must also respond within its assumed response time. At THERMAL POWER < [30%] RTP core reactivity is assumed to be low enough that the EOC-RPT is not required to be OPERABLE and the trip functions are automatically bypassed when turbine first stage pressure is ≤ [30%].

The EOC-RPT has inputs to the trip logic from the TSV Closure and TCV Fast Closure functions. These functions are discussed below.

<u>TSV Closure</u>

Closure of the TSVs is determined by measuring the position of each valve. There are two separate position switches associated with each stop valve, the signal from each switch being assigned to a separate trip channel. The logic for the TSV Closure function is such that two or more TSVs must be closed to produce an EOC-RPT. Four channels of TSV Closure, with two channels in each trip system, are available and required to be OPERABLE to ensure that no single failure will preclude an EOC-RPT from this function on a valid signal. The TSV Closure ALLOWABLE VALUE is

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BASES (continued)

LCO (continued)

selected to detect imminent TSV Closure, but low enough so as to not generate spurious trips due to normal system-pressure fluctuations.

This protection is required, consistent with the safety analysis assumptions, whenever the THERMAL POWER is ≥ 30% RTP. There is an automatic bypass of this trip function below the turbine first-stage pressure value equivalent to THERMAL POWER < [30%] RTP, since the Reactor Vessel Steam Dome Pressure--High and the Neutron Flux--High Functions of the Reactor Protection System (RPS) are adequate to maintain the necessary safety margins.

This automatic bypass setpoint is feedwater temperature dependent as a result of the subcooling changes that affect the turbine first-stage pressure/reactor power relationship. For operation with feedwater temperature ≥ 420°F, an ALLOWABLE VALUE setpoint of ≤ 26.3% of control valve wide-open turbine first-stage pressure is provided by the bypass function. The ALLOWABLE VALUE setpoint is reduced to ≤ 22.5% of control valve wide-open turbine first-stage pressure for operation with a feedwater temperature between 370°F and 420°F.

TCV Fast Closure

Fast closure of the TCVs is determined by measuring the EHC fluid pressure at each control valve. There is one pressure transmitter associated with each control valve, and the signal from each transmitter is assigned to a separate trip channel. Four channels of TCV Fast Closure, with two channels in each trip system, are available and required to be OPERABLE to ensure that no single failure will preclude an EOC-RPT from this function on a valid signal. The TCV Fast Closure ALLOWABLE VALUE is selected high enough to detect imminent TCV Fast Closure, but low enough not to generate spurious trips due to normal system-pressure fluctuations.

This protection is required consistent with the safety analysis whenever the THERMAL POWER is ≥ [30%] RTP. As with TSV Closure, this function is also bypassed below [30%] RTP. The basis for the setpoint of this automatic bypass is identical to that described for TSV Closure.

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BASES (continued)

LCO (continued)

[For this facility, the following support systems are required OPERABLE to ensure EOC-RPT Instrumentation OPERABILITY:]

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

[For this facility, the supported systems impacted by the inoperability of the EOC-RPT instrumentation and the justification for each being declared inoperable are as follows:]

APPLICABILITY

The TSV Closure function and TCV Fast Closure function are required to be OPERABLE whenever THERMAL POWER is ≥ [30%] RTP, consistent with assumptions in the safety analyses. Below [30]% RTP the Reactor Vessel Steam Dome Pressure--High and high neutron flux functions of the RPS are adequate to maintain safety margins.

The bases for Applicability requirements are discussed on a function-by-function basis in the LCO section.

ACTIONS

A protection function channel is inoperable when it does not satisfy the OPERABILITY criteria for the channel's function. These criteria are outlined for the functions in the LCO section of the bases.

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 bistable 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.

The Conditions and Required Actions apply to each one of the EOC-RPT instrumentation functions. A Note has been added to indicate that the Completion Time is on a Condition basis for each function.

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BASES (continued)

ACTIONS (continued)

Condition A

Since one or more channels per function may be inoperable at a given time, it is necessary, in order to avoid loss of EOC-RPT instrumentation function, to maintain each channel in at least one trip system either OPERABLE or in trip. As an example, referring to Fig. B 3.3.4.1-1, EOC-RPT Logic Diagram, if both TCV channels A_2 and B_2 are faulty in trip system B and neither one is placed in trip, the system can still function as long as all channels of trip system A are either OPERABLE or one channel is in trip and the other is OPERABLE. Depending on the configuration of failed and tripped channels, the system may or may not be capable of withstanding an additional single failure. In the example above, another failure of either the A_1 or B_1 TSV channel will disable the EOC-RPT function. Several cases of failure, however, are tolerable without losing single-failure tolerance. In fact, up to six channel failures may be acceptable under the single-failure criterion. For instance, all of the A trip system functions, such as TCV A_1 and B_1, TSV A_1 and B_1, and both channels of first-stage turbine pressure > [30%] RTP is acceptable. The time allowed for this action (Required Action A.1) is 1 hour which is sufficient for operations personnel to ensure functional capability. In addition, Required Action A.2.1 or Required Action A.2.2 allows the facility 72 hours in which to restore the inoperable channels or place them in the trip position, thus assuring that the system's ability to withstand a single failure is restored. The 72-hour Completion Time is sufficient for operations personnel to complete the corrective actions of Required Action A.2.1 or Required Action A.2.2. In Required Action A.2.2 a Note has been added to indicate that the Required Action applies only if placing inoperable channel(s) in trip would not result in a recirculation pump trip or scram.

Condition B

If Required Action A.1 and Required Action A.2.1, or Required Action A.2.2 are not carried out within the allotted time, Required Action B.1 or Required Action B.2 is required. Required Action B.1 allows 6 hours to place the plant within the calculated MCPR limit. Required Action B.2

(continued)

(continued)

BASES (continued)

ACTIONS (continued)

specifies that the plant must reduce THERMAL POWER to < [30%] RTP within 6 hours. The plant has a choice as to which of these actions is appropriate for their operation. The Completion Time of 6 hours is a reasonable time, based on operating experience, to make corrective actions in an orderly manner and without challenging plant systems.

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 trip system(s). The specified Completion Time of 1 hour is sufficient for plant operations personnel to make this verification.

Required Action C.1 ensures that those identified Required Actions associated with supported systems impacted by the inoperability of trip system(s) associated with each EOC-RPT 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 EOC-RPT function are as follows:]

Required Action C.2 verifies that all required support or supported features associated with the other trip system are determined to be OPERABLE within 1 hour. The Completion Time of 1 hour is sufficient for plant operations personnel to make this verification. If verification determines loss of functional capability determined to be LCO 3.0.3 must be immediately entered. However, if the support or supported feature LCOs take into consideration the loss-of-function situation then LCO 3.0.3 may not need to be entered.

SURVEILLANCE REQUIREMENTS

The SRs for each function are found in the SR section.

A Note has been added to the SRs for LCO 3.3.4.1 to allow placing one trip system for a single function in an inoperable status for up to 2 hours for performance of required surveillances as long as the unaffected trip system is OPERABLE. Upon completion of the Surveillance or expiration of the 2-hour allowance, the trip system must be

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

returned to OPERABLE status or the Required Actions taken. [For this facility, the 2-hour allowance is based on the following:]

SR 3.3.4.1.1

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 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 must be found within the ALLOWABLE VALUES 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 which cannot be corrected by recalibration.

This SR is modified by a Note that the RPT breakers are not required to be tripped during this test. This is due to the plant condition needed to perform the surveillance and the potential for unplanned plant transients if the surveillance is performed with the reactor at power.

[For this facility, the Surveillance interval is based upon the following:]

SR 3.3.4.1.2

Calibration of trip units consists of a test to determine actual trip setpoints. If, during trip unit calibration, the association trip setting is discovered to be less conservative than the specified ALLOWABLE VALUE, the channel must be declared inoperable under these conditions, the setpoint must be readjusted to be equal to or more conservative than accounted for in the appropriate setpoint methodology.

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SURVEILLANCE REQUIREMENTS (continued)

The Surveillance Frequency of 31 days is based on assumptions in the methodology included in the determination of the trip setpoint. SR 3.3.4.1.1 and SR 3.3.4.1.2 are often performed simultaneously using a common procedure.

SR 3.3.4.1.3

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 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 3.3.4.1.4

Performance of a LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required actuation logic for a specific channel. The LOGIC SYSTEM FUNCTIONAL TEST tests

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

all logic components (e.g., all relays and contacts, trip units, and solid-state logic elements) of a logic circuit, from sensor up to the actuated device. The system functional test of the pump breakers is included as a part of this test, overlapping the LOGIC SYSTEM FUNCTIONAL TEST to provide complete testing of the associated safety function. Therefore, if the breaker is incapable of operating, the associated instrument channel would also be inoperable.

The 18-month Surveillance Frequency was developed considering it was prudent that the surveillance 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.

SR 3.3.4.1.5

This SR ensures that scrams initiated from the TSV Closure and TCV Fast Closure functions will not be inadvertently bypassed when THERMAL POWER is < [30%] RTP. This involves calibration of the bypass channels and the incorporation of adequate margins for the instrument setpoint methodologies in the actual setpoint. If any bypass channel's setpoint is nonconservative (the functions are bypassed at > [30%] RTP), then either the affected TSV Closure and TCV Fast Closure functions must be declared inoperable and the appropriate Required Actions taken or, alternatively, the bypass channel must be placed in the conservative condition (non-bypass). If placed in the non-bypass condition, this SR is met and no additional actions are required.

[At this facility, the 18-month Surveillance Frequency is as follows:]

SR 3.3.4.1.6

This Surveillance Requirement 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

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

limit at the sensor to the point of recirculation pump power interruption. The acceptable response times of the relevant trip channels are given below. [For this facility, the response times include contributions from the following:] 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 breaker arc suppression time may be assumed from the most recent performance of SR 3.3.4.1.7.

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 x 18 months, where n is the number of channels per trip system in the function. Testing of the final actuation devices in a trip system, 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 which cause serious response time degradation, but not channel failure, are infrequent occurrences. Response times cannot be determined at power since equipment operation is required.

SR 3.3.4.1.7

This SR ensures that the RPT breaker arc suppression time is provided to the EOC-RPT response time test. The 60-month Frequency of the testing is based on the difficulty of performing the test and the reliability of the circuit breakers.

REFERENCES

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

2. [Unit Name] FSAR, "[Plant-specific Setpoint Methodology.]"

B 3.3 INSTRUMENTATION

B 3.3.4.2 Anticipated Transients Without Scram—Recirculation Pump Trip (ATWS-RPT) Instrumentation

BASES

BACKGROUND

The ATWS-RPT System initiates a recirculation pump trip following events in which a scram does not, but should occur. To lessen the effects of an ATWS event, negative reactivity must be added to the reactor core by another means. The means chosen is to trip the recirculation pumps. Tripping the recirculation pumps adds negative reactivity from the increase in steam voiding in the core area as core flow decreases. When reactor pressure reaches [1120] psig, or a reactor water level of [-55] inches, the recirculation pump drive-motor breakers trip. The trip setpoints used in the bistable are based on the analytical limits stated in Reference 1.

The ATWS-RPT System includes sensors, relays, bypass circuits, circuit breakers, and switches that are necessary to cause initiation of RPT. The input parameters to the ATWS-RPT logic are electrical signals that indicate limits on low reactor vessel water level and high reactor pressure.

The ATWS-RPT is made up of two independent trip systems, with two channels of Reactor Steam Dome Pressure--High and two channels of Reactor Vessel Water Level--Low Low, Level 2 in each trip system. Each ATWS-RPT trip system is a two-out-of-two logic for each function. Thus, either two Reactor Water Level--Low Low, Level 2 or two Reactor Pressure--High signals are needed to trip a trip system. For most ATWS-RPT designs, two independent trip systems are necessary to satisfy the reliability requirements of 10 CFR 50.62 (Ref. 2). The outputs of the channels in a trip system are combined in a logic so that either trip system will trip both recirculation pumps. [For this facility, the basis for the trip system logic is as follows:]

There is one drive motor breaker provided for each of the two recirculation pumps for a total of two breakers. The output of one trip system is provided to both recirculation pump breakers (see Figure B 3.3.4.2-1, "Anticipated Transients Without Scram-Recirculation Pump Trip Logic Diagram.")

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BASES (continued)

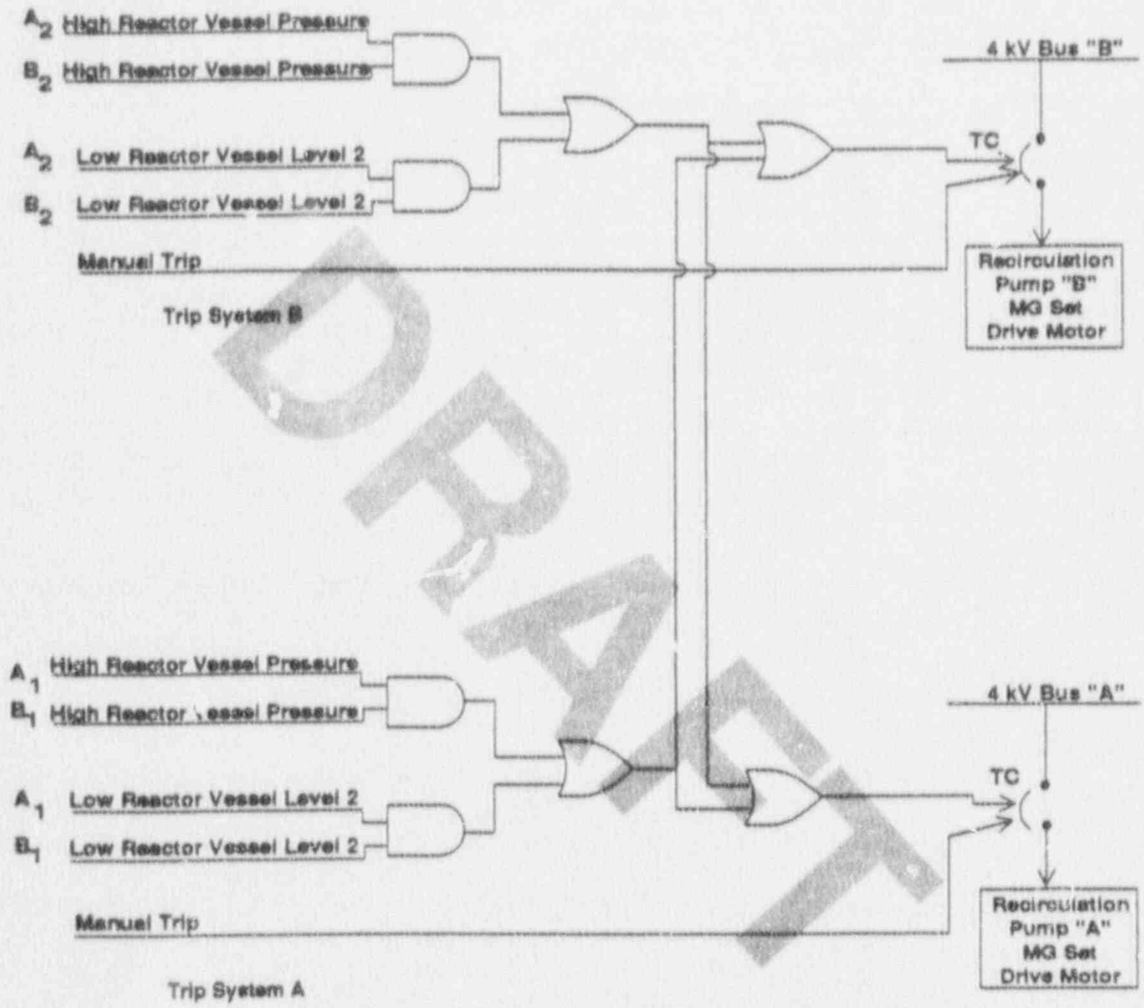

Figure B 3.3.4.2-1
Anticipated Transients Without Scram-Recirculation Pump Trip Logic Diagram
Typical Figure

Notes:

1. Both Reactor Vessel Pressure--High channels or both Reactor Vessel Water Level--Low Low, Level 2 channels in either trip system must trip to ensure trip of the recirculation pumps.

2. Trip of either trip system will trip both recirculation pumps.

3. Pump trip is accomplished by tripping the motor-generator (MG)-set drive-motor breakers or field breakers.

4. Manual trip is accomplished by remote manual trip of MG-set drive-motor breakers.

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BASES (continued)

APPLICABLE SAFETY ANALYSES

The ATWS-RPT System is required by 10 CFR 50.62 (Ref. 2) and is designed to provide diverse means for reactor shutdown in the event of an anticipated operational occurrence (AOO) followed by a failure of the Reactor Protection System (RPS). The system is not required to meet the single-failure criterion, but is required to perform its function in a reliable manner.

The ATWS-RPT protects against potential common-mode failures of the RPS. The ATWS-RPT initiates an RPT to aid in the reduction of reactor vessel pressure and to ensure fuel coverage following events during which scram has not occurred. Low reactor vessel water level indicates that the capability of cooling the fuel may be threatened. Should reactor vessel water level decrease too far, fuel damage could result. Therefore, the ATWS-RPT is initiated at level 2 to aid in maintaining the water level above the top of the active fuel. The RPT at level 2 is provided to decrease positive reactivity that occurs as a result of compression of the steam voids. The reduction of core flow reduces neutron flux and THERMAL POWER and, therefore, the rate of coolant boiloff.

The Reactor Vessel Steam Dome Pressure--High function initiates an RPT to counteract, by rapidly reducing core power generation, transients that cause a pressure increase. For the overpressurization event, the RPT aids in the termination of the ATWS event and, along with the safety/relief valves (S/RVs), limits the peak reactor vessel pressure to less than the [1500 psig] American Society of Mechanical Engineers (ASME) Code limit.

ATWS-RPT instrumentation satisfies the risk mitigation requirements of the NRC Interim Policy Statement.

LCO

The ATWS-RPT LCO requires all instrumentation performing an ATWS-RPT function to be OPERABLE, which ensures that a diverse reactor shutdown system is available in the event of an AOO and common-mode failure of the RPS.

The OPERABILITY of the ATWS-RPT is dependent on the OPERABILITY of the inputs to the trip logic from the instrumentation channels of the Reactor Steam Dome Pressure--High function and the Reactor Vessel Water

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LCO (continued)

Level--Low Low, Level 2 function. For its two-out-of-two logic configuration, each function must have all channels OPERABLE in both trip systems. Either trip system alone will perform the ATWS-RPT protective function.

Only the ALLOWABLE VALUES are specified for each function in the LCO. Nominal trip setpoints are specified in plant-specific setpoint calculations. The nominal setpoints are selected to ensure that 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 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.

A channel is OPERABLE when the following conditions are satisfied:

1. All channel components necessary to generate a trip signal are functional and in service;

2. Channel measurement uncertainties are known—via test, analysis, or design information—to be within the assumptions of the setpoint calculations; and

3. Required surveillance testing is current and has demonstrated performance within each surveillance test's acceptance criteria.

Each ATWS-RPT function is discussed below.

1. <u>Reactor Vessel Water Level--Low Low, Level 2</u>

Reactor vessel water level signals are initiated from four level transmitters that are arranged in pairs on separate taps (two transmitters per tap) that come from two physically separated locations.

Instrument lines are routed from each location through the drywell and terminate at the transmitters located

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BASES (continued)

LCO (continued)

in the containment. Four channels of Reactor Vessel Water Level--Low Low, Level 2, with two channels in each trip system, are connected to relays whose contacts are arranged in a two-out-of-two logic configuration. Both channels in each trip system are required to be OPERABLE. The Reactor Vessel Water Level--Low Low, Level 2 ALLOWABLE VALUE is chosen so that the system will be initiated after a Level 3 scram. [For this facility, the basis for the ALLOWABLE VALUE is as follows:]

2. Reactor Steam Dome Pressure--High

Excessively high reactor pressure may potentially rupture the reactor coolant pressure boundary (RCPB). An increase in the reactor pressure during reactor operation compresses the steam voids and results in a positive reactivity insertion. This increases neutron flux and THERMAL POWER, which could potentially result in fuel failure and reactor pressure vessel overpressurization.

Reactor pressure is measured at two physically separated locations. An instrument line is routed from each location through the drywell and terminates in the containment, where two pressure transmitters monitor the pressure from each tap. Four channels of Reactor Steam Dome Pressure--High, with two channels in each trip system, are connected to relays whose contacts are arranged in a two-out-of-two logic configuration. Both channels in each trip system are required to be OPERABLE. The Reactor Steam Dome Pressure--High ALLOWABLE VALUE is chosen above the maximum normal operating pressure to permit normal operation without spurious RPT and yet provide an adequate margin to the ASME Code allowable Reactor Coolant System (RCS) pressure.

[For this facility, the following support systems are required to be OPERABLE to ensure ATWS-RPT instrumentation OPERABILITY:]

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

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(continued)

BASES (continued)

LCO (continued)

[For this facility, the supported systems impacted by the inoperability of the ATWS-RPT instrumentation and the justification of whether or not each supported system is declared inoperable are as follows:]

APPLICABILITY

The individual functions are required to be OPERABLE in MODE 1 to protect against common-mode failures of the RPS by providing a diverse trip to mitigate the consequences of a postulated ATWS event. The Reactor Steam Dome Pressure--High and Reactor Water Level--Low Low, Level 2 functions are required to be OPERABLE in MODE 1 when the RCS is pressurized and the potential for a pressure transient exists. In MODE 2, the recirculation system is at low flow, so the RPT is not necessary. In MODES 3 and 4, the reactor is shut down with all control rods inserted, and the possibility of a significant pressure increase is negligible. In MODE 5, the reactor head is not fully tensioned and no pressure-transient threat to the RCPB exists.

ACTIONS

A protection function 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 to within specification. If the trip setpoint is less conservative than the ALLOWABLE VALUE in SR 3.3.4.2.4, the channel must be declared inoperable immediately, and the appropriate Conditions must be entered immediately.

In the event a channel's trip setpoint is found nonconservative with respect to the ALLOWABLE VALUE, or the transmitter, instrument loop, signal processing electronics, or bistable is found inoperable, then all affected functions

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BASES (continued)

ACTIONS (continued)

provided by that channel must be declared inoperable and the LCO Condition entered for the particular protection function affected.

The Conditions of these Actions apply to each one of the required ATWS-RPT functions. A Note has been added to indicate that the Completion Time is on a Condition basis for each function.

Condition A

Condition A applies to both of the ATWS-RPT functions.

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

Since one or more channels per function may be inoperable at a given time, it is necessary to maintain each channel in at least one trip system OPERABLE or tripped in order to preserve the ATWS-RPT function. If at least one trip system is not at all times capable of initiating a trip, then the reactor must be placed in a mode outside the Applicability.

As an example, referring to Figure B 3.3.4.2-1, ATWS-RPT Logic Diagram, if both channels A_1 and B_1 are faulty in trip system A and neither one placed in trip, the system can still function, as long as all channels of trip system B are either operable or one channel is in trip and the other OPERABLE.

The 1-hour Completion Time allowed for Required Action A.1 is sufficient for operations personnel to ensure functional capability. In addition, Required Action A.2.1 or Required Action A.2.2 allows 72 hours in which to repair faulty channels or place them in tripped condition, thus assuring that the system's ability to withstand a single failure is restored. The 72-hour Completion Time is sufficient for operations personnel to complete either Required Action A.2.1 or Required Action A.2.2. Required Action A.2.2 has been modified by a Note to indicate that the Required Action only applies if placing inoperable channel(s) in trip would not result in a recirculation pump trip. If a channel becomes inoperable for a function in one trip system when its redundant channel in the same trip system is in trip, placing the second channel in trip will result in an RPT trip. Required Action A.2.2 was not intended to force an unnecessary trip. In this event,

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ACTIONS (continued)

Required Action A.2.1 would have to be met. If inoperable channel(s) are not restored to OPERABLE status or placed in trip within the allowed Completion Time, Required Action B.1 should be met.

Condition B

B.1

If Required Action A.1 and Required Action A.2.1, or Required Action A.2.2 are not carried out within the allotted time, Required Action B.1 is required. Required Action B.1 allows 6 hours to place the plant in MODE 2. The Completion Time of 6 hours is reasonable, based on operating experience, to reach the required mode from full power in an orderly manner and without challenging plant systems.

Condition C

C.1 and C.2

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 trip system(s). The specified Completion Time of 1 hour is sufficient for plant operations personnel to make this verification.

Required Action C.1 ensures that those identified Required Actions associated with supported systems impacted by the inoperability of trip systems(s) associated with each ATWS-RPT 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 ATWS-RPT funct[illegible] are as follows:]

Required Action C.2 verifies that [illegible] required support or supported features associated with the other trip system are OPERABLE within a Completion Time of 1 hour. The specified Completion Time is sufficient for plant operations personnel to make this verification. If verification determines loss of functional capability, LCO 3.0.3 must be immediately entered. However, if the support or supported feature LCOs 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

The SRs for LCO 3.3.4.2 are provided in the SR section and have been clarified by a Note that allows placing one trip system for a single function in inoperable status for up to 2 hours for performance of required surveillances as long as one trip system remains OPERABLE. Upon completion of the surveillance or expiration of the 2-hour allowance, the trip system must be returned to OPERABLE status or the Required Actions taken. [For this facility, the 2-hour allowance is based on the following:]

SR 3.3.4.2.1

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 the two instrument channels could be an indication of excessive instrument drift in one of the channels or something even more serious. CHANNEL CHECK will detect gross channel failure, thus it is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff based on a 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 surveillances are 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 [illegible] 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

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

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 3.3.4.2.2

A CHANNEL FUNCTIONAL TEST verifies the function of the trip, interlock, and alarm functions of a channel. The test inserts a simulated or actual signal as close to the sensor as practicable and verifies required trips, 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 must be found within the ALLOWABLE VALUES 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 which cannot be corrected by recalibration.

[For this facility, the basis for the 31-day test Frequency is as follows:]

This SR is modified by a Note that the RPT breakers are not required to be tripped during this test. This is 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.

SR 3.3.4.2.3

Calibration of trip units consists of a test to determine actual trip setpoints. Trip setpoints are adjusted if found outside the acceptable "as found" tolerance. If during trip unit calibration, the associated trip setting is discovered to be less conservative than the ALLOWABLE VALUE specified in SR 3.3.4.2.4, the channel must be declared inoperable.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

Mea-urement and setpoint error determination and r-justment must be performed consistent with the -mptions of the plant-specific setpoint analysis.

The 31-day Surveillance Frequency is based on assumptions in the methodology included in the determination of the trip setpoint. SR 3.3.4.2.2 and SR 3.3.4.2.3 are often performed simultaneously using a common procedure.

SR 3.3.4.2.4

CHANNEL CALIBRATION is a complete check of the instrument channel including the detector. The test verifies that the channel responds to measured parameter valves 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 analyses. 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 setpoints 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 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.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.4.2.5

Performance of a LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required actuation logic for a specific channel. The LOGIC SYSTEM FUNCTIONAL TEST tests all logic components (i.e., all relays and contacts, trip units, solid-state logic elements, etc.) of a logic circuit, from sensor up to the actuated device. The system functional test of the pump breakers is included as part of this test and overlaps the CHANNEL FUNCTIONAL TEST to provide complete testing of the assumed safety function. If the breakers are incapable of operating, the associated instrument channel would be inoperable.

The 18-month Frequency was developed considering it is prudent that the Surveillance be performed only 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 at the 18-month Frequency.

REFERENCES

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

2. Title 10, Code of Federal Regulations, Part 50.62, "Requirements for Reduction of Risks from Anticipated Transients Without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants."

B 3.3 INSTRUMENTATION

B 3.3.5.1 Emergency Core Cooling System (ECCS) Instrumentation

BASES

BACKGROUND

ECCS instrumentation channels generate signals to automatically initiate those safety systems which ensure that the fuel is adequately cooled in the event of a design basis transient or accident.

The ECCS instrumentation is designed to include the three subsystems identified below:

- Field transmitters or process sensors;
- Signal processing and bistable modules; and
- Trip logic.

Field Transmitters or Process Sensors

Field transmitters or process sensors provide a measurable electronic signal based on the physical characteristics of the parameter being measured.

Typically, four measurement channels with physical separation are provided for each parameter. These are typically organized into two trip systems which are physically and electrically separated. Four measurement channels are necessary to meet the redundancy and testability of 10 CFR 50, Appendix A, GDC 21 (Ref. 1) and to implement the one-out-of-two taken twice logic arrangement discussed for the ECCS instrumentation.

[For this facility, a discussion of the ECCS parameters that do not have four measurement channels and their conformance to redundancy requirements and testability of 10 CFR 50. Appendix A, GDC 21 is as follows:]

For most anticipated operational occurrences (AOOs) and Design Basis Accidents (DBAs), a wide range of dependent and independent parameters are monitored.

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BACKGROUND (continued)

Signal Processing and Bistable Modules

Each process parameter measurement channel includes electronic equipment which provides signal conditioning, comparable output signals for main control board instruments, comparison of measured input signals with setpoints established by safety analyses, and output to the trip logic channels. This output to the trip logic channels is taken from a bistable device, which can be mechanical switches that are part of the process sensors or electronic comparators that receive input from the process transmitters or sensors. In either case, the bistable output contacts are considered to be part of the trip logic channel.

Trip Logic, Trip Setpoints, and ALLOWABLE VALUES

Trip setpoints are those predetermined values of output voltage or current against which the output voltage or current related to the present value of the process parameter is compared. If the present measured output value of the process parameter exceeds the setpoint, the associated bistable changes state. The trip setpoints are the nominal values at which the bistables are set. They are derived from the limiting values of the process parameters obtained from the accident analyses (analytical limits) through a process of correction for uncertainties and errors set forth in the plant-specific setpoint methodology (Ref. 4). The analytical limits, corrected for analytical and process uncertainties, become the ALLOWABLE VALUES, which when further corrected by the methodology of Reference 4, become the calculated trip setpoint values.

The setpoints derived in this manner provide adequate protection because sensor and processing time delays are accounted for as well as calibration tolerances, instrumentation uncertainties, instrument drift, and severe environment errors for channels that must function in harsh environments as defined by 10 CFR 50.49 (Ref. 6).

The actual nominal trip setpoint entered into the bistable is usually still more conservative than that calculated by the plant-specific setpoint methodology. If the setpoint measured for the bistable by the surveillance test does not exceed the documented surveillance test acceptance criteria, the bistable is considered OPERABLE.

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BASES (continued)

BACKGROUND (continued)

Setpoints set consistent with the ALLOWABLE VALUE will ensure that safety limits are not violated during AOOs and that the consequences of DBAs will be acceptable, provided the plant is being operated within the LCOs at the onset of the AOO or DBA and the equipment functions as designed.

Each channel of the process control equipment can be tested on line to verify that the signal and setpoint accuracy is within the specified allowance requirements for the setpoint as in Reference 4. 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 ALLOWABLE VALUES listed in Table 3.3.5.1-1 are based upon the setpoint 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.

The ECCS instrumentation actuates Core Spray (CS) Low Pressure Coolant Injection (LPCI), High Pressure Coolant Injection (HPCI), and Automatic Depressurization System (ADS). The equipment involved with each of these systems is identified in LCO 3.5.1, "ECCS—Operating" Bases.

<u>Core Spray System</u>

The CS System will be initiated automatically and is backed up by manual means. Automatic initiation occurs for conditions of Reactor Vessel Water Level--Low Low Low, Level 1 or Drywell Pressure--High. The outputs of the trip units are connected to relays whose contacts are arranged in a one-out-of-two taken twice logic for each function.

The high drywell pressure initiation signal is a latching signal and must be manually reset. The CS System can be reset if reactor water level has been restored, even if the high drywell pressure condition persists.

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BASES (continued)

BACKGROUND (continued)

The CS pump discharge flow is monitored by a flow transmitter. When the pump is running and discharge flow is low enough that pump overheating may occur, the minimum flow return line valve is opened. The valve is automatically closed if flow is normal to allow the full system flow assumed in the accident analysis.

The CS System also monitors the pressure in the reactor to ensure that, before the injection valves open, the reactor pressure has fallen to a value below the CS System's maximum design pressure.

Low Pressure Coolant Injection System

The LPCI System will be initiated automatically and is backed up by manual means. Automatic initiation occurs for conditions of Reactor Vessel Water Level--Low Low Low, Level 1, Drywell Pressure--High, or both. The outputs of the trip units are connected to relays whose contacts are arranged in a one-out-of-two taken twice logic for the combined functions. Once an initiation signal is received by the LPCI control circuitry, the signal is sealed in until manually reset.

If an automatic-initiation loss-of-coolant accident (LOCA) signal occurs concurrent with a loss of power signal, the LPCI C pump starts immediately when power to the 4160 V emergency buses is restored. The LPCI A, B, and D pumps start after a 10-second delay to limit the loading of the standby power sources.

If the pump is running but discharge flow is low enough that pump overheating may occur, the minimum-flow return line valve is signalled to open. If flow is normal, the valve is automatically closed to allow the full system flow assumed in the analyses.

The LPCI System monitors the pressure in the reactor to ensure that, before the injection valves open, the reactor pressure has fallen to a value below the LPCI System's maximum design pressure. Additionally, instruments are provided to close the recirculation-pump discharge valves to ensure that LPCI flow does not bypass the core when it injects into the recirculation lines.

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BASES (continued)

BACKGROUND (continued)

Low reactor water level in the shroud is detected by two additional instruments to automatically isolate other modes of residual heat removal (RHR) (e.g., suppression pool cooling) when LPCI is required. Manual overrides for these isolations are provided.

High Pressure Coolant Injection System

The HPCI System will be initiated automatically and is backed up by manual means. Automatic initiation occurs for conditions of Reactor Vessel Water Level--Low Low, Level 2 or Drywell Pressure--High. The outputs of the trip units are connected to relays whose contacts are arranged in a one-out-of-two taken-twice logic for each function. The HPCI System initiation signal can be reset if the initiating conditions have cleared; the HPCI pump can then be stopped and the system realigned for automatic initiation. Automatic restart will occur if either of the two initiation conditions recurs.

On receipt of high water level signals (Level 8), the HPCI turbine trips, which causes the turbine's stop valve and the injection valves to be closed. The logic is two-out-of-two to provide high reliability of the HPCI System. The HPCI System restarts if a low low water level signal is subsequently received.

The HPCI System also monitors the water levels in the CST and the suppression pool, because these are the two sources of water for HPCI operation. Reactor-grade water in the CST is the normal source. On receipt of an HPCI initiation signal, the CST suction valves are automatically signaled to open (they are normally in the open position) unless both suppression pool suction valves are open. If the water level in the CST falls below a preselected level, first the suppression pool suction valves automatically open, and then the CST suction valves automatically close. Two level transmitters are used to detect low water level in the CST. Either transmitter and its associated trip unit can cause the suppression pool suction valves to open and the CST suction valves to close. The suppression pool suction valves also automatically open and the CST suction valves close if high water level is detected in the suppression pool. To prevent losing suction to the pump, the suction

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(continued)

BASES (continued)

BACKGROUND (continued)

valves are interlocked so that one suction path must be open before the other automatically closes.

To protect the HPCI pump, a minimum-flow bypass line has been installed. If flow is normal, the valve in this line is automatically closed to allow the full system flow assumed in the analyses.

Automatic Depressurization System

The ADS will be initiated automatically and is backed up by manual means. Automatic initiation occurs when signals indicating Reactor Vessel Water Level--Low Low Low, Level 1; Drywell Pressure--High, or Low Water Level Actuation Timer; and confirmed Reactor Vessel Water Level--Low, Level 3; and CS or LPCI Pump Discharge Pressure High are all present and the ADS actuation timer has timed out.

Each ADS trip system includes a time delay between satisfying the initiation logic and the actuation of the relief valve. The time delay chosen is long enough that the HPCI has sufficient operating time to recover to a level above Level 1, yet not so long that the LPCI and CS Systems are unable to adequately cool the fuel if the HPCI fails to maintain that level. An alarm in the control room is annunciated when either of the timers is timing. Resetting the ADS initiating signals resets the timers.

The ADS also monitors the discharge pressures of the four LPCI pumps and the two CS pumps. Their signals are used as permissives for ADS relief valve actuation, indicating that there is a source of core coolant available once the ADS has depressurized the vessel.

The ADS logic in each trip system is arranged in two strings. Each string has a contact from each of the following variables: Reactor Vessel Water Level--Low Low Low, Level 1; Drywell Pressure--High; Low Water Level Actuation Timer; and confirmed Reactor Vessel Water Level--Low, Level 3. All contacts in both logic strings must close, the ADS initiation timer must time out, and a CS or LPCI discharge pressure signal must be present to initiate an ADS trip system. Either the A or B trip system will cause all the ADS relief valves to open. Once the Drywell

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(continued)

BASES (continued)

BACKGROUND (continued)

Pressure--High signal or the ADS initiation signal are present, they are individually sealed in until manually reset.

Manual inhibit switches are provided in the control room for the ADS. Their function is required for ADS OPERABILITY, since the ADS is inoperable if the manual inhibit switches are enabled when the ADS is required to be OPERABLE.

APPLICABLE SAFETY ANALYSES

The core cooling function of the ECCS, together with the scram action of the RPS assures that the fuel peak-cladding temperature remains below the limits of 10 CFR 50.46. It also assures that the fuel and reactor pressure safety limits are not exceeded. The capability of the ECCS to be initiated during the analyzed transients and accidents is presented in Reference 4. Each of the ECCS actuation instruments in Table 3.3.5.1-1 support their system OPERABILITY requirements that are provided by LCO 3.5.1 and LCO 3.5.2. The specific safety analyses applicability is discussed on a function by function basis below.

<u>Reactor Vessel Water Level--Low Low Low, Level 1</u>

The Reactor Vessel Water Level--Low Low Low, Level 1 is one of the functions assumed to be OPERABLE and capable of initiating the ECCS during the transients and accidents analyzed in the plant-specific FSAR. For these events, the ECCS ensures that high pressure and fuel coverage safety limits are not exceeded. The Reactor Vessel Water Level--Low Low Low, Level 1 function is directly assumed in the plant-specific analysis of the recirculation line break. The core-cooling function of the ECCS, along with the scram action of the Reactor Protection System (RPS), assures that the fuel peak-cladding temperature remains below the limits of 10 CFR 50.46.

<u>Drywell Pressure--High</u>

The Drywell Pressure--High is assumed to be OPERABLE and capable of initiating the ECCS during the transients and accidents analyzed in the plant-specific FSAR. For these events, the ECCS ensures that high pressure and fuel coverage safety limits are not exceeded. The Drywell

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

Pressure--High function, along with the level function, is directly assumed in the analysis of the recirculation line break. The core-cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak-cladding temperature remains below the limits of 10 CFR 50.46.

Reactor Steam Dome Pressure--Low (Injection Permissive)

The Reactor Steam Dome Pressure--Low is one of the functions assumed to be OPERABLE and capable of permitting initiation of the ECCS during the transients analyzed in the plant-specific FSAR. For these events, the ECCS ensures that high pressure and fuel coverage safety limits are not exceeded. The core-cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak-cladding temperature remains below the limits of 10 CFR 50.46. The Reactor Steam Dome Pressure--Low function is directly assumed in the analysis of the recirculation line break.

Core Spray Pump Discharge Flow--Low (Bypass)

The minimum flow interlock (MFI) closure is one of the functions assumed to be OPERABLE and capable of closing the valve during the transients analyzed in the FSAR. For these events, the CS Pump Discharge Flow--Low function supports the ECCS to ensure that high pressure and fuel coverage safety limits are not exceeded. The core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak-cladding temperature remains below the limits of 10 CFR 50.46.

Reactor Steam Dome Pressure--Low (Recirculation Discharge Valve Permissive)

The Reactor Steam Dome Pressure--Low is one of the functions assumed to be OPERABLE and capable of closing the valve during the transients analyzed in the FSAR. For these events, the ECCS ensures that high pressure and fuel coverage safety limits are not exceeded. The core-cooling function of the ECCS, along with the scram action of the RPS, assures that the fuel peak-cladding temperature remains below the limits of 10 CFR 50.46. The Reactor Steam Dome Pressure--Low function is directly assumed in the analysis of the recirculation line break (Ref. 2).

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

Reactor Vessel Shroud Level--Level 0

The Reactor Vessel Shroud Level--Level 0 function is implicitly assumed in the analysis of the recirculation line break, because the analysis assumes no LPCI flow diversion when reactor vessel water level is below Level 0.

Low Pressure Coolant Injection Pump Start--Time Delay Relay

The LPCI Pump Start--Time Delay Relays are assumed to be OPERABLE in the accident and transient analyses requiring ECCS initiation. That is, the analysis assumes that the pumps will initiate when required and that excess loading will not cause failure of the power sources.

Low Pressure Coolant Injection Pump Discharge Flow--Low (Bypass)

The MFI closure is one of the functions assumed to be OPERABLE and capable of closing the valve during the transients analyzed in the FSAR. For these events, the ECCS ensures that high reactor pressure and fuel coverage safety limits are not exceeded. The core-cooling function of the ECCS, along with the scram action of the RPS, assures that the fuel peak-cladding temperature remains below the limits of 10 CFR 50.46.

Reactor Vessel Water Level--Low Low, Level 2

The Reactor Vessel Water Level--Low Low, Level 2 is one of the functions assumed to be OPERABLE and capable of initiating HPCI during the transients analyzed in the FSAR. For these events, the ECCS ensures that high pressure and fuel coverage safety limits are not exceeded. The Reactor Vessel Water Level--Low Low, Level 2 function associated with HPCI is directly assumed in the analysis of the recirculation line break. The core-cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak-cladding temperature remains below the limits of 10 CFR 50.46.

Reactor Vessel Water Level--High, Level 8

The Level 8 signal is used to close the HPCI turbine stop and injection valves to prevent vessel water overflow into the steam lines.

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

Condensate Storage Tank Level--Low

If the water level in the CST falls below a preselected level, however, first the suppression pool suction valves automatically open and then the CST suction valves automatically close. To prevent losing suction to the pump, the suction valves are interlocked so that one suction path must be open before the other automatically closes. This ensures that an adequate supply of makeup water is available to the HPCI pump. This function is implicitly assumed in the accident and transient analyses (which take credit for HPCI) because the analyses assume that the HPCI suction source is the suppression pool.

Suppression Pool Water Level--High

With excessively high suppression pool water levels, the loads on the suppression pool could exceed design values should there be a blowdown of the reactor vessel pressure through the safety/relief valves. Therefore, signals indicating high suppression pool water level are used to transfer the suction source of HPCI from the CST to the suppression pool to eliminate the possibility of HPCI continuing to provide additional water from a source outside the containment. This function is implicitly assumed in the accident and transient analyses (which take credit for HPCI), because the analyses assume the HPCI suction source is the suppression pool.

High Pressure Coolant Injection Pump Discharge Flow--Low (Bypass)

The MFI closure is one of the functions assumed to be OPERABLE and capable of closing the valve during the transients analyzed. For these events, the ECCS ensures that high pressure and fuel coverage safety limits are not exceeded. The core-cooling function of the ECCS, along with scram action of the RPS, assures that the fuel peak-cladding temperature remains below the limits of 10 CFR 50.46.

Automatic Depressurization System Initiation Timer

The ADS Initiation Timer function is assumed OPERABLE for the accident and transient analyses that require ECCS initiation and assume failure of the HPCI System.

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

Reactor Vessel Water Level--Low, Level 3 (Confirmatory)

The Reactor Vessel Water Level 3 function is used by the ADS portion of ECCS only as a confirmatory low water level signal. ADS is initiated on receipt of a Reactor Vessel Water Level--Low Low Low, Level 1 signal. To prevent spurious initiation of the ADS due to spurious Level 1 signals, a Level 3 signal must also be received before the ADS initiation commences. The Level 3 trip function must be OPERABLE in order for the ADS to initiate.

Core Spray Pump Discharge Pressure--High

CS Pump Discharge Pressure--High is one of the functions assumed to be OPERABLE and capable of permitting ADS initiation during the events analyzed in the FSAR with an assumed HPCI failure. For these events, the ADS depressurizes the reactor vessel so that the low-pressure ECCS subsystems can perform the core-cooling functions. This core-cooling function of the ECCS, along with the scram action of the RPS, assures that the fuel peak-cladding temperature remains below the limits of 10 CFR 50.46.

Low Pressure Coolant Injection Pump Discharge Pressure--High

LPCI Pump Discharge Pressure--High is one of the functions assumed to be OPERABLE and capable of permitting ADS initiation during the events analyzed in the FSAR with an assumed HPCI failure. For these events, the ADS depressurizes the reactor vessel so that the low-pressure ECCS subsystems can perform the core-cooling functions. This core-cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak-cladding temperature remains below the limits of 10 CFR 50.46.

Automatic Depressurization System Low Water Level Actuation Timer

One of the signals required for ADS initiation is Drywell Pressure--High. If the event requiring ADS initiation occurs outside the drywell, however—for example, a steam line break (SLB) outside containment—a high drywell pressure signal may never be present. Therefore, the ADS Low Water Level Actuation Timer is used to bypass the Drywell Pressure--High function after a certain time period has elapsed, after which ADS initiation requires only the

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

reactor vessel water level and one LPCI pump running signal. The instrumentation is a component of the ADS and is therefore part of the primary success path for mitigation of a DBA or transient.

ECCS instrumentation satisfies Criterion 3 of the NRC Interim Policy Statement.

LCO

The LCO requires all instrumentation performing an ECCS function to be OPERABLE. Failure of any instrument renders the affected channel(s) inoperable and reduces the reliability of the affected functions. Each function must have the required number of OPERABLE channels per ECCS instrument function, with their setpoints within the specified ALLOWABLE VALUES. Actuation setpoints are calibrated to be consistent with applicable setpoint methodology assumptions. Each channel must also respond within its assumed response time.

ALLOWABLE VALUES are specified for each ECCS actuation 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 measurements 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 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 (Ref. 4).

Table 3.3.5.1-1 footnote (b) is added to show that certain ECCS instrumentation functions also perform diesel generator initiation.

Violation of the LCO for any ECCS function could allow the plant to reach conditions during steady-state and transient operation that are beyond those evaluated for safe plant

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BASES (continued)

LCO (continued)

operation. If exceeded, these conditions could lead to fuel failure or vessel overpressurization.

The ECCS functions specified in Table 3.3.5.1-1 are OPERABLE when the following conditions are satisfied:

1. All channel components necessary to provide an ECCS actuation signal are functional and in service;

2. Channel measurement uncertainties are known via test, analysis, or design information to be within the assumptions of the setpoint calculations; and

3. Required surveillance testing is current and has demonstrated performance within each surveillance test's acceptance criteria.

The Bases for the LCO requirements for each function are discussed below.

1. Core Spray System

1.a. Reactor Vessel Water Level--Low Low Low, Level 1

Low reactor vessel water level indicates that the capability of cooling the fuel may be threatened. Should the water level decrease too far, the core may become uncovered and fuel damage could result.

Reactor Vessel Water Level--Low Low Low, Level 1 signals are initiated from four level transmitters connected to relays whose contacts are arranged in a one-out-of-two taken twice logic.

The Reactor Vessel Water Level--Low Low Low, Level 1 ALLOWABLE VALUE is set high enough to allow time for the low-pressure core flooding systems to activate and provide adequate cooling, but low enough to prevent decreases in level that are due to operational transients from causing spurious initiation.

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BASES (continued)

LCO (continued)

The Reactor Vessel Water Level--Low Low Low, Level 1 function is required to be OPERABLE only when the CS ECCS is required to be OPERABLE.

Thus, four channels are required to be OPERABLE in MODES 1, 2, and 3. In MODES 4 and 5, the Level 1 function is required only for those low-pressure ECCS subsystems required to be OPERABLE by LCO 3.5.2.

1.b. Drywell Pressure--High

High pressure in the drywell could indicate a break in the reactor coolant pressure boundary (RCPB). All ECCS injection and spray subsystems are initiated on receipt of the Drywell Pressure--High signal to minimize the possibility of fuel damage.

High drywell pressure signals are initiated from four pressure transmitters that are connected to relays whose contacts are arranged in a one-out-of-two taken twice logic.

The ALLOWABLE VALUE was selected to be as low as possible to minimize heat loads on equipment located in the drywell, but not so low as to cause spurious trips.

The Drywell Pressure--High function is required to be OPERABLE when the CS ECCS is required to be OPERABLE in conjunction with times when the primary containment is required to be OPERABLE. Thus, four channels of the Drywell Pressure--High function are required to be OPERABLE in MODES 1, 2, and 3. In MODES 4 and 5, the Drywell Pressure--High function is not required, because there is insufficient energy in the reactor to pressurize the primary containment to the Drywell Pressure--High function setpoint.

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BASES (continued)

LCO (continued)

1.c. Reactor Steam Dome Pressure--Low (Injection Permissive)

Low reactor steam dome pressure signals are used as permissives for the low-pressure ECCS subsystems. This ensures that, prior to opening the injection valves of the low-pressure ECCS subsystems, the reactor pressure has fallen to a value below these subsystems' maximum design pressure.

The Reactor Steam Dome Pressure--Low function is directly assumed in the analysis of the recirculation line break. The Reactor Steam Dome Pressure--Low signals are initiated from four pressure transmitters that sense the reactor steam dome pressure. The transmitters provide input to four channels arranged in a one-out-of-two taken twice logic configuration.

The ALLOWABLE VALUE is set low enough to prevent overpressurizing the equipment in the low-pressure ECCS, but high enough to ensure that the ECCS injection prevents the peak cladding temperature (PCT) from exceeding 2200°F.

The Reactor Steam Dome Pressure--Low function is required to be OPERABLE only when the CS ECCS is required to be OPERABLE. Thus, four channels of the Reactor Steam Dome Pressure--Low function are required to be OPERABLE in MODES 1, 2, and 3. In MODES 4 and 5, the function is required to be OPERABLE only as required by LCO 3.5.2.

1.d. Core Spray Pump Discharge Flow--Low (Bypass)

The MFIs provide equipment protection. They support pump OPERABILITY by protecting the pump from overheating when the associated ECCS pumps initiate. The CS pump minimum-flow bypass valve is opened when low flow is sensed and automatically closed at normal flow conditions.

One flow transmitter per CS pump is used to detect the associated system flow rate. The CS

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BASES (continued)

LCO (continued)

System has two transmitters. The logic is arranged so that each transmitter causes its associated minimum-flow valve to open or close. The ALLOWABLE VALUES for the minimum-flow interlocks are high enough to ensure that the pump is operating, and low enough to ensure that closure of the minimum-flow bypass is initiated to allow full flow into the core.

The MFIs are required to be OPERABLE when the CS System is required to be OPERABLE. Thus, two channels of CS MFIs are required to be OPERABLE in MODES 1, 2, and 3. In MODES 4 and 5, the CS minimum-flow function is required only if the CS ECCS subsystem is required to be OPERABLE by LCO 3.5.2.

1.e. Manual Initiation

The Manual Initiation push button switches introduce signals into the appropriate ECCS subsystem that are redundant to the automatic protective instrumentation and provide manual initiation capability. One switch per subsystem is supplied for CS initiation which is capable of initiating its associated subsystem.

2. Low Pressure Coolant Injection System

2.a. Reactor Vessel Water Level--Low Low Low, Level 1

Low Reactor Vessel water level indicates that the capability of cooling the fuel may be threatened. Should water level decrease too far, the core may become uncovered and fuel damage could result.

Reactor Vessel Water Level--Low Low Low, Level 1 signals are initiated from four level transmitters connected to relays whose contacts are arranged in a one-out-of-two taken twice logic. The Reactor Vessel Water Level--Low Low Low, Level 1 ALLOWABLE VALUE is high enough to allow time for the low-pressure core flooding systems to activate and provide adequate cooling,

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BASES (continued)

LCO (continued)

but low enough to ensure that decreases in level that result from operational transients will not cause spurious initiation.

The Reactor Vessel Water Level--Low Low Low, Level 1 function is required to be OPERABLE when the LPCI System is required to be OPERABLE. Thus, four channels of the Level 1 function are required to be OPERABLE in MODES 1, 2, and 3. In MODES 4 and 5, the Level 1 function is required only for those low-pressure ECCS subsystems required to be OPERABLE by LCO 3.5.2.

2.b. <u>Drywell Pressure--High</u>

High pressure in the drywell could indicate a break in the RCPB. All ECCS injection and spray subsystems are initiated on receipt of the Drywell Pressure--High signal in order to minimize the possibility of fuel damage.

High drywell pressure signals are initiated from four transmitters that are connected to relays whose contacts are arranged in a one-out-of-two taken twice logic.

The ALLOWABLE VALUE selected is set low enough to minimize heat loads on equipment located in the drywell, but not so low that it could cause spurious trips.

The Drywell Pressure--High function is required to be OPERABLE when the associated ECCS is required to be OPERABLE in conjunction with times when the primary containment is required to be OPERABLE. Thus, four channels of the LPCI Drywell Pressure--High function are required to be OPERABLE in MODES 1, 2, and 3. In MODES 4 and 5, the Drywell Pressure--High function is not required, because there is insufficient energy in the reactor to pressurize the primary containment to the Drywell Pressure--High function setpoint.

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BASES (continued)

LCO (continued)

2.c. Reactor Steam Dome Pressure--Low (Injection Permissive)

Low reactor steam dome pressure signals are used as permissives for the low-pressure ECCS subsystems. This ensures that, prior to opening the injection valves of the low-pressure ECCS subsystems, the reactor pressure has fallen to a value below these subsystems' maximum design pressure.

The Reactor Steam Dome Pressure--Low signals are initiated from four pressure transmitters that sense the reactor steam dome pressure. The transmitters provide input to four channels arranged in a one-out-of-two taken twice logic.

The ALLOWABLE VALUE is set low enough to prevent overpressurizing the ECCS, but high enough to ensure that the ECCS injection prevents the PCT from exceeding 2200°F.

The Reactor Steam Dome Pressure--Low function is required to be OPERABLE when the LPCI ECCS is required to be OPERABLE. Thus, four channels of the Reactor Steam Dome Pressure--Low function are required to be OPERABLE in MODES 1, 2, and 3. In MODES 4 and 5, the function is required to be OPERABLE only as required by LCO 3.5.2.

2.d. Reactor Steam Dome Pressure--Low (Recirculation Discharge Valve Permissive)

Low reactor steam dome pressure signals are used as permissives for recirculation discharge valve closure. This ensures that, prior to closing the valves, the reactor pressure has fallen to a value below the closing maximum design pressure.

The Reactor Steam Dome Pressure--Low signals are initiated from four pressure transmitters that sense the reactor dome pressure. The four transmitters are connected to relays whose contacts are arranged in a one-out-of-two taken twice logic.

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BASES (continued)

LCO (continued)

The ALLOWABLE VALUE is set low enough to prevent overpressurizing the recirculation discharge valves, but high enough to ensure that the injection prevents the PCT from exceeding 2200°F.

The Reactor Steam Dome Pressure--Low function is required to be OPERABLE when the associated ECCS is required to be OPERABLE. Thus, four channels of the LPCI Reactor Steam Dome Pressure--Low function are required to be OPERABLE in MODES 1, 2, and 3. In MODES 4 and 5, the function is required only for those LPCI subsystems required to be OPERABLE by LCO 3.5.2.

2.e. Reactor Vessel Shroud Level--Level 0

The Reactor Vessel Shroud Level--Level 0 function is provided as a permissive to allow the RHR System to be manually aligned from the LPCI mode to the suppression pool cooling and spray or drywell spray modes. The permissive ensures that water in the vessel is approximately two-thirds of the core-height before the manual transfer is allowed. This ensures that LPCI is available to prevent or minimize fuel damage. This function may be overridden during accident conditions as allowed by plant procedures.

Reactor Vessel Shroud Level--Level 0 signals are initiated from two level transmitters that are connected to relays whose contacts are arranged in a two-out-of-two logic.

The ALLOWABLE VALUE for the Reactor Vessel Shroud Level--Level 0 is high enough to allow the low-pressure core-flooding systems to activate and provide adequate cooling, but low enough that decreases in level resulting from operational transients will not cause spurious actuations resulting in isolation of the RHR containment-cooling modes.

The Reactor Vessel Shroud Level--Level 0 function is required to be OPERABLE when the associated ECCS is required to be OPERABLE. Thus, two

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BASES (continued)

LCO (continued)

channels of the LPCI--Level 0 function are required to be OPERABLE in MODES 1, 2, and 3. In MODES 4 and 5, the LPCI--Level 0 function is required only for those LPCI subsystems required to be OPERABLE by LCO 3.5.2.

2.f. Low Pressure Coolant Injection Pump Start--Time Delay Relay

The purpose of this time delay is to stagger the start of the four main low-pressure LPCI pumps, thus limiting the starting transients on the power buses.

There are four LPCI Pump Start--Time Delay Relays, one in each of the RHR pump start logic circuits. Because each time delay relay is dedicated to a single pump start logic, a single failure of a LPCI Pump Start--Time Delay Relay will cause only the one LPCI pump to fail to start. This leaves five of the six LPCI pumps still OPERABLE.

The ALLOWABLE VALUE for the LPCI Pump Start--Time Delay Relays is long enough that the majority of the starting transient of the first pump is complete before starting the second pump on the same 4160 V emergency bus and short enough that ECCS operation is not degraded.

The LPCI Pump Start--Time Delay Relay function is required to be OPERABLE only when the associated ECCS is required to be OPERABLE. Thus, one channel of LPCI Pump Start--Time Delay Relay per LPCI pump is required to be OPERABLE in MODES 1, 2, and 3. In MODES 4 and 5, the function is required only for those LPCI pumps required to be OPERABLE by LCO 3.5.2.

2.g. Low Pressure Coolant Injection Pump Discharge Flow--Low (Bypass)

The MFIs are provided to allow the associated ECCS pumps to initiate and to protect the pump from overheating. The LPCI pump minimum-flow

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BASES (continued)

LCO (continued)

bypass valve is opened when low flow is sensed and automatically closed at normal flow conditions.

One flow transmitter per LPCI pump is used to detect the LPCI System's flow rates. The logic is arranged such that each transmitter causes its associated minimum-flow valve to open or close.

The ALLOWABLE VALUES for MFIS are high enough to ensure that the pump is operating and low enough to ensure that closure of the minimum-flow bypass valve is initiated to allow full flow into the core.

The MFIS are required to be OPERABLE when the LPCI is required to be OPERABLE. Thus, four channels of the LPCI MFIs, one per pump, are required to be OPERABLE in MODES 1, 2, and 3. In MODES 4 and 5, the LPCI minimum-flow functions are required only for those low-pressure ECCS subsystems required to be OPERABLE by LCO 3.5.2.

2.h. Manual Initiation

Manual Initiation signals are redundant to the automatic protective instrumentation and provide Manual Initiation capability. One minimum-flow interlocks push button switch is supplied per subsystem for LPCI initiation which is capable of initiating its associated subsystem.

3. High Pressure Coolant Injection System

3.a. Reactor Vessel Water Level--Low Low, Level 2

Low reactor vessel water level indicates that the capability of cooling the fuel may be threatened. Should reactor vessel water level decrease too far, fuel damage could result. Therefore, the HPCI System is initiated at Level 2 to maintain the water level above the top of the active fuel.

Reactor vessel water level signals are initiated from four level transmitters that are connected

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BASES (continued)

LCO (continued)

to relays whose contacts are arranged in a one-out-of-two taken twice logic.

The Reactor Vessel Water Level--Low Low, Level 2 ALLOWABLE VALUE is high enough that, for complete loss of feedwater flow, the Reactor Core Isolation Cooling (RCIC) System flow (with HPCI assumed to fail) will be sufficient to avoid initiation of low-pressure ECCS at Level 1. The ALLOWABLE VALUE is also low enough that, after a scram caused by a Level 3 trip with no loss of feedwater flow, the RCIC and HPCI Systems will not be initiated.

The Reactor Vessel Water Level--Low Low, Level 2 function is required to be OPERABLE only when the HPCI ECCS is required to be OPERABLE. Thus, four channels of the HPCI Level 2 function are required to be OPERABLE in MODE 1, and in MODES 2 and 3 when reactor steam dome pressure is > 150 psig.

3.b. Drywell Pressure--High

High pressure in the drywell could indicate a break in the RCPB. All ECCS injection and spray subsystems are initiated on receipt of the Drywell Pressure--High function to minimize the possibility of fuel damage.

High drywell pressure signals are initiated from four pressure transmitters that are connected to relays whose contacts are arranged in a one-out-of-two taken twice logic.

The ALLOWABLE VALUE selected is set low enough to minimize heat loads on equipment located in the drywell, but not so low that it could cause spurious trips.

The Drywell Pressure--High function is required to be OPERABLE when the HPCI ECCS required to be OPERABLE in conjunction with times when the primary containment is required to be OPERABLE.

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BASES (continued)

LCO (continued)

Four channels of the HPCI Drywell Pressure--High function are required to be OPERABLE in MODE 1, and in MODES 2 and 3 when reactor steam dome pressure is > 150 psig.

3.c. Reactor Vessel Water Level--High, Level 8

High reactor vessel water level indicates there is sufficient cooling water inventory in the reactor vessel that there is no danger to the fuel. Therefore, the Level 8 signal is used to close the HPCI turbine stop and injection valves to prevent overflow into the main steam lines.

Reactor Vessel Water Level--High, Level 8 signals for HPCI are initiated from two level transmitters from the wide-range level measurement system. The transmitters used for the Level 8 signal generation provide input to two channels. Both Level 8 signals are required in order to close the HPCI turbine and stop injection valves. The Reactor Vessel Water Level--High, Level 8 ALLOWABLE VALUE is high enough to preclude tripping of the HPCI System when it is required for core cooling, yet low enough to trip the HPCI System prior to water overflowing into the main steam lines.

The Reactor Vessel Water Level--High, Level 8 function is required to be OPERABLE only when the associated ECCS is required to be OPERABLE. Thus, two channels of the HPCI Level 8 function are required to be OPERABLE in MODE 1, and in MODES 2 and 3 when reactor steam dome pressure is > 150 psig.

3.d. Condensate Storage Tank Level--Low

Low level in the CST indicates the unavailability of an adequate supply of makeup water from this normal source. Normally, the suction valve between the HPCI and the CST is open and, upon receiving a HPCI initiation signal, water for HPCI would be taken from the CST.

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BASES (continued)

LCO (continued)

Two level transmitters are used to detect low water level in the CST. The logic is arranged so that either transmitter and its associated trip unit can cause the suppression pool suction valves to open and the CST suction valve to close.

The ALLOWABLE VALUE for the CST Level--Low function is high enough to ensure adequate pumpsuction head while water is being taken from the CST, and low enough not to cause spurious cycling of the suction valves due to normal tank level fluctuations.

The CST Level--Low function is required to be OPERABLE only when the HPCI ECCS is required to be OPERABLE. Thus, two channels of the HPCI CST Level--Low function are required to be OPERABLE in MODE 1, and in MODES 2 and 3 when reactor steam dome pressure is > 150 psig.

3.e. Suppression Pool Water Level--High

Suppression pool water level signals are initiated from two level transmitters that measure the pressure difference between two sealed diaphragms: one sensing pressure due to water level in the suppression pool and the other measuring the pressure above the maximum expected water level.

The logic is arranged such that either transmitter and its associated trip unit can cause the suppression pool suction valves to open and the CST suction valves to close.

The ALLOWABLE VALUE for the Suppression Pool Water Level--High function is high enough that normal water level fluctuations do not cause cycling of the HPCI suction valves, but low enough to ensure that HPCI will be aligned for suction from the suppression pool before the water level reaches the point at which suppression pool design loads would be exceeded.

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BASES (continued)

LCO (continued)

The Suppression Pool Water Level--High function is required to be OPERABLE only when the HPCI ECCS is required to be OPERABLE. Thus, two channels of the HPCI Suppression Pool Water Level--High function are required to be OPERABLE in MODE 1, and in MODES 2 and 3 when reactor steam dome pressure is > 150 psig.

3.f. High Pressure Coolant Injection Pump Discharge Flow--Low (Bypass)

The minimum-flow interlock is provided to allow the associated ECCS pumps to initiate and protect the pump from overheating. The minimum-flow bypass valve is opened when low flow is sensed, and the valve is automatically closed at normal flow conditions.

One flow transmitter per ECCS pump is used to detect the associated systems' flow rates. The HPCI System has one transmitter. The logic is arranged so that the transmitter causes its associated minimum-flow valve to open or close.

The ALLOWABLE VALUE for minimum-flow interlock is high enough to ensure that the closure of the minimum-flow bypass valve is initiated to allow full flow into the core.

The MFIs are required to be OPERABLE when the HPCI ECCS subsystem is required to be OPERABLE. One channel of the HPCI pump MFI is required to be OPERABLE in MODE 1, and in MODES 2 and 3 when reactor steam dome pressure is > 150 psig.

3.g. Manual Initiation

The Manual Initiation push button switches introduce signals into the appropriate ECCS subsystem that are redundant to the automatic actuation signals and provide Manual Initiation capability. There are two switches for HPCI initiation.

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BASES (continued)

LCO (continued)

4. Automatic Depressurization System Trip Systems A and B

4.a. Reactor Vessel Water Level--Low Low Low, Level 1

Low reactor vessel water level indicates that the capability of cooling the fuel may be threatened. Should reactor vessel water level decrease too far, fuel damage could result. Reactor Vessel Water Level--Low Low Low, Level 1 signals are initiated from four level transmitters that are connected to relays whose contacts are arranged in a one-out-of-two taken twice logic.

The Reactor Vessel Water Level--Low Low Low, Level 1 ALLOWABLE VALUE is high enough to allow time for the low-pressure, core flooding systems to activate and provide adequate cooling, but low enough that decreases in the level resulting from operational transients will not cause spurious initiation.

The Reactor Vessel Water Level--Low Low Low, Level 1 function is required to be OPERABLE only when the ADS ECCS is required to be OPERABLE. For the ADS Level 1 function, the requirements are that two channels in each ADS trip system be OPERABLE during MODE 1, and during MODES 2 and 3 when reactor steam dome pressure is > 150 psig.

4.b. Drywell Pressure--High

High pressure in the drywell could indicate a break in the RCPB. All ECCS injection and spray subsystems are initiated on receipt of the Drywell Pressure--High function to minimize the possibility of fuel damage.

High drywell pressure signals are initiated from four pressure transmitters that sense the pressure at two different locations in the drywell. These transmitters are connected to relays whose contacts are arranged in a one-out-of-two taken twice logic.

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BASES (continued)

LCO (continued)

The ALLOWABLE VALUE selected is low enough to minimize heat loads on equipment located in the drywell, but not so low that it could cause spurious trips.

The Drywell Pressure--High function is required to be OPERABLE when the ADS ECCS is required to be OPERABLE in conjunction with the times when the primary containment is required to be OPERABLE.

Two channels of Drywell Pressure--High function in each ADS trip system are required to be OPERABLE in MODE 1, and in MODES 2 and 3 when reactor steam dome pressure is > 150 psig.

4.c. <u>Automatic Depressurization System Initiation Timer</u>

The purpose of the ADS Initiation Timer is to delay depressurization of the reactor vessel to give the HPCI System time to maintain reactor vessel water level. Because the rapid depressurization caused by ADS operation is one of the most severe transients on the reactor vessel, its occurrence should be limited. By delaying initiation of the ADS function, the operator is given the chance to monitor the success or failure of the HPCI System in maintaining water level and then decide whether to allow ADS to initiate, to delay initiation further by recycling the timer, or to inhibit initiation permanently. The manual ADS inhibit and delay circuits support the OPERABILITY of the ADS function. If either manual capability becomes inoperable, the associated ADS System "A" or "B" is inoperable.

There are two ADS Initiation Timer relays, one in each of the two ADS trip systems. The ALLOWABLE VALUE for the ADS Initiation Timer is long enough for the HPCI to maintain reactor vessel water level and for the operator to evaluate plant conditions, yet short enough that there is still time after depressurization for the low-pressure ECCS subsystems to provide adequate core cooling.

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BASES (continued)

LCO (continued)

The ADS Initiation Timer function is required to be OPERABLE only when the associated ADS is required to be OPERABLE. Thus, one channel of the ADS Initiation Timer function in each ADS trip system is required to be OPERABLE in MODE 1, and in MODES 2 and 3 when reactor steam dome pressure is > 150 psig.

4.d. Reactor Vessel Water Level--Low, Level 3 (Confirmatory)

The Reactor Vessel Water Level--Low, Level 3 signals are initiated from two level transmitters that are connected to relays whose contacts are arranged in a one-out-of-one logic per trip system. These taps are different from those used for the Level 1 measurement.

The ALLOWABLE VALUE for Reactor Vessel Water Level--Low, Level 3 is selected at the RPS Level 3 scram setpoint for convenience. Refer to LCO 3.3.1.1, "Reactor Protective System Instrumentation" for Bases discussion of this function.

The Reactor Vessel Water Level--Low, Level 3 function is required to be OPERABLE only when the ADS is required to be OPERABLE. Thus, one channel of the ADS Level 3 function in each ADS trip system is required to be OPERABLE in MODE 1, and in MODES 2 and 3 when reactor steam dome pressure is > 150 psig.

4.e. Core Spray Pump Discharge Pressure--High

The CS Pump Discharge Pressure--High signals are used as permissives for ADS initiation, indicating that there is a source of low-pressure cooling water available once the ADS has depressurized the vessel.

To generate an ADS permissive in one trip system, it is only necessary that one pump indicate the

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BASES (continued)

LCO (continued)

high-discharge-pressure condition. The relay contacts are arranged in a one-out-of-two logic per pump.

The ALLOWABLE VALUE for CS Pump Discharge Pressure--High is lower than that of the pump discharge pressure when the pump is operating in a full-flow mode, but high enough to avoid any condition that results in a discharge pressure permissive when the CS pumps are aligned for injection and the pumps are not running. The actual operating point of this function is not assumed in any transient or accident analysis.

The CS Pump Discharge Pressure--High function is required to be OPERABLE only when the associated ADS is required to be OPERABLE. Thus, two channels of CS Pump Discharge Pressure--High function are required to be OPERABLE in each ADS trip system in MODE 1, and in MODES 2 and 3 when the reactor steam dome pressure is > 150 psig.

4.f. Low Pressure Coolant Injection Pump Discharge Pressure--High

The LPCI Pump Discharge Pressure--High signals are used as permissives for ADS initiation, indicating that there is a source of low-pressure cooling water available once the ADS has depressurized the vessel.

Pump discharge pressure signals are initiated from eight pressure transmitters, two on the discharge side of each of the four low-pressure ECCS pumps. To generate an ADS permissive in one trip system, it is necessary that only one pump indicate the high-discharge-pressure condition. The pump relay contacts are arranged in a one-out-of-two per pump logic.

The ALLOWABLE VALUE for LPCI Pump Discharge Pressure--High is lower than that of the pump discharge pressure when the pump is operating in a full-flow mode, but high enough to avoid any condition that results in a discharge pressure

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BASES (continued)

LCO (continued)

permissive when the LPCI pumps are aligned for injection and the pumps are not running.

The LPCI Pump Discharge Pressure--High function is required to be OPERABLE only when the associated ADS is required to be OPERABLE. Thus, four channels of the LPCI Pump Discharge Pressure--High function are required to be OPERABLE in each ADS trip system in MODE 1, and in MODES 2 and 3 when the reactor steam dome pressure is > 150 psig.

4.g. Automatic Depressurization System Low Water Level Actuation Timer

There are four relays for the ADS Low Water Level Actuation Timer, two in each of the two ADS trip systems. The relay contacts are arranged in a two-out-of-two actuation logic.

The ALLOWABLE VALUE for the ADS timer is long enough for the HPCI to restore reactor vessel water level and for the operator to evaluate plant conditions, yet short enough that there is still time after depressurization for the low-pressure ECCS subsystems to provide adequate core cooling.

The ADS Low Water Level Actuation Timer function is required to be OPERABLE only when the associated ADS is required to be OPERABLE. Thus, two channels of the ADS Low Water Level Actuation Timer function in each ADS trip system are required to be OPERABLE in MODE 1, and in MODES 2 and 3 when the reactor steam dome pressure is > 150 psig.

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

[For this facility, those required support systems which, upon their failure do not result in ECCS instrumentation being declared inoperable, and their justification, are as follows:]

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BASES (continued)

LCO (continued)

[For this facility, the supported systems impacted by the inoperability of the ECCS instrumentation and the justification for whether or not each supported system is declared inoperable are as follows:]

APPLICABILITY

The individual functions are required to be OPERABLE only in the MODES or other specified conditions, as defined in Table 3.3.5.1-1, that may require ECCS initiation to mitigate the consequences of a DBA or transient. The Bases for Applicability requirements are discussed on a function-by-function basis in the LCO section.

A Note is added to provide clarification that for this LCO, each function specification in Table 3.3.5.1-1 shall be treated as an independent entity with an independent Completion Time.

ACTIONS

In order for a facility to take credit for topical reports for the basis for justifying Completion Times, the topical reports should be supported by an NRC staff SER that establishes the acceptability of each topical report for that facility.

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 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 a 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 3.3.5.1-1, the channel must be declared inoperable immediately and the appropriate Conditions from Table 3.3.5.1-1 must be entered immediately.

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BASES (continued)

ACTIONS (continued)

In the event a channel's trip setpoint is found nonconservative with respect to the ALLOWABLE VALUE or the transmitter, instrument loop, signal processing electronics, or bistable 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.

Condition A

Condition A applies to each of the ECCS initiation functions in Table 3.3.5.1-1.

A.1

Required Action A.1 addresses the situation where one or more channels for one or more functions are inoperable at the same time. The Required Action is to refer to Table 3.3.5.1-1 and to take the Required Actions for the protection functions affected. The Completion Times are those from the referenced Conditions and Required Actions.

Condition B

Condition B applies to the following functions: Reactor Vessel Water Level--Low Low Low, Level 1; Reactor Vessel Water Level--Low Low, Level 2; Reactor Vessel Shroud Level--Level 0 in MODES 4 and 5 when the associated ECCS subsystem is required to be OPERABLE; Drywell Pressure; and Reactor Steam Dome Pressure (injection permissive) in MODES 4 and 5 when the associated ECCS subsystem is required to be OPERABLE.

B.1, B.2.1, B.2.2, and B.2.3

Required Action B.1 verifies that each function has the capability to initiate the associated subsystems. This Required Action includes a Note that states that the associated supported subsystems are declared inoperable immediately if there is a loss of functional capability in the ECCS instrumentation function concerning the initiation of the associated subsystems. The Completion Time of 1 hour is sufficient for plant operations personnel to make this verification.

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BASES (continued)

ACTIONS (continued)

Required Actions B.2.1, B.2.2, or B.2.3 must be met; otherwise, at the end of 24 hours the plant must be placed in a MODE for which the function is no longer required to be OPERABLE, by LCO 3.0.3.

B.2.1 is the preferred action because it restores the full functional capability of the ECCS. The Completion Time of 24 hours is provided to permit restoration of any inoperable channel(s) to OPERABLE status and is justified by the reliability analysis described by Reference 3. If the inoperable channel(s) cannot be restored within the Completion Time, the channel(s) must be placed in the tripped condition (B.2.2), which performs the intended function of the channel or the associated subsystems must be declared inoperable (B.2.3) and the Required Actions of the affected subsystems initiated (LCO 3.5.1 and LCO 3.5.2).

Required Action B.2.2 is modified by a Note to indicate that this Required Action is only applicable if placing inoperable channel(s) in trip would not result in an initiation.

Condition C

Condition C applies to the following functions: Reactor Steam Dome Pressure--Low (injection permissive) and (recirculation discharge valve permissive) in MODES 1, 2, and 3; Manual Initiation functions; Reactor Vessel Shroud Level--Level 0 in MODES 1, 2, and 3; LPCI Pump Start--Time Delay Relay; and Reactor Vessel Water Level--High, Level 8.

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

Required Action C.1 verifies that each function has the capability to initiate the associated subsystems. This Required Action includes a Note that states that the associated supported subsystems are declared inoperable immediately if there is a loss of functional capability in the ECCS instrumentation function concerning the initiation of the associated subsystems. The Completion Time of 1 hour is sufficient for plant operations personnel to make this verification.

Certain functions, LPCI Pump Start--Time Delay, injection permissive, Manual Initiation, and Reactor Vessel Water

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BASES (continued)

ACTIONS (continued)

Level--High, Level 8 do not allow for placing the channel in trip. Placing a channel in trip would either cause the initiation or would not necessarily result in the safe condition for the channel in all events. If the inoperable channel(s) cannot be restored per Required Action C.2.1, then the associated subsystem must be declared inoperable by Required Action C.2.2 and the Required Actions of the affected systems must be initiated (LCO 3.5.1 and LCO 3.5.2).

The 24-hour Completion Time is acceptable (Ref. 3) to permit restoration of any channel(s) inoperable to OPERABLE status.

Condition D

Condition D applies to the HPCI CST level and suppression pool water level functions.

D.1, D.2.1, D.2.2, D.2.3, and D.2.4

Required Action D.1 verifies that each function has the capability to initiate the associated subsystems. This Required Action includes a Note that states that the associated supported subsystems are declared inoperable if there is a loss of functional capability in the ECCS instrumentation function concerning the initiation of the associated subsystems. The Completion Time of 1 hour is sufficient for plant operations personnel to make this verification.

Required Action D.2.1 is the preferred action because it restores full functional capability of the HPCI initiation. The Completion Time of 24 hours is provided to permit restoration of any inoperable channel(s) to OPERABLE status. If the inoperable channel(s) cannot be restored in the Completion Time, the channel(s) must be placed in the tripped condition (Required Action D.2.2), which performs the intended function of the channel, or the associated subsystem must be declared inoperable (Required Action D.2.4) and Required Actions of the affected systems initiated (LCO 3.5.1 and LCO 3.5.2).

For HPCI, CST Level--Low and Suppression Pool Water Level--High functions, Required Action D.2.3 allows, as an alternative, the alignment of the HPCI suction to the

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BASES (continued)

ACTIONS (continued)

suspression pool. If this Required Action is performed, measures should be taken to ensure that the HPCI System piping remains filled with water.

Reference 3 provides a 24-hour Completion Time as an acceptable basis for the HPCI instrumentation.

Condition E

Condition E applies to the Core Spray LPCI and HPCI Pump Discharge Flow Low (Bypass) functions.

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

Required Action E.1 verifies that each function has the capability to initiate the associated subsystems. This Required Action includes a Note which states that the associated supported subsystems are declared inoperable immediately if there is a loss of functional capability in the ECCS instrumentation function concerning the initiation of the associated subsystems. The Completion Time of 1 hour is sufficient for plant operations personnel to make this verification.

If the instrumentation that controls the pump minimum-flow valve is inoperable, extended pump operation with no injection path available could lead to pump overheating and failure. If there were a failure of the instrumentation, a portion of the pump flow could be diverted from the reactor vessel injection path, causing insufficient core cooling. These consequences can be averted by the operator's manual control of the valve, which would be adequate to maintain ECCS pump OPERABILITY. Furthermore, the 7 days is justifiable because other ECCS pumps will be sufficient to complete the assumed safety function if no additional single failure were to occur. The Completion Time to restore or declare the associated ECCS System inoperable is reasonable based on the remaining capability of the associated ECCS subsystems, the redundancy available in the ECCS design, and the low probability of a DBA occurring during the repair time.

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BASES (continued)

ACTIONS (continued)

Condition F

Condition F applies to each one of the ADS functions in Table 3.3.5.1-1.

F.1, F.2.1, F.2.2, and F.2.3

Required Action F.1 verifies that each function has the capability to initiate the associated subsystems. This Required Action includes a Note which states that the associated supported subsystems are declared inoperable immediately if there is a loss of functional capability in the ECCS instrumentation function concerning the initiation of the associated subsystems. The Completion Time of 1 hour is sufficient for plant operations personnel to make this verification.

The preferred Required Action is F.2.1 because it restores full functional capability to the ADS. The 24-hour Completion Time is acceptable (Ref. 3) to permit restoration of any inoperable channel(s) to OPERABLE status. If the inoperable channel(s) cannot be restored, the inoperable channel(s) must be placed in trip (Required Action F.2.2), which performs the channel's intended function. Required Action F.2.2 is modified by a Note that, if placing the channel(s) in trip would result in an actuation, the Required Action is not applicable.

Required Action F.2.3 provides the option to declare the associated subsystem(s) inoperable if the channel(s) cannot be restored to OPERABLE status or placed in trip in the 24-hour Completion Time of Reference 3. Under this Required Action the corresponding LCOs are entered to accomplish the Required Actions.

Condition G

Condition G is applicable to each one of the ECCS Instrumentation functions presented in Table 3.3.5.1-1.

G.1 and G.2

Required Action G.1 verifies that the Required Actions have been initiated for those supported systems declared inoperable because of the inoperability of the support

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BASES (continued)

ACTIONS (continued)

channel(s) within a Completion Time of 1 hour. The specified Completion Time is sufficient for plant operations personnel to make this verification.

Required Action G.1 ensures that those identified Required Actions associated with supported systems impacted by the inoperability of channel(s) associated with each ECCS instrumentation 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 G of this LCO.

[For this facility, the identified supported systems Required Actions associated with each ECCS instrumentation function are as follows:]

Required Action G.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 verification. 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 ECCS instrumentation function are found in the SRs column of Table 3.3.5.1-1 for that function. Most functions are subject to CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, CHANNEL CALIBRATION, and some are subject to ECCS RESPONSE TIME testing.

In order for a facility 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.

The Surveillances are modified by a Note to indicate that a channel may be placed in an inoperable status for up to 6 hours for required Surveillance without placing the trip system in trip, provided at least one OPERABLE channel in

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

the same trip system is monitoring the parameter. Upon completion of the Surveillance, or expiration of the 6-hour allowance (Ref. 3), the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken.

It is not acceptable to routinely remove channels from service for more than 6 hours to perform required Surveillance testing. Such a practice would be contrary to the assumptions of the reliability analysis which justified the LCO Completion Times.

SR 3.3.5.1.1

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 the instrument channels could be an indication of excessive instrument drift in one of the channels or something even more serious. CHANNEL CHECK will detect gross channel failure, thus it is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff, based on a 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 Surveillances are 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 outright channel failure is

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

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 these channels required by the LCO.

SR 3.3.5.1.2

A CHANNEL FUNCTIONAL TEST verifies the function of the trip, interlock, and alarm functions of the channel. The inserts a simulated or actual signal as close to the sensor as practicable and verifies required trip, interlocks, and alarms function when the input is beyond the trip point. The "as found" and "as left" values for bistable trip setpoints are recorded. Bistable setpoints must be found within the ALLOWABLE VALUES 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 plant-specific 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 92-day Surveillance Frequency of SR 3.3.5.1.2 is based on the reliability analyses of Reference 3.

SR 3.3.5.1.3

Calibration of trip units consists of a test to determine actual trip setpoints. Trip setpoints are adjusted if found outside of the acceptable "as left" tolerance. If, during trip unit calibration, the associated trip setting is discovered to be less conservative than the specified ALLOWABLE VALUE, the channel must be declared inoperable. Measurement and setpoint error determination and readjustment must be performed consistent with the assumptions of the plant-specific setpoint analysis.

The Surveillance Frequency of 92 days is based on the assumptions in the methodology included in the determination

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

of the trip setpoint. SR 3.3.5.1.3 and SR 3.3.5.1.4 are often performed simultaneously using a common procedure.

SR 3.3.5.1.4

The CST level instruments are calibrated every 92 days. CHANNEL CALIBRATION is a complete check of the instrument channel, including the detector. The test verifies that the channel responds to measured parameter values 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 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.

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 a 92-day calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

SR 3.3.5.1.5

CHANNEL CALIBRATIONS are performed every 18 months or at approximately every refueling and are based on the determination of the magnitude of equipment drift in the

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

setpoint analysis. CHANNEL CALIBRATION was previously discussed in SR 3.3.5.1.4.

SR 3.3.5.1.6

The 18-month LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required actuation logic for a specific channel. The LOGIC SYSTEM FUNCTIONAL TEST tests all logic components (i.e., all relays and contacts, trip units, solid-state logic elements, etc.) of a logic circuit, from sensor up to the actuated device. The system functional testing performed in LCO 3.5.1, "ECCS—Operating" and LCO 3.5.2, "ECCS—Shutdown" overlaps this test to complete testing of the assumed safety function. The 18-month Frequency was developed considering it is prudent that the Surveillance only be performed during 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 at the 18-month Frequency.

SR 3.3.5.1.7

SR 3.3.5.1.7 is a CHANNEL FUNCTIONAL TEST performed on the ECCS manual initiation functions. This test verifies that the initiation push buttons are capable of providing the initiation function.

The 18-month Frequency was developed considering it is prudent that the Surveillance be performed only during plant outage. This is 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 that these components usually pass the Surveillance when performed at the 18-month Frequency.

SR 3.3.5.1.8

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

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

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 given in Table B 3.3.5.1-1, following this specification. [For this facility, the response times include contributions from the following:] This test may be performed in one measurement or in overlapping segments, with verification that all components are tested.

Response time tests are conducted on a 18-month STAGGERED TEST BASIS. This results in the interval between successive tests of a given channel per trip system of n times 18 months, where n is the number of channels in the function. Testing of the final actuation devices in a trip system, 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 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 because equipment operation is required. The response times for the applicable ECCS subsystems are the maximum values assumed in the safety analyses.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix A, General Design Criterion 21, "Protection System Reliability and Testability."

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

3. NEDC-30936-P-A, "BWR Owners' Group Technical Specification Improvement Analyses for ECCS Actuation Instrumentation, Part 2," December 1988.

4. [Unit Name] FSAR, Section [ ], "[Plant-Specific Setpoint Methodology]."

5. [Unit Name] FSAR, Section [7], "[Instrumentation and Controls]."

6. Title 10, Code of Federal Regulations, Part 50.49, "Environmental Qualifications of Electric Equipment Important to Safety for Nuclear Power Plants."

Table B 3.3.5.1-1 (page 1 of 1)
EMERGENCY CORE COOLING SYSTEM RESPONSE TIMES

Emergency Core Cooling System	Response Time (seconds)
1. Core Spray System	≤ [34]
2. LPCI System	≤ [64]
3. HPCI System	≤ [30]
4. ADS	N/A

B 3.3 INSTRUMENTATION

B 3.3.5.2 Reactor Core Isolation Cooling (RCIC) System Actuation Instrumentation

BASES

BACKGROUND

The purpose of the Reactor Core Isolation Cooling (RCIC) System is to provide coolant makeup to the reactor vessel to ensure adequate core cooling when the reactor vessel is isolated from its primary heat sink (the main condenser) and normal coolant makeup flow from the reactor feedwater system is unavailable. The RCIC instrumentation may also initiate operation of the system to aid the High Pressure Core Injection (HPCI) System in providing core cooling to help mitigate the consequences of accidents and transients and during reactor shutdown prior to depressurizing the RCS and initiating shutdown cooling. A more complete discussion of RCIC operation is provided in the Bases of LCO 3.5.1, "ECCS—Operating" and LCO 3.5.2, "ECCS—Shutdown."

ALLOWABLE VALUES, in conjunction with the LCO, provide assurance that the RCIC System will be OPERABLE and perform its design functions which are important to safety.

The RCIC System is a single train system and, therefore, is not designed to withstand certain single failures; however, portions of the actuation instrumentation and the control logic are redundant to prevent single measurement channel or bistable failures from compromising the performance of the RCIC function. Redundant portions of the actuation instrumentation are typically powered from a single electrical supply. The RCIC System uses a single, steam turbine-driven pump that can deliver water to the reactor vessel from either the condensate storage tank (CST) or the suppression pool.

The RCIC instrumentation consists of measurement channels, bistables, actuation logic, actuation devices, and actuated equipment which is described below.

Measurement channels, consisting of field transmitters and associated signal processing, provide measurable electronic output signals based on the physical characteristics of the

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BASES (continued)

BACKGROUND (continued)

parameter being measured. Each measurement channel provides an analog input signal to an associated bistable trip unit and may also provide inputs to control-room indicators, recorders, alarms, and the plant computer. The redundant measurement channels for each monitored parameter are identical in design. Each bistable trip unit, mounted in an instrument cabinet in the control room, receives an analog input signal from its measurement channel in the field, compares the analog input to its corresponding setpoint value, and actuates a trip output relay when the input signal exceeds the setpoint value. Contacts from the bistable trip-unit output relays are used to make up the RCIC System actuation logic. The bistable trip units also provide local status indication and remote annunciation. When the actuation logic is satisfied, actuation relays are energized to initiate the appropriate equipment response. The specific monitored parameters, logic, and actuated equipment for the RCIC System are discussed below.

1. <u>Reactor Vessel Water Level--Low Low, Level 2</u>

Low reactor vessel water level indicates that normal feedwater flow is insufficient to maintain desired reactor vessel water level, and that the capability to cool the fuel may be threatened. Should reactor vessel water level decrease too far, fuel damage could result. Therefore, the RCIC System is automatically initiated at Level 2 to assist in maintaining water level above the top of the active fuel.

Reactor vessel water level signals are initiated from four level transmitters which sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel.

Contacts from the bistable trip-unit output relays associated with the four Reactor Vessel Water Level--Low Low, Level 2 measurement channels, which are used to automatically initiate the RCIC System are arranged in a one-out-of-two taken-twice logic configuration. This arrangement allows any one channel to be placed in the tripped condition during surveillance testing, without causing unwanted system actuations, and prevents the failure of a single measurement channel or bistable from precluding system operation.

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BASES (continued)

BACKGROUND (continued)

Actuation of RCIC System equipment to automatic initiation signals includes the following:

- The pump suction valve from the CST is signaled to open;
- The test/return line valves are signaled to close;
- The turbine steam inlet valve is signaled to open; and
- The pump discharge valve to the reactor vessel is signaled to open.

The RCIC System can be reset if the reactor water level has been restored. The RCIC pump can then be stopped and the system realigned for subsequent automatic initiation. Automatic restart will then occur if the Reactor Vessel Water Level--Low Low, Level 2 condition reoccurs.

2. Reactor Vessel Water Level--High, Level 8

High reactor vessel water level indicates that sufficient cooling water inventory exists in the reactor vessel so that there is no danger to the fuel. Therefore, the Level 8 signal is used to close the RCIC steam supply, steam supply bypass, and injection valves to prevent overflow of the coolant into the main steam lines.

Reactor Vessel Water Level--High, Level 8 signals for RCIC are initiated from two level transmitters from the wide-range, water level measurement system. Both Level 8 signals are required in order to close the RCIC steam supply, the steam bypass, and the injection valves. A two-out-of-two logic is used for high level termination of RCIC flow to prevent a single measurement channel or bistable trip unit failure from causing premature termination. The RCIC System automatically restarts if a Reactor Vessel Water Level--Low Low, Level 2 signal is subsequently received.

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BASES (continued)

BACKGROUND (continued)

3. Condensate Storage Tank Level--Low

The RCIC instrumentation also monitors the water levels in the CST and the suppression pool because these are the two sources of water for RCIC operation.

Reactor-grade water in the CST is the normal source. A low level in the CST indicates the unavailability of an adequate supply of makeup water from this normal source. On receipt of an RCIC initiation signal, the CST suction valve is automatically signaled to open even though it is normally in the open position. If the water level in the CST falls below a preselected level, however, first pump suction valves to the suppression pool automatically open, and then the CST suction valve automatically closes. This ensures that an adequate supply of makeup water is available to the RCIC pump. To prevent losing suction to the pump, the suction valve are interlocked so that one suction path must be open before the other automatically closes.

Two level transmitters are used to detect low water level in the CST. The logic is arranged so that either transmitter and associated trip unit can cause the suppression pool suction valves to open and the CST suction valve to close.

The suppression pool suction valves also automatically open and the CST suction valve closes if high water level is detected in the suppression pool.

4. Suppression Pool Water Level--High

An excessively high suppression pool water level could result in the loads on the suppression pool exceeding design values should there be a blowdown of the reactor vessel pressure through the safety/relief valves (S/RVs). Therefore, signals indicating high suppression pool water level are used to transfer the RCIC suction source from the CST to the suppression pool to eliminate the possibility of the RCIC System continuing to provide additional water from a source outside containment.

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BASES (continued)

BACKGROUND (continued)

Suppression pool water level signals are initiated from two level transmitters that each measure the pressure difference between two sealed diaphragms: one senses pressure due to water level in the suppression pool and the other measures the pressure above the maximum expected water level. The two transmitters are located at widely separated points in the suppression pool. The logic is arranged so that either transmitter and its associated trip unit can cause the suppression pool suction valves to open and the CST suction valve to close.

5. Manual Initiation

The manual initiation capability is provided by a single push-button switch that introduces signals into the RCIC System actuation circuits to cause the same equipment response as for automatic initiation.

APPLICABLE SAFETY ANALYSES

The RCIC System is used to provide makeup coolant to the reactor in response to transient events. Automatic operation of the RCIC System, in addition to the HPCI System, provides capability to inject water into the reactor vessel under high-pressure conditions to reduce vessel pressure and increase vessel water level for transient mitigation and for prevention of unnecessary challenges to the following ECCS Systems: Automatic Depressurization System (ADS), Low Pressure Coolant Injection (LPCI) System, and Low Pressure Core Spray (LPCS) System.

The RCIC System is retained in Technical Specifications because operating experience has shown that it is an important adjunct to the HPCI System for transient mitigation and thus is important to risk reduction.

LCO

The LCO requires OPERABILITY of all RCIC instrumentation needed to provide adequate assurance of successfully accomplishing the RCIC function to mitigate transients without challenging ECCS systems, even given failure of the HPCI System. The OPERABILITY of the individual RCIC

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BASES (continued)

LCO (continued)

instrumentation functions is confirmed through successful completion of required surveillance testing. Individual instrument channels (i.e., measurement channels and the associated bistable trip units) are considered OPERABLE when the following conditions are satisfied:

1. All channel components necessary to provide an actuating signal are functional and in service;

2. Channel measurement uncertainties are known (via test, analysis, or design information) to be within the assumptions of the setpoint calculations; and

3. Required surveillance testing is current and has demonstrated performance within each surveillance test's acceptance criteria.

Only the ALLOWABLE VALUES are specified for each RCIC actuation 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 measurements obtained 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 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 (Ref. 1).

Bases for the ALLOWABLE VALUES are as follows:

1. The Reactor Vessel Water Level--Low Low, Level 2, ALLOWABLE VALUE is set high enough that for complete loss of feedwater flow, the RCIC System flow with HPCI assumed to fail, will be sufficient to avoid initiation of the low-pressure ECCS at Level 1. This ALLOWABLE VALUE is also low enough that, after a scram caused by a Level 3 trip with no loss of feedwater flow, the RCIC and HPCI Systems will not be initiated.

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BASES (continued)

LCO (continued)

2. The Reactor Vessel Water Level--High, Level 8, ALLOWABLE VALUE is set high enough to preclude isolating the injection valve of the RCIC System during normal operation, yet low enough to trip the RCIC System prior to water overflowing into the main steam lines.

3. The Condensate Storage Tank Level--Low function ALLOWABLE VALUE is set high enough to ensure adequate pump suction head while water is being taken from the CST, and low enough not to cause spurious cycling of the suction valves due to normal tank level fluctuations.

4. The Suppression Pool Water Level--High function ALLOWABLE VALUE is set high enough that normal water level fluctuations do not cause cycling of the RCIC suction valves, but low enough to ensure that the RCIC will be aligned to take suction from the suppression pool before the water level reaches the point at which suppression pool design loads would be exceeded.

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

[For this facility, those required support systems which, upon their failure do not result in the RCIC instrumentation being declared inoperable and their justification are as follows:]

[For this facility, the supported systems impacted by the inoperability of the RCIC instrumentation and the justification of whether or not each supported system is declared inoperable are as follows:]

APPLICABILITY

The individual functions are required to be OPERABLE in MODE 1 and in MODES 2 and 3 when reactor steam dome pressure is > [150] psig. RCIC System operation is not needed to provide core cooling in MODE 3 with reactor steam dome pressure < [150] psig, because the shutdown cooling mode of

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BASES (continued)

APPLICABILITY (continued)

the RHR System is in service and is relied on to provide core cooling, and steam supply pressure to the RCIC turbine is not sufficient to establish pump OPERABILITY.

A Note is added to provide clarification that, for this LCO, each function specified in Table 3.3.5.2-1 shall be treated as an independent entity with an independent Completion Time.

ACTIONS

In order for a facility to take credit for topical reports for the basis for justifying Completion Times, topical reports should be supported by an NRC staff Safety Evaluation Report which establishes the acceptability of each topical report for that facility.

A protection function 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 in Table 3.3.5.2-1, the channel must be declared inoperable immediately, and the appropriate Conditions from Table 3.3.5.2-1 must be entered immediately.

In the event a channel's trip setpoint is found to be non-conservative with respect to the ALLOWABLE VALUE, or the transmitter, instrument loop, signal-processing electronics, or bistable 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.

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BASES (continued)

ACTIONS (continued)

Condition A

Condition A applies to each one of the functions listed in Table 3.3.5.2-1.

Required Action A.1 directs entry into all other Conditions referenced in Table 3.3.5.2-1. The applicable Condition specified in the table is function dependent. Each time an inoperable channel is discovered, Condition A is entered and provides for transfer to the appropriate subsequent Condition.

Condition B

Condition B applies to the Reactor Vessel Water Level--Low, Level 2 function.

Required Action B.1 ensures that each function has not lost the capability to initiate RCIC System components because of the number of inoperable channels. A random failure of an OPERABLE channel does not have to be assumed in making this determination. For example, this Required Action can be acceptably completed with three Level 2 channels inoperable if the trip system containing the two inoperable channels is in trip. A Note modifies this Required Action to indicate that in the event the function capability is not established, the RCIC System must be declared inoperable.

Because the RCIC initiation logic on Reactor Vessel Water Level--Low Low, Level 2 is one-out-of-two taken twice, if one initiation instrument channel becomes inoperable, a single failure of a remaining channel could prevent RCIC System actuation. Required Action B.2.1 to restore the affected channel to OPERABLE status within one hour is the preferred action because it restores full-functional capability of the RCIC automatic Level 2 initiation function. Required Action B.2.2, which requires placing the affected channel in the tripped condition within 1 hour, effectively changes the automatic actuation logic to one-out-of-two and thereby eliminates the vulnerability to single instrument channel failures created by an inoperable channel. Required Action B.2.2 is modified by a Note that the Required Action does not apply if it results in an initiation. If a channel in one trip system becomes

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BASES (continued)

ACTIONS (continued)

inoperable when one or more channels in the opposite trip system are already in trip, placing the inoperable channel in trip will cause an initiation. Required Action B.2.2 is not intended to force an unnecessary initiation. In this event, Required Action B.2.1 would have to be met. If the inoperable channel(s) are not restored to OPERABLE status or placed in trip within the time allowed, Required Action B.2.3 applies.

Required Action B.2.3 declares the supported RCIC System inoperable within 1 hour. This requires that the Required

Actions of LCO 3.5.1 be met. The Completion Time of 1 hour is sufficient for accomplishing the corrective actions.

Condition C

Condition C applies to the Reactor Vessel Water Level--High, Level 8 function.

Required Action C.1 ensures that each function has not lost the capability to initiate RCIC System components because of the number of inoperable channels. For the high level function, this Action can be completed with a single inoperable channel only if the inoperable channel is placed in trip. Required Action C.1 cannot be completed with two inoperable high level channels. A Note modifies this Required Action to indicate that in the event the functional capability is not established, the RCIC System must be declared inoperable.

Because the logic for termination of RCIC flow on reactor vessel high Level 8 is two-out-of-two and because placing an inoperable channel in the tripped condition does not result in the safe state for the channel in all events, Required Action C.2.1 requires restoration of the affected channel within 1 hour. This is the preferred Action because it restores full functional capability of the RCIC function. Alternatively, Required Action C.2.2 is entered, which requires initiation of the Required Actions of LCO 3.5.1, "ECCS--Operating" to restore the RCIC to OPERABLE status or place the reactor in a condition for which RCIC operation is not needed. Failure to restore the affected channel to OPERABLE status creates a situation in which a failure of

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BASES (continued)

ACTIONS (continued)

the remaining channel could result in failure to terminate RCIC flow when required. This could lead to water overflowing into the main steam lines. This overflows could potentially adversely affect the OPERABILITY of safety-related equipment (e.g., HPCI pump turbine, ADS, and S/RVs). The 1-hour Completion Time is considered a reasonable amount of time to ensure RCIC functional capability and to complete the corrective Actions to either restore RCIC OPERABILITY or declare the RCIC System inoperable.

Condition D

Condition D applies to the CST and suppression pool water levels.

Required Action D.1 ensures that each function has not lost the capability to initiate RCIC System components because of the number of inoperable channels. This Action cannot be achieved if any channel of CST or suppression pool level is in trip. This Action can also be completed with no tripped channels if one channel of suppression pool level and one channel of CST level are OPERABLE. A Note modifies this Required Action to indicate that in the event the functional capability is not established, the RCIC System must be declared inoperable.

Both the RCIC CST Level--Low and the Suppression Pool Water Level--High functions cause the automatic transfer of RCIC pump suction from the CST to the suppression pool. Either Required Action D.2.1 to restore the affected channel to OPERABLE status, or Required Action D.2.2 to place the affected channel in the tripped condition, or Required Action D.2.3 to align RCIC suction to the suppression pool are required to be completed within 1 hour if a channel becomes inoperable. If Action D.2.3 is performed, measures should be taken to ensure that the RCIC System piping remains filled with water. Performing any one of these Actions establishes a safe valve lineup for RCIC suction, because the suppression pool will be aligned or the required channel redundancy will be restored. Otherwise, Required Action D.2.4 is required, which declares the RCIC System

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BASES (continued)

ACTIONS (continued)

inoperable and requires the Required Actions of LCO 3.5.1, "ECCS—Operating." The 1-hour Completion Time is sufficient for operations personnel to ensure the RCIC Systems functional capability and to complete either of the corrective Actions to restore OPERABILITY, place the channel in trip, or transfer suction to the suppression pool to ensure the availability of a suction source.

Condition E

Condition E applies to Manual Initiation.

Required Action E.1 ensures that each function has not lost the capability to initiate RCIC System components because of the number of inoperable channels. A Note modifies this Required Action to indicate that in the event the function capability is not established, the RCIC System must be declared inoperable. The Completion Time of 1 hour is sufficient for the operator to ensure this capability.

Because of the redundancy of sensors available to provide initiation of the RCIC System and because manual actuation can be accomplished at the component level in addition to the system-level, manual-initiation push button, a Completion Time of 8 hours is provided to permit restoration of the manual initiation function to OPERABLE ACTIONS status. If the RCIC channel(s) are not restored to OPERABLE status, then Required Action E.2.2 requires the RCIC System to be declared inoperable. [For this facility, the basis for the 8-hour Completion Time is as follows:]

Condition F

Condition F is applicable to each one of the RCIC functions presented in Table 3.3.5.2-1.

Required Action F.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 verification.

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BASES (continued)

ACTIONS (continued)

Required Action F.1 ensures that those identified Required Actions associated with supported systems impacted by the inoperability of channel(s) associated with each RCIC instrumentation 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 F is entered. [For this facility, the identified supported systems Required Actions associated with each RCIC instrumentation function are as follows:]

Required Action F.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 verification. If verification determines the loss of function 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 RCIC function are found in Table 3.3.5.2-1. Most RCIC functions are subject to CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION.

A Note indicates that a channel may be placed in an inoperable status for up to 2 hours for the required Surveillance without placing the trip system in trip, provided at least one OPERABLE channel in the same trip system is monitoring that parameter. Upon completion of the Surveillance, or expiration of the 2-hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken. [For this facility, basis for the 2-hour allowance is as follows:]

SR 3.3.5.2.1

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.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the instrument channels could be an indication of excessive instrument drift in one of the channels or something even more serious. CHANNEL CHECK will detect gross channel failure, thus it is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff based on a 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 the Surveillances are 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 are 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 outright channel failure is limited to 12 hours. Because the probability of 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 the channels required by the LCO.

The Surveillance requirement includes verifying the functional capability of the CST suction valve position interlock for automatic swapover to the suppression pool on low CST water level.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.5.2.2

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 the 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. Bistable setpoints must be found to be within the ALLOWABLE VALUES 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 of Reference 1. 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 Surveillance Frequency of every 31 days is considered reasonable based on the reliability of the components and on operating experience, which has demonstrated that failure of more than one channel during the 31-day interval is unlikely.

SR 3.3.5.2.3

The calibration of bistable trip units consists of a test to determine the actual trip setpoints, and recalibration of the setpoint is necessary to ensure that it remains more conservative than the setpoint and the ALLOWABLE VALUE specified in Table 3.3.5.2-1. The channel must be declared inoperable if the setting is discovered to be less conservative than the ALLOWABLE VALUE. If the trip setting is discovered to be less conservative than the setting accounted for in the appropriate setpoint methodology but is not beyond the ALLOWABLE VALUE, the channel is still considered OPERABLE. Under these conditions, the setpoint must be readjusted to be more conservative than accounted for in the appropriate setpoint methodology.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

The Surveillance Frequency of every 31 days for SR 3.3.5.2.3 is based on the assumption of a 31-day calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis. SR 3.3.5.2.2 and SR 3.3.5.2.3 are often performed simultaneously using a common procedure.

SR 3.3.5.2.4

CHANNEL CALIBRATION is a complete check of the instrument channel including the detector. The test verifies the channel responds to measured parameter values 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 re-adjustments 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. 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 a 92-day calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

SR 3.3.5.2.5

A description of a CHANNEL CALIBRATION is given in SR 3.3.5.2.4. The Surveillance Frequency for this SR is 18 months.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

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 3.3.5.2.6

Performance of a LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required actuation logic. The LOGIC SYSTEM FUNCTIONAL TEST tests all logic components (e.g., all relays and contacts, trip units, solid-state logic elements, etc.) of a logic circuit, from the sensor up to the actuated device. The system functional testing performed in LCO 3.5.1 overlaps this test to provide complete testing of the safety function. The 18-month Frequency was developed considering it is prudent that the Surveillance be performed only during 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 at the 18-month Frequency.

[SR 3.3.5.2.7]

The CHANNEL FUNCTIONAL TEST for the Manual Initiation function confirms that the RCIC equipment functions as designed in response to a system-level, manual-initiation signal. The 18-month Surveillance Frequency for SR 3.3.5.2.7 is considered reasonable based on redundant, manual-initiation capability provided at the component level, the high reliability of the Manual Initiation function, and the desirability of performing the test during a plant refueling outage so as not to disrupt reactor operation or put additional stress cycles on plant equipment when the reactor is at operating temperatures. Operating experience has shown these components usually pass the surveillance when performed at the 18-month Frequency.

REFERENCES

1. [Unit Name], "[Setpoint Methodology]."

B 3.3 INSTRUMENTATION

B 3.3.6.1 Primary Containment Isolation (PCI) Instrumentation

BASES

BACKGROUND

The PCI instrumentation automatically initiates closure of appropriate isolation valves, which are necessary to prevent or limit the release of fission products from the reactor coolant system and the primary containment in the event of a loss-of-coolant accident or a reactor coolant pressure boundary (RCPB) leakage.

The isolation instrumentation includes the sensors, relays, and switches that are necessary to cause initiation of primary containment and RCPB system isolation. Functional diversity is provided by monitoring a wide range of dependent and independent parameters. The input parameters to the isolation logics are: reactor vessel water level; area ambient and differential temperatures; flow measurement; Standby Liquid Control System (SLCS) initiation; condenser vacuum; main steam line pressure; high pressure coolant injection (HPCI); reactor core isolation cooling (RCIC); steam line flow; main steam line and drywell radiation. Redundant sensor input signals from each parameter provided are for initiation of isolation. The only exception is SLCS initiation.

The PCI instrumentation is designed to include the three subsystems identified below:

- Field transmitters or process sensors;
- Signal processing and bistable modules; and
- Trip logic, setpoints, and ALLOWABLE VALUES.

Field Transmitters or Process Sensors

Field transmitters or process sensors provide a measurable electronic output signal based on the physical characteristics of the parameter being measured.

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BASES (continued)

BACKGROUND (continued)

Typically four measurement channels with physical separation are provided for each parameter. These are typically organized into two trip systems which are physically and electrically separated. Four measurement channels are necessary to meet the redundancy and testability of GDC 21 in Appendix A to 10 CFR 50 (Ref. 1) and to implement the one-out-of-two taken twice logic arrangement discussed for the PCI instrumentation.

[For this facility, a discussion of those PCI parameters that do not have four measurement channels and their conformance to redundancy and testability requirements of GDC 21 in Appendix A to 10 CFR 50 is as follows:]

For most anticipated operational occurrences (AOOs) and design basis accidents (DBAs) a wide range of dependent and independent parameters are monitored.

Signal Processing and Bistable Modules

Each process parameter measurement channel includes electronic equipment which provides signal conditioning, comparable output signals for main control board instruments, comparison of measured input signals with setpoints established by safety analyses, and output to the trip logic channels. This output to the trip logic channels in taken from a bistable device, which can be mechanical switches that are part of the process sensors, or from electronic comparators that receive input from the process transmitters or sensors. In either case, the bistable output contacts are considered to be part of the trip logic channel.

Trip Logic, Trip Setpoints, and ALLOWABLE VALUES

Trip setpoints are those predetermined values of output voltage or current against which the output voltage or current related to the present value of the process parameter is compared. If the present measured output value of the process parameter exceeds the setpoint, the associated bistable changes state. The trip setpoints are the nominal values at which the bistables are set. They are derived from the limiting values of the process parameters

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BASES (continued)

BACKGROUND (continued)

obtained from the accident analyses (analytical limits) through a process of correction for uncertainties and errors set forth in the plant-specific setpoint methodology (Ref. 3). The analytical limits, corrected for analytical and process uncertainties, become the ALLOWABLE VALUES, which when further corrected by the methodology of Reference 3 become the calculated trip setpoint values.

The setpoints derived in this manner provide adequate protection because sensor and processing time delays are accounted for as are calibration tolerances, instrumentation uncertainties, instrument drift, and severe environment errors for channels that must function in harsh environments as defined by 10 CFR 50.49 (Ref. 8). The actual nominal trip setpoint entered into the bistable is usually still more conservative than that calculated by the plant-specific setpoint methodology. If the setpoint measured for the bistable by the surveillance test does not exceed the documented surveillance test acceptance criteria, the bistable is considered OPERABLE.

Setpoints set in accordance with the ALLOWABLE VALUE will ensure that Safety Limits (SLs) are not violated during AOOs, and the consequences of DBAs will be acceptable, providing the plant is being operated within the LCOs at the onset of the AOO or DBA, and the equipment functions as designed, allowing for a single random active component failure.

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

The ALLOWABLE VALUES listed in Table 3.3.6.1-1 are based on the setpoint 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

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BASES (continued)

BACKGROUND (continued)

channels are assumed to operate within the allowances of these uncertainty magnitudes.

The PCI instrumentation automatically isolates the appropriate pipelines that penetrate the primary containment whenever monitored parameters exceed preselected setpoints. System level manual switches are provided in the control room to initiate isolation. A trip of a PCI instrument channel is annunciated in the control room. Motor- and air-operated isolation valves position indication is provided in the control room. PCI instrumentation interlocks isolate the reactor building ventilation system, and trip the drywell purge valves and purge fan units.

[For this facility, the PCI instrumentation provides isolated signals to valves grouped as follows:]

Group A isolation valves are in lines that communicate directly with the reactor vessel and penetrate the primary containment. These lines generally have two isolation valves in series; one inside the primary containment and the other outside the primary containment.

Group B isolation valves are in lines that do not communicate directly with the reactor vessel, but penetrate the primary containment and communicate with the primary containment free space. These lines have two isolation valves, both of which are outside the primary containment.

Group C isolation valves are in lines that penetrate the primary containment, but do not communicate directly with the reactor vessel, the primary containment free space, or the environs. These lines require one isolation valve outside the primary containment.

APPLICABLE SAFETY ANALYSES

The isolation signals generated by the PCI instrumentation initiate closure of valves to limit offsite doses. Each of the isolation instrumentation functions in Table 3.3.6.1-1 are implicitly assumed in the safety analyses of References 2 and 5.

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

The required channels of PCI instrumentation provide plant protection in the event of any of the analyzed accidents. PCI instrumentation protective functions are as follows:

1. Main Steam Line Isolation

Main steam line isolation is provided by the following functions:

Reactor Vessel Water Level--Low Low Low, Level 1

The Reactor Vessel Water Level--Low Low Low, Level 1 trip function provides isolation signals. The Reactor Vessel Water Level--Low Low Low, Level 1 function associated with isolation is assumed in the analysis of the recirculation line break (Ref. 5).

Main Steam Line Pressure--Low

The Main Steam Line Pressure--Low function is directly assumed in the analysis of the pressure regulator failure (Ref. 5). For this event, the closure of the main steam isolation valves (MSIVs) ensures that the temperature limit of the reactor vessel is not reached.

Main Steam Line Flow--High

The Main Steam Line Flow--High function is directly assumed in the analysis of the steam line break (SLB) (Ref. 5). The isolation action, along with the scram function of the RPS, assures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46 (Ref. 7).

Condenser Vacuum--Low

The Condenser Vacuum--Low function is provided to prevent overpressurization of the main condenser in the event of a loss of the main condenser vacuum. The integrity of the condenser is an assumption in offsite dose calculations. The closure of the MSIVs is initiated to prevent the addition of steam that would

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

lead to condenser over-pressurization and possible rupture of the turbine exhaust-hood rupture diaphragm, thereby limiting potential excessive offsite doses.

Area Temperature

The following discussion applies to the Main Steam Tunnel Temperature--High and to the Turbine Building Area Temperature--High functions. Area temperature is provided for early detection and isolation of a leak in the RCPB and for diversity to the main steam line flow detection instrumentation. The isolation occurs when a very small leak has occurred that, if allowed to continue, may result in offsite dose limits being exceeded.

Manual Initiation

[For this facility, the safety analyses basis for the Manual Initiation functions are as follows:]

2. Primary Containment Isolation

Primary containment will isolate on the following protection functions:

Reactor Vessel Water Level--Low, Level 3

The Reactor Vessel Water Level--Low, Level 3 function provides isolation signals to isolate systems that actually have a break. The isolation of some of the systems on Level 3 supports actions to ensure that SLs are not exceeded. [The systems that are isolated on reactor vessel water level 3 are bounded by breaks of the following larger systems:]

Drywell Pressure--High

High drywell pressure could indicate a break in the RCPB. An isolation of the primary containment is initiated in order to minimize offsite dose releases. Additionally, an isolation of systems which interface with the reactor vessel occurs to isolate potential sources of a break. The Drywell Pressure--High

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

function is one of the many functions capable of providing isolation signals. The isol= ion of some of the systems on high drywell pressure supports actions to ensure that SLs are not exceeded. The Drywell Pressure--High function associated with isolation is assumed in the Reference 5 analysis of the recirculation line break.

The HPCI and RCIC isolation of the turbine exhaust is provided to prevent communication with the drywell when high drywell pressure exists. A potential leak path exists from the primary containment to the secondary containment via each turbine's exhaust system.

Drywell Radiation--High

High Drywell radiation is an indication of possible gross failure of the fuel cladding. Therefore, when Drywell Radiation--High is detected, PCI isolation is initiated to limit the release of fission products.

Manual Initiation

Manual Initiation is required as a backup to automatic isolation and to allow operators to initiate containment isolation whenever any parameter is rapidly trending towards its trip setpoint. Manual Initiation was not modeled in the transient accident analysis but was qualitatively credited in the safety analysis and the NRC staff-approved licensing basis for the plant.

3, 4. High Pressure Coolant Injection System Isolation and Reactor Core Isolation Cooling System Isolation

HPCI and RCIC System isolations are provided by the following protection functions:

Drywell Pressure--High

Applicable safety analyses for this function have been previously discussed for PCI functions.

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

Steam Line Flow--High

The Steam Line Flow--High function is provided to detect a break of the RCIC or HPCI steam lines, and to initiate closure of the isolation valves of the appropriate system. If the steam were allowed to continue flowing out the break, the reactor would depressurize and the core could be uncovered. Therefore, the isolation is initiated on high steam flow to prevent or minimize core damage. The isolation action, along with the scram function of the RPS, assures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

Steam Line Pressure--Low

Low steam line pressure indicates that the pressure of the available steam may be too low to continue operation of the associated systems' turbine. [For this facility, the basis for the manual initiation of isolation to be delayed until the system becomes unavailable for injection (i.e., low steam line pressure) is as follows:]

Turbine Exhaust Diaphragm Pressure--High

Turbine Exhaust Diaphragm Pressure--High indicates that the pressure may be too high to continue operation of the appropriate system turbine; that is, one of two exhaust diaphragms has ruptured and pressure is reaching system pressure limitation. These isolations are for equipment overpressure protection.

Area and Differential Temperature

Area and differential temperatures are provided for each detection and isolation of a leak in the RCPB and for diversity to the steam flow detection instrumentation for the RCIC and HPCI Systems. The isolation occurs when a very small leak has occurred that, if allowed to continue, may result in offsite dose limits being exceeded.

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APPLICABLE SAFETY ANALYSES (continued)

Suppression Pool Area Temperature--Time Delay Relays

The Suppression Pool Area Temperature--Time Delay Relays are provided to allow all the other systems that may be leaking into the pool area (as indicated by the high temperature) to be isolated before HPCI or RCIC are isolated. This ensures maximum RCIC and HPCI System operating time by preventing their isolation due to leaks in other systems until the preset time delays have expired.

Emergency Area Cooler Temperature--High

[For this facility, the safety analyses basis for the Emergency Area Cooler Temperature--High function is as follows:]

Manual Initiation

Manual Initiation is required as a backup to automatic isolation and to allow operators to initiate containment isolation whenever any parameter is rapidly trending towards its trip setpoint. Manual Initiation was not modeled in the transient accident analysis but was qualitatively credited in the Safety Analysis and the NRC Staff-approved licensing basis for the plant.

5. Reactor Water Cleanup (RWCU) System Isolation

RWCU isolation is provided by the following functions:

Reactor Vessel Water Level--Low Low, Level 2

The Reactor Vessel Water Level--Low Low, Level 2 function provides isolation signals to isolate systems that actually have a break. The isolation of the RWCU System on Level 2 supports actions to ensure that SLs are not exceeded. [The systems that are being isolated are bounded by breaks of the following larger systems:]

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

RWCU System Differential Flow--High

The high differential flow signal is provided to detect a break in the RWCU System, which is part of the RCPB. This signal also provides a method of leak detection for the relatively cold portions of the RWCU System where area ambient temperature will not provide detection. Should the reactor coolant be allowed to continue to flow out of the break, reactor vessel water level would decrease and fuel damage could result. Therefore, isolation of the RWCU is initiated when high differential flow is sensed to prevent loss-of-coolant inventory and possible core damage. A time delay is provided to prevent spurious isolations during most RWCU operational transients.

Area and Differential Temperature--High

RWCU area and differential temperatures are provided for early detection and isolation of a leak in the RWCU System, which is part of the RCPB, and for diversity to the high differential flow detection instrumentation for the hot portions of the RWCU System. The isolation occurs when a very small leak has occurred that, if allowed to continue, might result in offsite dose limits being reached.

SLCS Initiation

The isolation of the RWCU System is required when the SLCS has been initiated to prevent dilution and removal of the boron solution by the RWCU System (Ref. 5).

RWCU Differential Pressure--High

[For this facility, the safety analyses basis for RWCU Differential Pressure--High is as follows:]

Manual Initiation

Manual Initiation is required as a backup to automatic isolation and to allow operators to initiate containment isolation whenever any parameter is rapidly trending towards its trip setpoint. Manual

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

Initiation was not modeled in the transient accident analysis but was qualitatively credited in the Safety Analysis and the NRC staff-approved licensing basis for the plant.

6. Shutdown Cooling System Isolation

Shutdown cooling isolation is provided by the following protection functions:

Reactor Vessel Water Level--Low, Level 3

Applicable safety analyses for this function have been previously discussed for primary containment isolation.

Reactor Steam Dome Pressure--High

The Reactor Steam Dome Pressure--High function is provided to isolate the low-pressure shutdown cooling portion of the Residual Heat Removal System.

The PCI instrumentation satisfies Criteria 1, 2, and 3 of the NRC Interim Policy Statement.

LCO

The LCO requires all instrumentation performing a PCI function to be OPERABLE. Failure of any instrument renders the affected channel(s) inoperable and reduces the reliability of the affected functions. Each function must have the required number of OPERABLE channels with their setpoints within the specified ALLOWABLE VALUES. Actuation setpoints are calibrated consistent with applicable setpoint methodology assumptions. Each channel must also respond within its assumed response time.

ALLOWABLE VALUES are specified for each function in the LCO. Nominal trip setpoints are specified in the plant- specific setpoint calculations (Ref. 3). The nominal setpoints are selected to ensure that the setpoint measured by CHANNEL FUNCTIONAL TESTS does not exceed the ALLOWABLE surveillance test's acceptance criteria.

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BASES (continued)

LCO (continued)

PCI instrumentation has inputs to the trip logic of the functions listed below.

1. Main Steam Line Isolation

Most PCI main steam line isolation functions receive inputs from four process (sensor and bistable) channels. The bistable outputs from these channels are combined in one-out-of-two taken-twice logic to initiate isolation of all MSIVs. The bistable outputs from the same channels are arranged into two two-out-of-two logic trip systems to isolate all main steam line drains. Each main steam line drain line has two isolation valves with one two-out-of-two logic system associated with each valve.

The exceptions to this arrangement are the Main Steam Line Flow--High function and area temperature functions. The Main Steam Line Flow--High function uses 16 flow channels, 4 for each steam line. The four channels on each steam line are connected in one-out-of-two taken twice logic to initiate isolation of the associated MSIV. Similarly, the four flow channels are connected in two two-out-of-two logic trip systems, each trip system isolating one of the two main steam line drain valves on the associated steam line.

The Main Steam Tunnel Temperature--High function receives input from 16 channels. The Turbine Building Area Temperature--High function receives input from 64 channels. [For this facility, these area temperature channels are arranged into one-out-of-two taken-twice logic for MSIV isolation, and two-out-of-two logic for main steam line drain isolation as follows:]

Main steam line isolation functions actuate the Group 1 valves. [For this facility, the constituents of this group are found in:]

1.a. Reactor Vessel Water Level--Low Low Low, Level 1

Low reactor vessel water level indicates that the capability of cooling the fuel may be threatened. Should reactor vessel water level decrease too

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BASES (continued)

LCO (continued)

far, fuel damage could result. Therefore, isolation of the MSIVs and other interfaces with the reactor vessel and primary containment occurs to isolate potential sources of a break.

Reactor vessel water level signals are initiated from level transmitters. Four channels of Reactor Vessel Water Level--Low Low Low, Level 1 function are available and are required to be OPERABLE to ensure that no single failure can preclude the isolation function. The Reactor Vessel Water Level--Low Low, Level 1 ALLOWABLE VALUE is specified to be the same as the ECCS ALLOWABLE VALUE in LCO 3.3.5.1.

1.b. <u>Main Steam Line Pressure--Low</u>

Low main steam line pressure indicates that there may be a problem with the turbine pressure regulation. This could result in a low reactor vessel water level condition and the reactor vessel cooling down more than 100°F/hour if the pressure loss is allowed to continue.

The main steam line low-pressure signals are initiated from four transmitters that are connected to the main steam line.

Four channels of Main Steam Line Pressure--Low function are available and are required to be OPERABLE to ensure that no single failure can preclude the isolation function. The ALLOWABLE VALUE was selected to be low enough not to interfere with normal plant operation, but high enough to prevent excessive reactor vessel depressurization.

1.c. <u>Main Steam Line Flow--High</u>

Main Steam Line Flow--High is provided to detect a break of the main steam line and to initiate closure of MSIVs. If the steam was allowed to continue flowing out of the break, the reactor would depressurize and the core could be uncovered. If the reactor vessel water level

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BASES (continued)

LCO (continued)

were to decrease too far, fuel damage could occur. Therefore, the isolation is initiated on high flow to prevent or minimize core damage.

The Main Steam Line Flow--High signals are initiated from 16 transmitters that are connected to the four main steam lines. The transmitters are arranged so that, even though physically separated, all four of those connected to one steam line would be able to detect the high flow. Sixteen channels of Main Steam Line Flow--High function are available and required to be OPERABLE so that no single failure will preclude the detection of a break in any individual main steam line of the isolation function. The ALLOWABLE VALUE is set high enough to prevent spurious trips during closure testing of the MSIVs, turbine stop valves (TSVs), and turbine control valves (TCVs) but low enough to ensure that the trip occurs to prevent fuel damage.

1.d. Condenser Vacuum--Low

Condenser vacuum pressure signals are derived from four pressure transmitters that sense the pressure in the condenser. Four channels of Condenser Vacuum--Low function are available and are required to be OPERABLE to ensure that no single failure can preclude the isolation function. The ALLOWABLE VALUE is set low enough to allow the Condenser Vacuum--Low turbine trip, which occurs prior to MSIV closure, to stabilize the vacuum, but high enough to prevent damage to the condenser. This function may be bypassed in MODES 2 and 3 when all TSVs are closed, because the condenser is not receiving steam from the main steam lines.

1.e, 1.f, 1.g. Area Temperature

Area temperature signals are initiated from thermocouples located in the area being monitored. Each thermocouple is considered a channel. Sixteen channels of Main Steam Tunnel Temperature--High function, [64] channels of

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BASES (continued)

LCO (continued)

Turbine Building Area Temperature--High function, and Main Steam Tunnel Differential Temperature--High function are available and required to be OPERABLE to ensure that no single failure can preclude the isolation function. [For this facility, the logic arrangement and basis for the logic arrangement is as follows:] The temperature monitoring ALLOWABLE VALUE is set high enough to allow for changes in the ambient conditions, but low enough to detect a leak equivalent to 25 gpm.

1.h. Manual Initiation

The Manual Initiation push button switches introduce signals into the appropriate system's isolation logic that are redundant to the automatic instrumentation channels and provide manual isolation capability. [For this facility, the push button arrangement for this isolation system is as follows:]

2. Primary Containment Isolation

Each PCI function receives inputs from four process sensor and bistable channels. The bistable outputs from these channels are arranged into two two-out-of-two logic trip systems. One trip system initiates isolation of all inner PCI valves, and one trip system initiates isolation of all outer PCI valves. Each logic closes one of the two valves on each penetration, so that operation of either logic isolates the penetration.

PCI functions actuate the Group 2, 6, 7, 10, and 12 valves and containment purge and vent valves. [For this facility the constituents of these valve groups are found in:]

2.a. Reactor Vessel Water Level--Low, Level 3

Low reactor vessel water level indicates that the capability of cooling the fuel may be threatened. Should reactor vessel water level decrease too

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BASES (continued)

LCO (continued)

far, fuel damage could result. Therefore, isolation of the primary containment, shutdown cooling and other interfaces with the reactor vessel occurs to begin isolating the potential sources of a break.

Reactor vessel water level signals are initiated from level transmitters. The transmitters are arranged on four sets of separated taps. Four channels of Reactor Vessel Water Level--Low, Level 3 function are available and are required to be OPERABLE to ensure that no single failure can preclude the isolation function. The Reactor Vessel Water Level--Low, Level 3 ALLOWABLE VALUE is specified to be the same as the RPS level 3 scram ALLOWABLE VALUE.

2.b. Drywell Pressure--High

High Drywell pressure signals are initiated from pressure transmitters that sense the pressure at four different locations in the Drywell. The transmitters are located outside the Drywell. Four channels of Drywell Pressure--High function are available to provide inputs to the two trip systems' logic and are required to be OPERABLE to ensure that no single failure can preclude the isolation function.

2.c. Drywell Radiation--High

The Drywell radiation signals are initiated from radiation detectors located in the drywell. The detectors are arranged so that, even though physically separated, each detector detects some radiation in a postulated accident. The signal from each detector is input to an individual monitor whose trip outputs are assigned to one of the two isolation channels in the trip system. Two channels of Drywell Radiation--High function are available and are required to be OPERABLE to ensure that no single failure can preclude the isolation function. The ALLOWABLE VALUE is high enough above background radiation levels to

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BASES (continued)

LCO (continued)

minimize spurious trips, yet low enough to promptly detect gross failures in the fuel cladding.

The ALLOWABLE VALUE was selected to be the same as the ECCS high drywell pressure ALLOWABLE VALUE (LCO 3.3.5.1).

2.d. Manual Initiation

The Manual Initiation push button switches introduce signals into the appropriate system's isolation logic that are redundant to the automatic instrumentation channels and provide manual isolation capability. [For this facility, the push button arrangement for this isolation system is as follows:]

3, 4. High Pressure Coolant Injection System Isolation and Reactor Core Isolation Cooling System Isolation

The HPCI and RCIC isolation function receives input from four turbine exhaust diaphram pressure and four turbine steam pressure channels. The bistable outputs from the turbine exhaust diaphram pressure and steam supply pressure channels are connected into two two-out-of-two trip systems. All of the other functions receive input from two channels with each channel in one trip system using one-out-of-one logic. Each of the two trip systems in each isolation group is connected to one of the two valves on each associated penetration.

HPCI and RCIC functions actuate the Group 3, 4, 8 and 9 valves. [For this facility, the constituents of these valve groups is as follows:]

3.a, 4.a. Steam Line Flow--High

The RCIC and HPCI steam high-flow signals are initiated from transmitters (two for RCIC and two for HPCI) that are connected to the system steam lines. The transmitters are arranged so that,

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(continued)

BASES (continued)

LCO (continued)

even though physically separated, both will be able to detect the high flow. Two channels of both HPCI and RCIC Steam Line Flow--High functions are available and are required to be OPERABLE to ensure that no single failure can preclude the isolation function. The ALLOWABLE VALUE is set high enough to prevent spurious isolations during system startup transients, but low enough to ensure that the isolation occurs to prevent fuel damage.

3.b, 4.b. <u>Steam Line Pressure--Low</u>

The steam line pressure signals are initiated from transmitters (four for RCIC and four for HPCI) that are connected to each system's steam line. The transmitters are arranged so that, even though physically separated, each transmitter is able to detect low steam line pressure. Four channels of both HPCI and RCIC Steam Line Pressure--Low functions are available and are required to be OPERABLE to ensure that no single failure can preclude the isolation function. The ALLOWABLE VALUE is selected to be low enough to allow system operation, but high enough to prevent damage to system turbines.

3.c, 4.c. <u>Turbine Exhaust Diaphragm Pressure--High</u>

The turbine exhaust diaphragm pressure signals are initiated from transmitters (four for RCIC and four for HPCI) that are connected to the area between the rupture diaphragms on each system's turbine exhaust line. The transmitters are arranged so that, even though physically separated, each transmitter is able to detect high turbine exhaust line pressure. Four channels of both HPCI and RCIC Turbine Exhaust Diaphragm Pressure--High functions are available and are required to be OPERABLE to ensure that no single failure can preclude the isolation function. The ALLOWABLE VALUE is low enough to allow system operation, but high enough to prevent damage to the low-pressure sections of each system.

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BASES (continued)

LCO (continued)

3.d, 4.d. Drywell Pressure--High

High drywell pressure signals are initiated from pressure transmitters that sense the pressure at four different locations in the drywell. The transmitters are located outside the drywell. Four channels of Drywell Pressure--High function are available to provide inputs to the two trip systems' logic and are required to be OPERABLE to ensure that no single failure can preclude the isolation function.

The ALLOWABLE VALUE was selected to be the same as the ECCS high drywell pressure ALLOWABLE VALUE LCO 3.3.5.1 of "ECCS Instrumentation."

Two channels are available and required to be OPERABLE for both HPCI and RCIC. [For this facility, the basis for the two-channel logic configuration is as follows:]

3.e, 3.f, 3.h, 4.e, 4.g, 4.i, 4.j. Area and Differential Temperature

Area and differential temperature signals are initiated from temperature elements that are appropriately located to protect the room that is being monitored. There are two instruments monitoring each area and the two channels of instruments are physically separated. Two channels for each HPCI and RCIC Area Temperature-High function are available and are required to be OPERABLE. Each channel of the Suppression Pool Area Differential Temperature--High function is initiated from thermocouples that are located in the inlet and outlet of the area cooling systems. Two channels for both HPCI and RCIC are available and are required to be OPERABLE to ensure that no single failure can prevent an isolation. The ambient and differential temperature ALLOWABLE VALUES are set high enough to allow changes in ambient conditions, but low enough to detect a leak equivalent to 25 gpm.

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BASES (continued)

LCO (continued)

3.g, 4.f. Suppression Pool Area Temperature--Time Delay Relays

There are two time delay relays and the two channels of instruments are physically separated. Two channels for both the HPCI and RCIC Suppression Pool Area Temperature--Time Delay Relay function are available and are required to be OPERABLE.

3.i, 4.h. Emergency Area Cooler Temperature--High

[For this facility, the bases for the logic configuration and ALLOWABLE VALUE are as follows:]

3.j, 4.k. Manual Initiation

The Manual Initiation push button switches introduce signals into the appropriate system's isolation logic that are redundant to the automatic instrumentation channels and provide manual isolation capability. [For this facility, the push button arrangement for this isolation system is as follows:]

5. Reactor Water Cleanup System Isolation

The RWCU isolation function receives input from four reactor vessel water level channels. The bistable outputs from the reactor vessel water level channels are connected into two two-out-of-two trip systems. Most of the other functions receive input from two channels, with each channel in one trip system using one-out-of-one logic. The Area Temperature--High function receives input from six temperature monitors, three to each trip system. The Area Ventilation Differential Temperature--High function receives input from 12 temperature monitors, 6 in each trip system. These are configured so that any one input high trips the associated trip system. Each of the two trip systems is connected to one of the two valves on each RWCU penetration.

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BASES (continued)

LCO (continued)

RWCU functions actuate the Group 5 outboard isolation valve only. [For this facility, the constituents of the valve group are found in:]

5.a. Differential Flow--High

The high differential flow signals are initiated from transmitters that are connected to the inlet and outlet of the RWCU System. The outputs of the transmitters are compared and, if the difference between the inlet and outlet flows is too large, an isolation signal is initiated. Two channels of RWCU System Differential Flow--High functions are available and are required to be OPERABLE to ensure that no single failure can prevent an isolation. The RWCU System Differential Flow--High ALLOWABLE VALUE is high enough to prevent spurious isolations during RWCU System transients, but low enough to ensure that the isolation occurs as soon as a break in the RWCU piping has been detected.

5.b, 5.c. Area and Differential Temperature--High

Area and differential temperature signals are initiated from temperature elements that are located in the room that is being monitored. Six temperature monitors provide input to the Area Temperature--High function's two trip systems in one-out-of-one logic, but, because of the redundancy of the initiating signals, only two are required to be OPERABLE to ensure that no single failure can prevent an isolation. [For this facility, the isolation logic is applied as follows:]

There are 12 temperature monitors that provide input to the Area Ventilation Differential Temperature--High function. The output of these monitors is used to determine the differential temperature. One channel consists of thermocouples that are located in the inlet and outlet of the area cooling system, for a total of six available channels. Because of the redundancy of the system, however, only two are

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BASES (continued)

LCO (continued)

required to be OPERABLE to ensure that no single failure can prevent an isolation. [For this facility, the sensor's physical layout and the bases for considering the sensors redundant are as follows:]

The ambient and differential temperature monitoring ALLOWABLE VALUES are set high enough to allow changes in ambient conditions, but low enough to detect a leak equivalent to 25 gpm.

5.d. SLCS Initiation

The one channel feeds one of the two SLCS subsystems and isolates on a one-out-of-one logic. There is no ALLOWABLE VALUE associated with this function.

5.e. Reactor Vessel Water Level--Low Low, Level 2

Low reactor vessel water level indicates that the capability of cooling the fuel may be threatened. Should reactor vessel water level decrease too far, fuel damage could result. Therefore, isolation of the RWCU occurs to isolate the potential sources of a break.

Reactor vessel water level signals are initiated from level transmitters. The transmitters are arranged on four sets of separated taps. Four channels of Reactor Vessel Water Level--Low Low, Level 2 function are available and are required to be OPERABLE to ensure that no single failure can preclude the isolation function. The Reactor Vessel Water Level--Low Low, Level 2 ALLOWABLE VALUE is specified to be the same as the ECCS instrumentation Level 2 ALLOWABLE VALUE, LCO 3.3.5.1, "ECCS Instrumentation."

5.f. RWCU Differential Pressure--High

[For this facility, the basis for the function logic configuration and its ALLOWABLE VALUE are as follows:]

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BASES (continued)

LCO (continued)

5.g. Manual Initiation

The manual initiation push button switches introduce signals into the appropriate system's isolation logic that are redundant to the automatic instrumentation channels and provide manual isolation capability. [For this facility, the push button arrangement for this isolation system is as follows:]

6. Shutdown Cooling System Isolation

The shutdown cooling isolation function receives input from four reactor vessel water level channels, and from two channels for all other functions. The bistable outputs from the reactor vessel water level channels are connected into two two-out-of-two trip systems. The Reactor Vessel Pressure--High function receives input from two channels, with each channel in one trip system using one-out-of-one logic. Each of the two trip systems is connected to one of the two valves on each shutdown cooling penetration.

Shutdown cooling functions actuate Group 11 valves. [For this facility, constituents of this group are found in:]

6.a. Reactor Steam Dome Pressure--High

The reactor steam dome pressure signals are initiated from two transmitters that are connected to different taps on the reactor vessel. The transmitters are arranged so that, even though physically separated, both will sense the high dome pressure. Two channels of Reactor Steam Dome Pressure--High function are available and are required to be OPERABLE. The ALLOWABLE VALUE is high enough to allow the Shutdown Cooling System to begin operation before HPCI and RCIC are isolated, but low enough to protect the system equipment from overpressure.

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BASES (continued)

LCO (continued)

6.b. Reactor Vessel Water Level--Low, Level 3

LCOs for this protection function have been previously discussed for the primary containment isolation function.

Certain ECCS and RCIC valves (e.g., minimum flow) also serve the dual function of automatic containment isolation valves. The signals that isolate these valves are also associated with the automatic initiation of the ECCS and RCIC. The instrumentation and ACTIONS associated with these signals are addressed in LCO 3.3.5.1, "ECCS Instrumentation," and LCO 3.3.5.2, "RCIC Instrumentation," and are not included in this LCO.

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

[For this facility, those required support systems which, upon their failure do not result in the PCI instrumentation being declared inoperable, and their justification are as follows:]

[For this facility, the supported systems impacted by the inoperability of the PCI instrumentation and the justification of whether or not each supported system is declared inoperable are as follows:]

APPLICABILITY

Each of the functions in Table 3.3.6.1-1 are required to be OPERABLE in MODES 1, 2, and 3 consistent with the Applicability for LCO 3.6.1.1, "Containment" OPERABILITY, except that:

- Main Steam Line Pressure--Low is required to be OPERABLE in MODE 1 when the reactor pressure is high enough that the low pressure function provides an indication of a steam line break.
- RWCU isolation on SLCS initiation is required to be OPERABLE in MODES 1 and 2 when the reactor can be critical per LCO 3.1.7; and

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BASES (continued)

APPLICABILITY (continued)

- Shutdown Cooling System isolation on Reactor Vessel Water Level--Low, Level 3 is required to be OPERABLE in MODES 3, 4, and 5. [For this facility, the bases for the Shutdown Cooling System applicability requirements are as follows:]

A Note has been added to provide clarification that each function specified in Table 3.3.6.1-1 shall be treated as an independent entity for this LCO with an independent Completion Time.

ACTIONS

A protection function 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 to within specification. If the trip setpoint is less conservative than the ALLOWABLE VALUE in Table 3.3.6.1-1, the channel must be declared inoperable immediately, and the appropriate Conditions from Table 3.3.6.1-1 must 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 bistable 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.

In order for a facility to take credit for topical reports as the basis for justifying Completion Times, such topical reports should be supported by an NRC staff Safety Evaluation Report that establishes and acceptability of each such topical report for that facility.

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BASES (continued)

ACTIONS (continued)

Condition A

Condition A is applicable to each of the PCI instrumentation isolation functions in Table 3.3.6.1-1. This includes all automatic valves and valve groups designed to receive an isolation signal assumed by the safety analyses to initiate closure of valves to preserve the integrity of the primary containment and limit offsite doses. It provides Required Actions that ensure the integrity of the penetration isolation capability and for evaluation of PCI instrumentation inoperability, including time for minor repairs subsequent to placing the channel in trip.

If more than one channel is inoperable for any primary containment isolation instrumentation function, either by itself or together with single channel failures in other PCI instrumentation functions, then the PCI instrumentation may not be capable of performing its intended function.

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

If one or more channels are inoperable for one or more functions, operation may continue provided each function is still capable of isolating the associated lines. Required Action A.1 requires an evaluation to ensure that the isolation capability exists for each function. Maintaining isolation capability refers to the ability to provide automatic and, if required, manual isolation of all affected lines, with OPERABLE and/or tripped channels. For a four-channel function, one-out-of-two logic, this would required only one OPERABLE channel in each function (i.e., only one valve per line required to isolate). [For this facility, the basis for one-out-of-two taken twice logic, one-out-of-one taken twice logic, and one-out-of-one logic are as follows:]

The Completion Time of 1 hour is sufficient for plant operations personnel to complete Required Action A.1.

Required Action A.2.1 is the preferred action because it restores full functional capability to PCI instrumentation. As an alternative to restoring a channel(s) to OPERABLE status (A.2.1), operation is allowed to continue provided the inoperable channel is placed in trip (A.2.2). Required Action A.2.2 is modified by a Note that the Required Action

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BASES (continued)

ACTIONS (continued)

does not apply if it results in an isolation. If a channel in one trip system becomes inoperable when one or more channels in the opposite system are already in trip, placing the inoperable channel in trip will cause an isolation. Required Action A.2.2 is not intended to force an unnecessary isolation. In this event, Required Action A.2.1 would have to be met. If the inoperable channel(s) are not restored to OPERABLE status or placed in trip within the time allowed, Condition B should be entered. The bases for the 12-hour and 24-hour Completion Times provided to allow continued operation with inoperable channels while taking actions to restore or trip the channels are in Reference 6.

<u>Condition B</u>

Condition B applies to each of the PCI instrumentation functions listed in Table 3.3.6.1-1.

<u>B.1</u>

This Required Action directs entry into all remaining Conditions (except A or B). When any Required Action of Condition A is not met, and the associated Completion Time has expired, the applicable Condition from Table 3.3.6.1-1 must be entered for function(s) whose inoperable channel(s) have not been placed in trip. The applicable Condition specified in Table 3.3.6.1-1 is function- and mode-dependent and may change as the Required Action of a previous Condition is completed.

<u>Condition C</u>

Condition C is applicable to the main steam line isolation functions except low pressure.

<u>C.1, C.2.1, and C.2.2</u>

If the required number of channels are not restored to OPERABLE status or placed in trip within the allowed Completion Time, the plant must be placed in a mode or other specified condition in which the LCO does not apply. The Completion Time of 6 hours to isolate all MSIVs is reasonable, based on operating experience, to initiate actions for a controlled reduction in power to establish

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BASES (continued)

ACTIONS (continued)

required plant conditions and isolate the main steam lines (equivalent to MODE 2) in an orderly manner and without challenging plant systems.

Twelve hours and 36 hours are reasonable, based on operating experience, to reach MODE 3 and MODE 4, respectively, in an orderly manner and without challenging plant systems.

Condition D

Condition D is applicable to the main steam line low-pressure isolation function.

D.1

If the required number of channels are not restored to OPERABLE status or placed in trip within the allowed Completion Time, the plant must be placed in a mode or other specified condition in which the LCO does not apply.

Six hours is reasonable, based on operating experience, to initiate actions for a controlled reduction in power and to reach MODE 2 in an orderly manner and without challenging plant systems.

Condition E

Condition E is applicable to PCI drywell radiation; HPCI and RCIC isolation functions; RWCU isolation functions except SLCS initiation and shutdown cooling isolation on high reactor vessel pressure.

E.1

If the required number of channels are not restored to OPERABLE status or placed in trip within the allowed Completion Time, plant operations may continue if the affected lines are isolated. Isolating the affected lines accomplishes the safety function of the inoperable channels. The 1-hour Completion Time is sufficient for plant operations personnel to take the appropriate action.

If, for some reason, these Required Actions cannot be completed within the time allowed, action must be taken in

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BASES (continued)

ACTIONS (continued)

accordance with Condition F to place the plant in a MODE or Condition in which the inoperable functions are not required.

Condition F

Condition F is applicable to the Required Actions of Condition E if they are not met, PCI Level 3, drywell pressure and each manual initiation function in Table 3.3.6.1-1.

F.1 and F.2

If the required number of channels are not restored to OPERABLE status or placed in trip within the allowed Completion Time, the plant must be placed in a mode or other specified condition in which the LCO does not apply. Twelve hours and 36 hours are reasonable, based on operating experience, to initiate actions for a controlled reduction in power and to reach MODE 3 and MODE 4, respectively, in an orderly manner and without challenging plant systems.

Condition G

Condition G is applicable to RWCU SLCS initiation.

G.1 and G.2

If the required number of channels are not restored to OPERABLE status or placed in trip within the allowed Completion Time, the associated SLCS subsystems are declared inoperable or the RWCU System is isolated. Since this function is required to ensure that the SLCS performs its intended function, sufficient remedial measures are provided by declaring the associated SLCS inoperable or isolating the RWCU System. The 1-hour Completion Time is sufficient for operations personnel to perform the corrective actions.

Condition H

Condition H is applicable to the shutdown cooling level 3 function.

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BASES (continued)

ACTIONS (continued)

H.1

[For this facility, the basis for the Required Action and the 1-hour Completion Time is as follows:]

Condition I

Condition I is applicable to each one of the primary containment isolation instrumentation functions presented in Table 3.3.6.1-1.

I.1 and I.2

Required Action I.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 verification.

Required Action I.1 ensures that those identified Required Actions associated with supported systems impacted by the inoperability of channel(s) associated with each primary containment isolation instrumentation 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 I is entered. [For this facility, the identified supported systems Required Actions associated with each PCI instrumentation function are as follows:]

Required Action I.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.

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BASES (continued)

SURVEILLANCE REQUIREMENTS

The SRs for any particular PCI instrumentation function are found in the SR column of Table 3.3.6.1-1 for that function. Most functions are subject to CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION.

In order for a facility to take credit for topical reports as the basis for justifying Surveillance Frequencies, the topical reports should be supported by an NRC staff Safety Evaluation Report that establishes the acceptability of each topical report for that facility.

The SRs are modified by Note 2 to indicate that a channel may be placed in an inoperable status for up to 6 hours for required surveillances provided at least one OPERABLE channel in the same trip system is monitoring that parameter. Upon completion of the surveillance, or expiration of the 6-hour allowance, the channel must be returned to OPERABLE status or the Required Actions taken. It is not acceptable to routinely remove channels from service for more than 6 hours to perform required surveillance testing. Such a practice would be contrary to the assumptions of the reliability analysis that justified LCO Completion Times.

SR 3.3.6.1.1

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 against the indication of the same parameter on other instrument channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure, thus it is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION. 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 operated from the control room, the CHANNEL CHECK should also note the detector's response to these sources.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

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 surveillances are 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 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 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 channels required by the LCO.

<u>SR 3.3.6.1.2</u>

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 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 must be found within the ALLOWABLE VALUES 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

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

over a small number of test intervals should be evaluated as potentially indicating a deterministic failure which cannot be corrected by recalibration. If, during a CHANNEL FUNCTIONAL TEST, the associated trip setting is discovered to be less conservative than the ALLOWABLE VALUE specified in Table 3.3.6.1-1, the channel must be declared inoperable.

The 92-day Surveillance Frequency is based on the reliability analysis described in Reference 6. A Note modifies this SR, which permits radiation detectors to be exempt from the CHANNEL FUNCTIONAL TEST requirements. [For this facility, the basis for excluding radiation detectors is as follows:]

SR 3.3.6.1.3

The calibration of bistable trip units consists of a test to determine the actual trip setpoints, and recalibration of the setpoint is necessary to ensure that it remains more conservative than the setpoint and the ALLOWABLE VALUE specified in Table 3.3.6.1-1. The channel must be declared inoperable if the setting is discovered to be less conservative than the ALLOWABLE VALUE. If the trip setting is discovered to be less conservative than accounted for in the appropriate setpoint methodology but is not beyond the ALLOWABLE VALUE, the channel is still considered OPERABLE. Under these conditions, the setpoint must be readjusted to be more conservative than that accounted for in the appropriate setpoint methodology.

The Surveillance Frequency is based on the assumption of a 92-day calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis of Reference 3.

SR 3.3.6.1.4

CHANNEL CALIBRATION is a complete check of the instrument channel including the detector. 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 with the setpoint within the assumptions of the plant-specific setpoint analysis. Transmitter

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

"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 setpoints 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 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 the assumption of a 92-day calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis (Ref. 6).

SR 3.3.6.1.5

The basis for performance of a CHANNEL CALIBRATION was previously discussed in SR 3.3.6.1.4.

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) or thermocouple (T/C) 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 or T/C should be replaced with a newly calibrated sensor during each refueling cycle to ensure accurate sensor cross-calibration. This replacement sensor must be the same model as the remaining RTDs or T/Cs. Using a newly calibrated sensor as a reference ensures that 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 sensor drift is random rather than

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

systematic, and is bounded by the plant-specific setpoint analysis assumptions. This determination may use results of statistical analyses of operating data and calibration data from similar plants using the same model of RTD or T/C 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 in the setpoint analysis (Ref. 3).

SR 3.3.6.1.6

Performance of a LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required actuation logic for a specific channel. The LOGIC SYSTEM FUNCTIONAL TEST tests all logic components (i.e., all relays and contacts, trip units, solid-state logic elements, etc.) of a logic circuit, from sensor up to the actuated device. [For this facility, the system functional testing performed on isolation valves overlaps this test to provide complete testing of the safety function as follows:] The Surveillance Frequency is based on experience that it is prudent that these surveillances be performed only during a plant outage. This is due to the plant conditions needed to perform the surveillance and the potential for system isolation during the LOGIC SYSTEM FUNCTIONAL TESTS. Operating experience has shown these components usually pass the surveillance when performed at the 18-month Frequency.

SR 3.3.6.1.7

SR 3.3.6.1.7 is a CHANNEL FUNCTIONAL TEST performed in the PCI manual isolation functions. This test verifies each initiation switch isolates the associated groups of valves as designed.

The 18-month surveillance interval is based upon experience that it is prudent that these surveillances only be performed during a plant outage. This is due to the plant conditions needed to perform the surveillance and the potential for system isolation 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.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.6.1.8

The basis for performance of a CHANNEL FUNCTIONAL TEST was previously discussed in SR 3.3.6.1.2.

[For this facility, the basis for the 184-day Surveillance Frequency is as follows:]

SR 3.3.6.1.9

This SR ensures that the ISOLATION SYSTEM RESPONSE TIMES for each channel 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 system isolation. [For this facility, the basis for the acceptable response times of the relevant trip channels are as follows: The response times include contributions from the following equipment:] 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 radiation detectors may be excluded from ISOLATION SYSTEM RESPONSE TIME testing. This note is necessary because of the difficulty of generating an appropriate detector input signal. [For this facility, the basis for excluding the detectors is acceptable as follows:]

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 x 18 months, where n is the number of channels per trip system 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 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. Response times cannot be determined at power since equipment operation is required.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, "Domestic Licensing of Production and Utilization Facilities."

2. [Unit Name] FSAR, Section [9], "[Title]."

3. [Unit Name] FSAR, "Plant-Specific Setpoint Methodology."

4. [Unit Name] FSAR, Section [7], "Instrumentation and Control]."

5. [Unit Name] FSAR, Section [15] "Title]."

6. NEDC-31677-P-A, "Technical Specification Improvement Analysis for BWR Isolation Actuation Instrumentation," June 1989.

7. Title 10, Code of Federal Regulations, Part 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for light water nuclear power reactors."

8. Title 10, Code of Federal Regulations, Part 50.49, "Environment Qualification of Electrical Equipment Important to Safety for Nuclear Power Plants."

B 3.3 INSTRUMENTATION

B 3.3.6.2 Secondary Containment Isolation (SCI) Instrumentation

BASES

BACKGROUND

The SCI instrumentation automatically initiates closure of appropriate unit SCI valves and starting of the unit's Standby Gas Treatment System (SGTS). This function is necessary to prevent or limit the release of fission products from the secondary containment in the event of a loss-of-coolant accident (LOCA) or a reactor coolant pressure boundary (RCPB) leak.

The SCI instrumentation includes the sensors, relays, and switches that are necessary to cause initiation of SCI. Functional diversity is provided by monitoring a wide range of dependent and independent parameters. The input parameters to the isolation logics are electrical signals that indicate limits on reactor vessel water level and drywell pressure. Other inputs into the isolation logic are reactor building (RB) exhaust and refueling floor exhaust high radiation. Redundant sensor input signals are provided from each of the SCI initiation parameters.

The SCI instrumentation is designed to include the subsystems identified below:

- Field transmitters or process sensors;
- Signal processing and bistable modules; and
- Trip logic, trip setpoints, and ALLOWABLE VALUES.

Field Transmitters or Process Sensors

Field transmitters or process sensors provide a measurable electronic output signal based on the physical characteristics of the parameter being measured.

Typically, four measurement channels with physical separation are provided for each parameter. These are typically organized into two trip systems which are physically and electrically separated. Four measurement channels are necessary to meet the redundancy and testability criteria of 10 CFR 50, Appendix A, GDC 21

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BASES (continued)

BACKGROUND (continued)

(Ref. 1), and to implement the one-out-of-two taken twice logic arrangement discussed below for the SCI instrumentation.

[For this facility, a discussion of those SCI parameters which do not have four measurement channels and their conformance to redundancy and testability requirements of GDC 21 in 10 CFR 50, Appendix A, is as follows:]

For most anticipated occupational occurrences (AOOs) and Design Basis Accidents (DBAs), a wide range of dependent and independent parameters are monitored.

Signal Processing and Bistable Modules

Each process parameter measurement channel includes electronic equipment that provides signal conditioning, comparable output signals for main control board instruments, comparison of measured input signals with setpoints established by safety analyses, and output to the trip logic channels. This output to the trip logic channels is taken from a bistable device which can be mechanical switches that are part of the process sensors or electronic comparators that receive input from the process transmitters or sensors. In either case, the bistable output contacts are considered to be part of the trip logic channel.

Trip Logic, Trip Setpoints, and ALLOWABLE VALUES

Trip setpoints are those predetermined values of output voltage or current against which the output voltage or current related to the present value of the process parameter is compared. If the present measured output value of the process parameter exceeds the setpoint, the associated bistable changes state. The trip setpoints are the nominal values at which the bistables are set. They are derived from the limiting values of the process parameters obtained from the accident analyses (analytical limits) through a process of correction for uncertainties and errors set forth in the plant-specific setpoint methodology (Ref. 3). The analytical limits, corrected for analytical and process uncertainties, become the ALLOWABLE VALUES, which when further corrected by the methodology of Reference 3 become the calculated trip setpoint values.

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BASES (continued)

BACKGROUND (continued)

The setpoints derived in this manner provide adequate protection because sensor and processing time delays are accounted for as well as calibration tolerances, instrumentation uncertainties, instrument drift, and severe environment errors for channels that must function in harsh environments as defined by 10 CFR 50.49 (Ref. 7). The actual nominal trip setpoint entered into the bistable is usually still more conservative than that calculated by the plant-specific setpoint methodology. If the setpoint measured for the bistable by the surveillance test does not exceed the documented surveillance test acceptance criteria, the bistable is considered OPERABLE.

The outputs of the logic channels in a trip system are combined in a logic so that both channels are required to trip the associated trip system. The logic is one-out-of-two for each trip system. Typically, automatically isolated secondary containment penetrations are isolated by two isolation values, so that operation of either trip system isolates the penetration.

The trip setpoints are the nominal 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 and comparator setting accuracy).

The trip setpoints used in the bistables are based on the analytical limits stated in Reference 3. 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 SCI channels that must function in harsh environments, as defined by 10 CFR 50.49 (Ref. 7), ALLOWABLE VALUES, specified in Table 3.3.6.1-1, 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 the plant-specific setpoint methodology. The actual nominal trip setpoint entered into the bistable is normally more conservative than that required by the plant-specific setpoint calculations. If

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BASES (continued)

BACKGROUND (continued)

the measured setpoint does not exceed the documented surveillance test acceptance criteria, the bistable is considered OPERABLE.

Setpoints set in accordance with the ALLOWABLE VALUE will ensure that the consequences of DBAs will be acceptable, providing the plant is being operated within the LCOs at the onset of the AOO or DBA, and the equipment functions as designed, allowing for a single random active component failure.

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

The ALLOWABLE VALUES listed in Table 3.3.6.2-1 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.

APPLICABLE SAFETY ANALYSES

The actions of the instrumentation are implicitly assumed in the safety analyses of References 2, 4, and 5. The isolation initiates closure of valves to ensure secondary containment OPERABILITY and limit offsite doses.

The SCI instrumentation automatically isolates the appropriate pipelines that penetrate the secondary containment whenever the monitored parameters exceed preselected setpoints. System level manual switches are also provided in the control room to initiate isolation. A trip of an SCI instrumentation channel is annunciated and position indication is provided for both motor-operated and

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

air-operated isolation valves in the control room. SCI instrumentation is interlocked to initiate the SGTS, isolate the reactor building ventilation system, and trip the drywell purge valves and purge fan units. The required channels of SCI instrumentation functions provide plant protection in the event of any of the analyzed accidents.

SCI instrumentation functions are as follows:

1. Reactor Vessel Water Level--Low Low, Level 2

Low reactor vessel water level indicates that the capability to cool the fuel may be threatened. Should reactor vessel water level decrease too far, fuel damage could result. An isolation of the secondary containment and actuation SGTS is initiated in order to minimize the potential of an offsite dose release. The Reactor Vessel Water Level--Low Low, Level 2 function is assumed to be OPERABLE and capable of providing isolation and actuation signals if a break has occurred in the lines that are isolated. The isolation and actuation of systems on reactor vessel water level 2 supports actions to ensure that SLs are not exceeded. The Reactor Vessel Water Level--Low Low, Level 2 function associated with isolation of secondary containment is implicitly assumed in safety analyses.

2. Drywell Pressure--High

High drywell pressure could indicate a break in the reactor coolant pressure boundary. An isolation of the secondary containment and actuation of an SGTS is initiated in order to minimize the potential of an offsite dose release. The isolation of some of the systems on high drywell pressure supports actions to ensure that SLs are not exceeded. The Drywell Pressure--High function associated with isolation is assumed in the analysis of the recirculation line break (Ref. 5).

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

3,4. Reactor Building Exhaust Radiation--High and Refueling Floor Exhaust Radiation--High

The following discussion applies to both the RB Exhaust Radiation--High and Refueling Floor Exhaust Radiation--High functions.

High RB exhaust radiation is an indication of possible gross failure of the fuel cladding. The release may have originated in the fuel pool or the reactor vessel. When high RB exhaust radiation is detected, secondary containment closure is initiated to limit the release of fission products.

5. Manual Initiation

The Manual Initiation push-button switches introduce signals into the appropriate system's isolation logic that are redundant to the automatic protective instrumentation channels and provide manual isolation capability. The push button arrangement is a plant-unique design, but the Technical Specifications were written for an arrangement that has one push button per [group of dampers].

The secondary containment isolation instrumentation satisfies Criteria 1, 2, and 3 of the NRC Interim Policy Statement.

LCO

The LCO requires all instrumentation performing an SCI function to be OPERABLE. Failure of any instrument renders the affected channel(s) inoperable and reduces the reliability of the affected functions.

The OPERABILITY of the SCI instrumentation is dependent on the OPERABILITY of the individual instrumentation channel functions. Each function must have the number of OPERABLE channels per trip system as shown in Table 3.3.6.2-1 with their setpoints set within the specified ALLOWABLE VALUES. Actuation setpoints are calibrated consistent with applicable setpoint methodology assumptions. Each channel must also respond within its assumed response time.

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BASES (continued)

LCO (continued)

Only the ALLOWABLE VALUES are specified for each function in the LCO. Nominal trip setpoints are specified in the plant-specific setpoint calculations of Reference 3. The nominal setpoints are selected to ensure that 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 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 of Reference 3. Each channel must also respond within its assumed response time.

A channel is OPERABLE when the following conditions are satisfied:

1. All channel components necessary to provide a trip signal are functional and in service;

2. Channel measurement uncertainties are known via test, analysis, or design information to be within the assumptions of the setpoint calculations;

3. Required surveillance testing is current and has demonstrated performance within each surveillance test's acceptance criteria; and

The SCI instrumentation has inputs to the trip logic from the functions listed below:

1. <u>Reactor Vessel Water Level--Low Low, Level 2</u>

   Reactor vessel water level 2 signals are initiated from level transmitters which are arranged on two sets of separated taps with a logic that initiates on a one-out-of-two per trip system. Four channels of Reactor Vessel Water Level--Low Low, Level 2 function are available and required to be OPERABLE to ensure that no single failure can preclude the isolation

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BASES (continued)

LCO (continued)

function. The Reactor Vessel Water Level--Low Low, Level 2 ALLOWABLE VALUE was selected to be the same as the level at which high pressure coolant injection and reactor core isolation cooling functions are initiated. HPGI and RGIC initiation could indicate that the capability to cool the fuel is being threatened. This could result in the need for secondary containment to minimize the potential for occurrence of an offsite release of radioactivity.

The Reactor Vessel Water Level--Low Low, Level 2 function is required to be OPERABLE in MODES 1, 2, and 3 where considerable energy exists in the Reactor Coolant System (RCS). In addition, the instrumentation must be OPERABLE during CORE ALTERATIONS and operations with a potential for draining the reactor vessel because the capability of isolating potential sources of leakage must be provided to ensure core coverage.

This function isolates the [secondary containment automatic isolation dampers] and starts the SGTS.

2. Drywell Pressure--High

High drywell pressure signals are initiated from pressure transmitters that sense the pressure at four different locations in the drywell. The transmitters are located outside the drywell. Four channels of Drywell Pressure--High functions with a logic which initiate on one-out-of-two per trip system are available and required to be OPERABLE to assure that no single failure can preclude the isolation function. The ALLOWABLE VALUE was selected to be the same as the Emergency Core Cooling System instrumentation Drywell Pressure--High function ALLOWABLE VALUE, since this value could indicate a break in the RCPB and the consequent need for SCI.

The Drywell Pressure--High function is required to be OPERABLE in MODES 1, 2, and 3 where considerable energy exists in the RCS. In MODES 4 and 5, the reactor is shut down and any LOCA would not cause pressurization of the drywell.

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BASES (continued)

LCO (continued)

This function isolates the [Unit 2] secondary containment automatic isolation dampers [RB dampers] and starts the [Unit 2] SGTS.

3,4. Reactor Building Exhaust Radiation--High and Refueling Floor Exhaust Radiation--High

The following discussion applies to both the RB Exhaust Radiation--High and Refueling Floor Exhaust Radiation--High functions.

The RB exhaust radiation signals are initiated from radiation detectors that are located on the ventilation exhaust piping coming from the reactor building and the [Unit 2] refueling floor zones, respectively. The radiation detectors are arranged so that, even though physically separated from each other, each detector detects some radiation in a postulated accident. The signal from each detector is input to an individual monitor whose trip outputs are assigned to an isolation channel. Four channels of RB Exhaust Radiation--High function and four channels of refueling floor exhaust radiation with a logic that initiates on one-out-of-two per trip system are available and required to be OPERABLE to assure that no single failure can preclude the isolation function. The ALLOWABLE VALUES are high enough above background radiation levels to minimize spurious trips, yet low enough to promptly detect gross failure of the fuel cladding.

The RB Exhaust Radiation--High trip function is required to be OPERABLE in MODES 1, 2, and 3 where considerable energy exists. When the potential exists for draining the vessel or when fuel handling is allowed, the equipment is required to be OPERABLE to detect radiation release from postulated fuel failures.

The RB Exhaust Radiation--High function isolates the [Unit 2] SCI dampers [Unit 2 RB dampers] and starts the [Unit 2] SGTS. The Refueling Floor Exhaust Radiation--High function isolates the [Unit 1] SCI dampers [the Unit 1 RB and refueling floor dampers

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BASES (continued)

LCO (continued)

and the Unit 2 refueling floor dampers] and starts the building and refueling floor dampers] and starts the [Unit 1 and Unit 2] SGTS.

5. Manual Initiation

The Manual Initiation push button switches introduce signals into the appropriate system's isolation logic that are redundant to the automatic protective instrumentation channels and provide manual isolation capability. [For this facility, the push button arrangement is as follows:]

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

[For this facility, those required support systems which, upon their failure, do not result in the SCI instrumentation being declared inoperable and their justification are as follows:]

[For this facility, the supported systems impacted by the inoperability of the SCI instrumentation and the justification of whether or not each supported system is declared inoperable are as follows:]

APPLICABILITY

In general, the individual functions are required to be OPERABLE in the modes or other specified conditions when SCI valves and the SGTS are required. Refer to the LCO section of the Bases for the specific APPLICABILITY requirements of each function.

ACTIONS

A protection function 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

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BASES (continued)

ACTIONS (continued)

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 to within specification. If the trip setpoint is less conservative than the ALLOWABLE VALUE in Table 3.3.6.2-1, the channel must be declared inoperable immediately and the appropriate Conditions from Table 3.3.6.2-1 must 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 bistable 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.

In order for a facility to take credit for topical reports as the basis for justifying Completion Times, topical reports should be supported by an NRC staff Safety Evaluation Report that establishes the acceptability of each topical report for that facility.

Condition A

Condition A applies to each of the SCI functions on Table 3.3.6.2-1. This includes all automatic valves designed to receive an isolation signal assumed by the safety analyses to initiate closure of valves to limit offsite doses. It provides Required Actions that ensure the integrity of the penetration isolation capability and provides for evaluation of SCI instrumentation inoperability, including time for repairs subsequent to placing the channel in trip.

With more than one channel inoperable for any SCI instrumentation function, either by itself or together with single channel failures in other SCI instrumentation functions, the SCI instrumentation may not be capable of performing its intended function.

A Note is added to Condition A to indicate that the Completion Time shall be on a Condition basis for each function.

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BASES (continued)

ACTIONS (continued)

A.1

If one or more channels are inoperable for one or more functions, operation may continue provided each function is still capable of isolating the associated lines. Required Action A.1 requires an evaluation to ensure the isolation capability exists for each function. "Maintains isolation capability" refers to the ability to provide automatic and manual isolation of all affected lines, with OPERABLE and tripped channels. For a four-channel function, one-out-of-two logic for each of two valves per line, this would require only one OPERABLE channel in each function (i.e., only one valve per line required to isolate). This Required Action can be met with three inoperable channels if the inoperable channel in the trip system containing only one inoperable channel is placed in trip.

The Completion Time of 1 hour is sufficient time for plant operations personnel to complete Required Action A.1.

A.2.1

Required Action A.2.1 is the preferred action because it restores full functional capability to the SCI instrumentation.

A.2.2

As an alternative to restoring a channel(s) to OPERABLE status per Required Action A.2.1, operation is allowed to continue provided the inoperable channel is placed in trip per Required Action A.2.2. Required Action A.2.2 is modified by a Note that the Required Action does not apply if it results in an isolation. If a channel in one trip system becomes inoperable when one or more channels in the opposite system are already in trip, placing the inoperable channel in trip will cause an isolation. Required Action A.2.2 is not intended to force an unnecessary isolation. In this event, Required Action A.2.1 would have to be met. If the inoperable channel(s) are not restored to OPERABLE status or placed in trip within the time allowed, Condition B should be entered. The bases for the 12-hour and 24-hour Completion Times provided to allow continued

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BASES (continued)

ACTIONS (continued)

operation with inoperable channels while taking action to restore or trip the inoperable channel(s) are in Reference 6.

Condition B

If the Required Actions of Condition A or B are not met, Condition B normally accomplishes the function of the SCI instrumentation.

One hour is sufficient for plant operations personnel to establish required plant conditions or isolate the associated zone(s) or to declare the associated SCI valve(s) inoperable or declare the associated SGTS subsystem inoperable without unnecessarily challenging plant systems.

Condition C

Condition C is applicable to each one of the SCI instrumentation functions presented in Table 3.3.6.2-1.

C.1 and C.2

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 verification.

Required Action C.1 ensures that those identified Required Actions associated with supported systems impacted by the inoperability of channel(s) associated with each SCI instrumentation 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 Required Actions associated with each SCI instrumentation 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 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 SCI instrumentation function are found in the SR column of Table 3.3.6.2-1 for that function. Most functions are subject to CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION.

In order for a facility to take credit for topical reports as the basis for justifying Surveillance Frequencies, topical reports should be supported by an NRC staff Safety Evaluation Report that establishes the acceptability of each topical report for that facility.

The SRs are modified by a Note to indicate that a channel may be placed in an inoperable status for up to 6 hours for required SRs provided one OPERABLE channel in the same trip system is monitoring that parameter. Upon completion of the Surveillance or expiration of the 6-hour allowance, the channel must be returned to OPERABLE status or the Required Actions taken.

It is not acceptable to routinely remove channels from service for more than 6 hours to perform required surveillance testing. Such a practice would be contrary to the assumptions of the reliability analysis that justified the LCO Completion Times.

SR 3.3.6.2.1

SR 3.3.6.2.1 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.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or something even more serious. CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION. 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 SRs are 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 outright 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 required channels by the LCO.

SR 3.3.6.2.2

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, interlocks, and alarms function when the input is beyond the trip point.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

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 VALUES 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. If during a CHANNEL FUNCTIONAL TEST the associated trip setting is discovered to be less conservative than the ALLOWABLE VALUE specified in Table 3.3.6.1-1, the channel must be declared inoperable.

A Note excludes radiation detectors from this SR. [For this facility, the basis for excluding radiation detectors is as follows:]

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

SR 3.3.6.2.3

The calibration of bistable trip units consists of a test to determine the actual trip setpoints, and recalibration of the setpoint is necessary to ensure that it remains more conservative than the setpoint and the ALLOWABLE VALUE specified in Table 3.3.6.2-1. The channel must be declared inoperable if the setting is discovered to be less conservative than the ALLOWABLE VALUE. If the trip setting is discovered to be less conservative than accounted for in the appropriate setpoint methodology but is not beyond the ALLOWABLE VALUE, the channel is still considered OPERABLE. Under these conditions, the setpoint must be readjusted to be more conservative than accounted for in the appropriate setpoint methodology.

The Surveillance Frequency is based on the assumption of a 92-day calibration interval in the determination of the magnitude of trip unit drift in the setpoint analysis (Ref. 3).

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.6.2.4

CHANNEL CALIBRATION is a complete check of the instrument channel including the detector. 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 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 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.

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 3.3.6.2.5

Performance of a LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required actuation logic for a specific channel. The LOGIC SYSTEM FUNCTIONAL TEST tests all logic components (i.e., all relays and contacts, trip units, solid-state logic elements) of a logic circuit, from sensor up to the actuated device. [For this facility, the system functional testing performed on isolation valves

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SURVEILLANCE REQUIREMENTS (continued)

overlaps this test to provide complete testing of the assumed safety function as follows:]

The 18-month Surveillance Frequency is based on experience that it is prudent that these SRs be performed only during a plant outage. This is due to the plant conditions needed to perform the Surveillance and the potential for system isolation if the surveillance is performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed on the 18-month Frequency.

SR 3.3.6.2.6

SR 3.3.6.2.6 is a CHANNEL FUNCTIONAL TEST performed on the secondary containment isolation manual initiation function. This test verifies that each initiation switch isolates the associated group of values as designed. The 18-month Surveillance Frequency was developed considering it is prudent that these SRs be performed only during a plant outage. This is 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 that these components usually pass the Surveillance when performed on the 18-month Frequency.

SR 3.3.6.2.7

This SR ensures that the ISOLATION SYSTEM RESPONSE TIME for each channel is less than or equal to the maximum value 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 system isolation. The acceptable response times of the relevant trip channels are given below. [The response times include contributions from the following equipment:] 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 radiation detectors may be excluded from ISOLATION SYSTEM RESPONSE TIME testing. This Note is necessary because of the

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

difficulty of generating an appropriate detector input signal. [At this facility, the basis for allowing exclusion of radiation detectors from response time testing is as follows:]

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 x 18 months, where n is the number of channels per trip system 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 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 since equipment operation is required.

[SR 3.3.6.2.8]

A CHANNEL CALIBRATION is required to be performed on a 92-day Frequency. The basis for a CHANNEL CALIBRATION was previously discussed in SR 3.3.6.2.4. [For this facility, the basis for the 92-day Frequency is as follows:]

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, "Domestic Licensing of Production and Utilization Facilities."
2. [Unit Name] FSAR, Section [ ], "[Title]."
3. [Unit Name] "[Plant Specific Setpoint Methodology]."
4. [Unit Name] FSAR, Section [7], "[Instrumentation and Control]."
5. [Unit Name] FSAR, Section [15], "[Title]."

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BASES (continued)

REFERENCES (continued)

6. NEDC-31677-P-A, "Technical Specification Improvement Analysis for BWR Isolation Actuation Instrumentation," June 1989.

7. Title 10, Code of Federal Regulations, Part 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants."

BASES (continued)

ACTIONS (continued)

does not apply if it results in an isolation. If a channel in one trip system becomes inoperable when one or more channels in the opposite system are already in trip, placing the inoperable channel in trip will cause an isolation. Required Action A.2.2 is not intended to force an unnecessary isolation. In this event, Required Action A.2.1 would have to be met. If the inoperable channel(s) are not restored to OPERABLE status or placed in trip within the time allowed, Condition B should be entered. The bases for the 12-hour and 24-hour Completion Times provided to allow continued operation with inoperable channels while taking actions to restore or trip the channels are in Reference 6.

Condition B

Condition B applies to each of the PCI instrumentation functions listed in Table 3.3.6.1-1.

B.1

This Required Action directs entry into all remaining Conditions (except A or B). When any Required Action of Condition A is not met, and the associated Completion Time has expired, the applicable Condition from Table 3.3.6.1-1 must be entered for function(s) whose inoperable channel(s) have not been placed in trip. The applicable Condition specified in Table 3.3.6.1-1 is function- and mode-dependent and may change as the Required Action of a previous Condition is completed.

Condition C

Condition C is applicable to the main steam line isolation functions except low pressure.

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

If the required number of channels are not restored to OPERABLE status or placed in trip within the allowed Completion Time, the plant must be placed in a mode or other specified condition in which the LCO does not apply. The Completion Time of 6 hours to isolate all MSIVs is reasonable, based on operating experience, to initiate actions for a controlled reduction in power to establish

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BASES (continued)

ACTIONS (continued)

required plant conditions and isolate the main steam lines (equivalent to MODE 2) in an orderly manner and without challenging plant systems.

Twelve hours and 36 hours are reasonable, based on operating experience, to reach MODE 3 and MODE 4, respectively, in an orderly manner and without challenging plant systems.

Condition D

Condition D is applicable to the main steam line low-pressure isolation function.

D.1

If the required number of channels are not restored to OPERABLE status or placed in trip within the allowed Completion Time, the plant must be placed in a mode or other specified condition in which the LCO does not apply.

Six hours is reasonable, based on operating experience, to initiate actions for a controlled reduction in power and to reach MODE 2 in an orderly manner and without challenging plant systems.

Condition E

Condition E is applicable to PCI drywell radiation; HPCI and RCIC isolation functions; RWCU isolation functions except SLCS initiation and shutdown cooling isolation on high reactor vessel pressure.

E.1

If the required number of channels are not restored to OPERABLE status or placed in trip within the allowed Completion Time, plant operations may continue if the affected lines are isolated. Isolating the affected lines accomplishes the safety function of the inoperable channels. The 1-hour Completion Time is sufficient for plant operations personnel to take the appropriate action.

If, for some reason, these Required Actions cannot be completed within the time allowed, action must be taken in

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BASES (continued)

ACTIONS (continued)

accordance with Condition F to place the plant in a MODE or Condition in which the inoperable functions are not required.

Condition F

Condition F is applicable to the Required Actions of Condition E if they are not met, PCI Level 3, drywell pressure and each manual initiation function in Table 3.3.6.1-1.

F.1 and F.2

If the required number of channels are not restored to OPERABLE status or placed in trip within the allowed Completion Time, the plant must be placed in a mode or other specified condition in which the LCO does not apply. Twelve hours and 36 hours are reasonable, based on operating experience, to initiate actions for a controlled reduction in power and to reach MODE 3 and MODE 4, respectively, in an orderly manner and without challenging plant systems.

Condition G

Condition G is applicable to RWCU SLCS initiation.

G.1 and G.2

If the required number of channels are not restored to OPERABLE status or placed in trip within the allowed Completion Time, the associated SLCS subsystems are declared inoperable or the RWCU System is isolated. Since this function is required to ensure that the SLCS performs its intended function, sufficient remedial measures are provided by declaring the associated SLCS inoperable or isolating the RWCU System. The 1-hour Completion Time is sufficient for operations personnel to perform the corrective actions.

Condition H

Condition H is applicable to the shutdown cooling level 3 function.

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BASES (continued)

ACTIONS (continued)

H.1

[For this facility, the basis for the Required Action and the 1-hour Completion Time is as follows:]

Condition I

Condition I is applicable to each one of the primary containment isolation instrumentation functions presented in Table 3.3.6.1-1.

I.1 and I.2

Required Action I.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 verification.

Required Action I.1 ensures that those identified Required Actions associated with supported systems impacted by the inoperability of channel(s) associated with each primary containment isolation instrumentation 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 I is entered. [For this facility, the identified supported systems Required Actions associated with each PCI instrumentation function are as follows:]

Required Action I.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.

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BASES (continued)

SURVEILLANCE REQUIREMENTS

The SRs for any particular PCI instrumentation function are found in the SR column of Table 3.3.6.1-1 for that function. Most functions are subject to CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION.

In order for a facility to take credit for topical reports as the basis for justifying Surveillance Frequencies, the topical reports should be supported by an NRC staff Safety Evaluation Report that establishes the acceptability of each topical report for that facility.

The SRs are modified by Note 2 to indicate that a channel may be placed in an inoperable status for up to 6 hours for required surveillances provided at least one OPERABLE channel in the same trip system is monitoring that parameter. Upon completion of the surveillance, or expiration of the 6-hour allowance, the channel must be returned to OPERABLE status or the Required Actions taken. It is not acceptable to routinely remove channels from service for more than 6 hours to perform required surveillance testing. Such a practice would be contrary to the assumptions of the reliability analysis that justified LCO Completion Times.

SR 3.3.6.1.1

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 against the indication of the same parameter on other instrument channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure, thus it is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION. 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 operated from the control room, the CHANNEL CHECK should also note the detector's response to these sources.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

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 surveillances are 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 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 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 channels required by the LCO.

SR 3.3.6.1.2

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 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 must be found within the ALLOWABLE VALUES 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

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

over a small number of test intervals should be evaluated as potentially indicating a deterministic failure which cannot be corrected by recalibration. If, during a CHANNEL FUNCTIONAL TEST, the associated trip setting is discovered to be less conservative than the ALLOWABLE VALUE specified in Table 3.3.6.1-1, the channel must be declared inoperable.

The 92-day Surveillance Frequency is based on the reliability analysis described in Reference 6. A Note modifies this SR, which permits radiation detectors to be exempt from the CHANNEL FUNCTIONAL TEST requirements. [For this facility, the basis for excluding radiation detectors is as follows:]

SR 3.3.6.1.3

The calibration of bistable trip units consists of a test to determine the actual trip setpoints, and recalibration of the setpoint is necessary to ensure that it remains more conservative than the setpoint and the ALLOWABLE VALUE specified in Table 3.3.6.1-1. The channel must be declared inoperable if the setting is discovered to be less conservative than the ALLOWABLE VALUE. If the trip setting is discovered to be less conservative than accounted for in the appropriate setpoint methodology but is not beyond the ALLOWABLE VALUE, the channel is still considered OPERABLE. Under these conditions, the setpoint must be readjusted to be more conservative than that accounted for in the appropriate setpoint methodology.

The Surveillance Frequency is based on the assumption of a 92-day calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis of Reference 3.

SR 3.3.6.1.4

CHANNEL CALIBRATION is a complete check of the instrument channel including the detector. 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 with the setpoint within the assumptions of the plant-specific setpoint analysis. Transmitter

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

"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 setpoints 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 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 the assumption of a 92-day calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis (Ref. 6).

SR 3.3.6.1.5

The basis for performance of a CHANNEL CALIBRATION was previously discussed in SR 3.3.6.1.4.

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) or thermocouple (T/C) 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 or T/C should be replaced with a newly calibrated sensor during each refueling cycle to ensure accurate sensor cross-calibration. This replacement sensor must be the same model as the remaining RTDs or T/Cs. Using a newly calibrated sensor as a reference ensures that 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 sensor drift is random rather than

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

systematic, and is bounded by the plant-specific setpoint analysis assumptions. This determination may use results of statistical analyses of operating data and calibration data from similar plants using the same model of RTD or T/C 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 in the setpoint analysis (Ref. 3).

SR 3.3.6.1.6

Performance of a LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required actuation logic for a specific channel. The LOGIC SYSTEM FUNCTIONAL TEST tests all logic components (i.e., all relays and contacts, trip units, solid-state logic elements, etc.) of a logic circuit, from sensor up to the actuated device. [For this facility, the system functional testing performed on isolation valves overlaps this test to provide complete testing of the safety function as follows:] The Surveillance Frequency is based on experience that it is prudent that these surveillances be performed only during a plant outage. This is due to the plant conditions needed to perform the surveillance and the potential for system isolation during the LOGIC SYSTEM FUNCTIONAL TESTS. Operating experience has shown these components usually pass the surveillance when performed at the 18-month Frequency.

SR 3.3.6.1.7

SR 3.3.6.1.7 is a CHANNEL FUNCTIONAL TEST performed in the PCI manual isolation functions. This test verifies each initiation switch isolates the associated groups of valves as designed.

The 18-month surveillance interval is based upon experience that it is prudent that these surveillances only be performed during a plant outage. This is due to the plant conditions needed to perform the surveillance and the potential for system isolation 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.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.6.1.8

The basis for performance of a CHANNEL FUNCTIONAL TEST was previously discussed in SR 3.3.6.1.2.

[For this facility, the basis for the 184-day Surveillance Frequency is as follows:]

SR 3.3.6.1.9

This SR ensures that the ISOLATION SYSTEM RESPONSE TIMES for each channel 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 system isolation. [For this facility, the basis for the acceptable response times of the relevant trip channels are as follows: The response times include contributions from the following equipment:] 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 radiation detectors may be excluded from ISOLATION SYSTEM RESPONSE TIME testing. This note is necessary because of the difficulty of generating an appropriate detector input signal. [For this facility, the basis for excluding the detectors is acceptable as follows:]

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 x 18 months, where n is the number of channels per trip system 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 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. Response times cannot be determined at power since equipment operation is required.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, "Domestic Licensing of Production and Utilization Facilities."

2. [Unit Name] FSAR, Section [9], "[Title]."

3. [Unit Name] FSAR, "Plant-Specific Setpoint Methodology."

4. [Unit Name] FSAR, Section [7], "Instrumentation and Control]."

5. [Unit Name] FSAR, Section [15] "Title]."

6. NEDC-31677-P-A, "Technical Specification Improvement Analysis for BWR Isolation Actuation Instrumentation," June 1989.

7. Title 10, Code of Federal Regulations, Part 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for light water nuclear power reactors."

8. Title 10, Code of Federal Regulations, Part 50.49, "Environment Qualification of Electrical Equipment Important to Safety for Nuclear Power Plants."

B 3.3 INSTRUMENTATION

B 3.3.6.2 Secondary Containment Isolation (SCI) Instrumentation

BASES

BACKGROUND

The SCI instrumentation automatically initiates closure of appropriate unit SCI valves and starting of the unit's Standby Gas Treatment System (SGTS). This function is necessary to prevent or limit the release of fission products from the secondary containment in the event of a loss-of-coolant accident (LOCA) or a reactor coolant pressure boundary (RCPB) leak.

The SCI instrumentation includes the sensors, relays, and switches that are necessary to cause initiation of SCI. Functional diversity is provided by monitoring a wide range of dependent and independent parameters. The input parameters to the isolation logics are electrical signals that indicate limits on reactor vessel water level and drywell pressure. Other inputs into the isolation logic are reactor building (RB) exhaust and refueling floor exhaust high radiation. Redundant sensor input signals are provided from each of the SCI initiation parameters.

The SCI instrumentation is designed to include the subsystems identified below:

- Field transmitters or process sensors;
- Signal processing and bistable modules; and
- Trip logic, trip setpoints, and ALLOWABLE VALUES.

Field Transmitters or Process Sensors

Field transmitters or process sensors provide a measurable electronic output signal based on the physical characteristics of the parameter being measured.

Typically, four measurement channels with physical separation are provided for each parameter. These are typically organized into two trip systems which are physically and electrically separated. Four measurement channels are necessary to meet the redundancy and testability criteria of 10 CFR 50, Appendix A, GDC 21

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BASES (continued)

BACKGROUND (continued)

(Ref. 1), and to implement the one-out-of-two taken twice logic arrangement discussed below for the SCI instrumentation.

[For this facility, a discussion of those SCI parameters which do not have four measurement channels and their conformance to redundancy and testability requirements of GDC 21 in 10 CFR 50, Appendix A, is as follows:]

For most anticipated occupational occurrences (AOOs) and Design Basis Accidents (DBAs), a wide range of dependent and independent parameters are monitored.

Signal Processing and Bistable Modules

Each process parameter measurement channel includes electronic equipment that provides signal conditioning, comparable output signals for main control board instruments, comparison of measured input signals with setpoints established by safety analyses, and output to the trip logic channels. This output to the trip logic channels is taken from a bistable device which can be mechanical switches that are part of the process sensors or electronic comparators that receive input from the process transmitters or sensors. In either case, the bistable output contacts are considered to be part of the trip logic channel.

Trip Logic, Trip Setpoints, and ALLOWABLE VALUES

Trip setpoints are those predetermined values of output voltage or current against which the output voltage or current related to the present value of the process parameter is compared. If the present measured output value of the process parameter exceeds the setpoint, the associated bistable changes state. The trip setpoints are the nominal values at which the bistables are set. They are derived from the limiting values of the process parameters obtained from the accident analyses (analytical limits) through a process of correction for uncertainties and errors set forth in the plant-specific setpoint methodology (Ref. 3). The analytical limits, corrected for analytical and process uncertainties, become the ALLOWABLE VALUES, which when further corrected by the methodology of Reference 3 become the calculated trip setpoint values.

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BASES (continued)

BACKGROUND (continued)

The setpoints derived in this manner provide adequate protection because sensor and processing time delays are accounted for as well as calibration tolerances, instrumentation uncertainties, instrument drift, and severe environment errors for channels that must function in harsh environments as defined by 10 CFR 50.49 (Ref. 7). The actual nominal trip setpoint entered into the bistable is usually still more conservative than that calculated by the plant-specific setpoint methodology. If the setpoint measured for the bistable by the surveillance test does not exceed the documented surveillance test acceptance criteria, the bistable is considered OPERABLE.

The outputs of the logic channels in a trip system are combined in a logic so that both channels are required to trip the associated trip system. The logic is one-out-of-two for each trip system. Typically, automatically isolated secondary containment penetrations are isolated by two isolation values, so that operation of either trip system isolates the penetration.

The trip setpoints are the nominal 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 and comparator setting accuracy).

The trip setpoints used in the bistables are based on the analytical limits stated in Reference 3. 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 SCI channels that must function in harsh environments, as defined by 10 CFR 50.49 (Ref. 7), ALLOWABLE VALUES, specified in Table 3.3.6.1-1, 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 the plant-specific setpoint methodology. The actual nominal trip setpoint entered into the bistable is normally more conservative than that required by the plant-specific setpoint calculations. If

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BASES (continued)

BACKGROUND (continued)

the measured setpoint does not exceed the documented surveillance test acceptance criteria, the bistable is considered OPERABLE.

Setpoints set in accordance with the ALLOWABLE VALUE will ensure that the consequences of DBAs will be acceptable, providing the plant is being operated within the LCOs at the onset of the AOO or DBA, and the equipment functions as designed, allowing for a single random active component failure.

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

The ALLOWABLE VALUES listed in Table 3.3.6.2-1 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.

APPLICABLE SAFETY ANALYSES

The actions of the instrumentation are implicitly assumed in the safety analyses of References 2, 4, and 5. The isolation initiates closure of valves to ensure secondary containment OPERABILITY and limit offsite doses.

The SCI instrumentation automatically isolates the appropriate pipelines that penetrate the secondary containment whenever the monitored parameters exceed preselected setpoints. System level manual switches are also provided in the control room to initiate isolation. A trip of an SCI instrumentation channel is annunciated and position indication is provided for both motor-operated and

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BASES (continued)

LCO (continued)

The ALLOWABLE VALUE is set to be the same as the Main Steam Line Flow--High function in LCO 3.3.6.1, "PCI Instrumentation."

The Main Steam Line Flow--High function is required to be OPERABLE in MODES 1, 2, and 3 to ensure control room personnel are protected during a main steam line break (MSLB) LOCA. In MODES 4 and 5, the reactor is depressurized; thus, MSLB LOCA protection is not required.

4. Refueling Floor Area Radiation--High

High radiation in the refueling floor area could be the result of a fuel handling accident or the release of gaseous activity from the suppression pool due to mixing of the pool water with reactor coolant. A refueling floor high radiation signal will automatically initiate the MCREC.

The refueling floor area radiation equipment consists of two independent monitors and channels located in the refueling floor area. Either channel will initiate the MCREC System in the pressurization mode. Two channels of Refueling Floor Area Radiation--High function are available and are required to be OPERABLE and ensure that a single failure will not prevent the protective action. [For this facility, the logic configuration and the basis for the logic configuration is as follows:] The ALLOWABLE VALUE is set just high enough to prevent spurious trips.

The Refueling Floor Area Radiation--High function is required to be OPERABLE in MODES 1, 2, and 3, and when handling irradiated fuel assemblies or loads over irradiated fuel in the [primary or secondary] containment and during CORE ALTERATIONS or OPDRV to ensure control room personnel are protected.

5. Control Room Air Inlet Radiation--High

The control room air inlet radiation monitors measure radiation levels exterior to the inlet ducting of the

(continued)

(continued)

BASES (continued)

LCO (continued)

MCR. A high radiation level may pose a threat to MCR personnel, thus automatically initiating the MCREC System.

Control Room Air Inlet Radiation--High is assumed to be OPERABLE and capable of automatically initiating the MCREC. The monitors are implicitly assumed to function in the FSAR accident analyses.

The control room air inlet radiation equipment consists of two independent monitors and channels. Either channel will initiate the MCREC System in the pressurization mode. Two channels of Control Room Air Inlet Radiation--High are available and are required to be OPERABLE to ensure that a single failure will not prevent the protective action. [For this facility, the logic configuration and the basis for the trip logic configuration is as follows:] The ALLOWABLE VALUE is just high enough to prevent spurious actuation of the MCREC System.

The Control Room Air Inlet Radiation--High Function is required to be OPERABLE in MODES 1, 2, and 3, during CORE ALTERATIONS or OPDRV, and during handling of irradiated fuel assemblies or loads over irradiated fuel in the primary or secondary containment to ensure protection of control room personnel.

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

[For this facility, those required support systems which, upon their failure, do not result in the MCREC instrumentation being declared inoperable and their justification are as follows:]

[For this facility, the supported systems impacted by the inoperability of the MCREC instrumentation and the justification of whether or not each supported system is declared inoperable are as follows:]

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BASES (continued)

APPLICABILITY

Refer to Table 3.3.7.1-1 for the specific Applicability requirements of each function. The bases for Applicability requirements are discussed on a function-by-function basis in the LCO section.

A Note is added to provide clarification that for this LCO, each function specified in Table 3.3.7.1-1 shall be treated as an independent entity with an independent Completion Time.

ACTIONS

In order for a facility to take credit for topical reports for the basis for justifying Completion Times, topical reports should be supported by an NRC staff Safety Evaluation Report (SER) that establishes the acceptability of each topical report for that facility.

A protection function 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 in Table 3.3.7.1-1, the channel must be declared inoperable immediately, and the appropriate Conditions from Table 3.3.7.1-1 must be entered immediately.

In the event a channel's trip setpoint is found nonconservative with respect to the ALLOWABLE VALUE, or the transmitter, instrument loop, signal processing electronics, or bistable 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.

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BASES (continued)

ACTIONS (continued)

Condition A

Condition A is applicable to each of the functions noted in Table 3.3.7.1-1. With a channel of the instrumentation inoperable in a trip system, the trip system is still capable of performing its intended function, provided the other channel of the same function is OPERABLE. [Note, this assumes a one-out-of-two actuation logic for each train.] The 1-hour Completion Time is sufficient for the operator to verify functional trip capability for each trip system.

Required Action A.2 directs entry to all other Conditions referenced in Table 3.3.7.1-1. The applicable Condition specified in the table is function dependent. Each time an inoperable channel is discovered, Condition A is entered and the Required Action of the appropriate subsequent Condition is taken.

Condition B

Condition B applies to the failure of a single channel of the ECCS common reactor vessel water level 1 function.

Required Action B.1 is the preferred Action because it restores full functional capability of the function. Required Action B.2 allows the inoperable channel(s) to be placed in trip. This Required Action is modified by a Note that the Required Action does not apply if it results in MCREC actuation. If a channel in one trip system becomes inoperable when one or more channels in the opposite system are already in trip, placing the inoperable channel in trip will cause an actuation. Required Action B.2 is not intended to force an unnecessary actuation. In this event, Required Action B.1 would have to be met. If the inoperable channel(s) are not restored to OPERABLE status or placed in trip within the time allowed, Condition E should be entered. For this facility, an allowed outage time of 24 hours is acceptable, according to the justification provided in Reference 2.

Condition C

Condition C applies to the failure of a single channel of the RPS common drywell high pressure function.

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BASES (continued)

ACTIONS (continued)

Required Action C.1 is the preferred Action because it restores full functional capability of the function. Alternately, Required Action C.2 allows the inoperable channel(s) to be placed in trip. A Note indicates that this Required Action is only applicable if placing inoperable channel(s) in trip would not result in initiation. Required Action C.2 is not intended to force an unnecessary actuation. In this event, Required Action C.1 would have to be met or Required Action E would have to be entered. For this facility, the 12-hour Completion Time is acceptable, based on the justification provided in Reference 2.

Condition D

Condition D applies to the failure of radiation instrumentation functions. For this facility, the 6-hour Completion Time is based on the consideration that radiation monitor OPERABILITY is the primary success path in Design Basis Accident (DBA) or transient analyses. Required Action D.1 is the preferred Action because it restores full functional capability of the function. Alternately, Required Action D.2 allows the inoperable channel(s) to be placed in trip. A Note indicates that this Required Action is only applicable if placing inoperable channel(s) in trip would not result in an MCREC initiation. Required Action D.2 is not intended to force an unnecessary actuation. In this event, Required Action D.1 would have to be met or Required Action E would have to be entered.

Condition E

The associated MCREC subsystem must be declared inoperable if Required Actions of Condition A, B, C, or D are not met and associated Completion Times are not met within 1 hour. This is necessary to ensure that control room personnel will be protected in the event of an accident or if the associated subsystem(s) are declared inoperable. The Completion Time of immediately recognizes the fact that Condition E, in the limit, addresses complete failure of the MCREC function.

Condition F

Condition F is applicable to each of the MCREC instrumentation functions in Table 3.3.7.1-1.

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BASES (continued)

ACTIONS (continued)

Required Action F.1 verifies that the Required Actions have been initiated for those supported systems declared inoperable because of the inoperability of the support channels(s) or within a Completion Time of 1 hour. The specified Completion Time is sufficient for plant operations personnel to make this determination.

Required Action F.1 ensures that those identified Required Actions associated with supported systems impacted by the inoperability of channel(s) associated with each MCREC instrumentation 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 F is entered. [For this facility, the identified supported systems' Required Actions associated with each MCREC instrumentation function are as follows:]

Required Action F.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 take into consideration the loss of function situation, then LCO 3.0.3 may not need to be entered.

SURVEILLANCE REQUIREMENTS

The SRs for any MCREC instrumentation function are found in the SR column of Table 3.3.7.1-1 for that function. All functions are subjected to CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, CHANNEL CALIBRATION, and LOGIC SYSTEM FUNCTIONAL TEST.

In order for facilities to take credit for topical reports for the basis for justifying Surveillance Frequencies, topical reports should be supported by an NRC SER that establishes the acceptability of each topical report for that facility.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.7.1.1

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 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 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 Surveillances are 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 interval, about once every shift, is based on operating experience that demonstrates the rarity of channel failure. Since the probability of two random failures in redundant channels in any 12-hour period is low, the CHANNEL CHECK minimizes the chance of loss of protective function due to failure of redundant channels. Thus, performance of the CHANNEL CHECK guarantees that undetected overt channel failure is limited to 12 hours. The CHANNEL CHECK supplements less formal, but more frequent checks of

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

channel OPERABILITY during normal operational use of the displays associated with the LCO required channels.

SR 3.3.7.1.2

A CHANNEL FUNCTIONAL TEST verifies 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, 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 must be found within the ALLOWABLE VALUES 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.

If, during the CHANNEL FUNCTIONAL TEST, the associated trip setting is discovered to be less conservative than the ALLOWABLE VALUE specified in Table 3.3.7.1-1, the channel must be declared inoperable. [For this facility, the basis for the Surveillance Frequency, once every 31 days, is as follows:].

A Note in the SR clarifies that this Surveillance does not requires the radiation detectors to be included in the CHANNEL FUNCTIONAL TEST because of the difficulty of injecting the radiation signal at the detectors. OPERABILITY of radiation detectors is demonstrated by the CHANNEL CALIBRATION and monitored by the CHANNEL CHECK.

SR 3.3.7.1.3

The basis for a CHANNEL FUNCTIONAL TEST was previously discussed in SR 3.3.7.1.2.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

If, during the CHANNEL FUNCTIONAL TEST, the associated trip setting is discovered to be less conservative than the ALLOWABLE VALUE specified in Table 3.3.7.1-1, the channel must be declared inoperable. The basis for the Surveillance Frequency, once every 92 days, is the reliability analysis of Reference 3.

[SR 3.3.7.1.4]

The calibration of bistable trip units consists of a test to determine the actual trip setpoints, and recalibration of the setpoint is necessary to ensure that it remains more conservative than the setpoint and the ALLOWABLE VALUE specified in Table 3.3.7.1-1. The channel must be declared inoperable if the setting is discovered to be less conservative than the ALLOWABLE VALUE. If the trip setting is discovered to be less conservative than accounted for in the appropriate setpoint methodology but is not beyond the ALLOWABLE VALUE, the channel is still considered OPERABLE. Under these conditions, the setpoint must be readjusted to be more conservative than accounted for in the appropriate setpoint methodology. The Surveillance Frequency is based on assumption of a 92-day calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

SR 3.3.7.1.5

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 setpoints 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.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

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 3.3.7.1.6

The LOGIC SYSTEM FUNCTIONAL TEST is performed every 18 months. Performance of a LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required actuation logic for a specific channel. The LOGIC SYSTEM FUNCTIONAL TEST tests all logic components (e.g., all relays and contracts, all trip units, solid-state logic elements) of a logic circuit, from sensors up to the actuated device. [For this facility, the system functional testing performed in LCO 3.7.5 overlaps this test to provide complete testing of the assumed safety function]. [For this facility, the Surveillance Frequency of 18 months is based on the following:]

REFERENCES

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

2. NEDC-30936-A, "BWR Owners' Group's Technical Specification Improvement Analyses for ECCS Actuation Instrumentation, Part 2," December 1988.

3. [Unit Name] "[Plant-Specific Setpoint Methodology]."

B 3.3 INSTRUMENTATION

B 3.3.8.1 Loss of Power (LOP) Instrumentation

BASES

BACKGROUND

Successful operation of the required safety functions of the Emergency Core Cooling System (ECCS) is dependent on the availability of adequate power sources for energizing the various components such as pump motors, motor-operated valves, and the associated control components. The LOP instrumentation monitors the 4.16 kV emergency buses. Offsite power is the preferred source of power for the 4.16 kV emergency buses. If the monitors determine that insufficient power is available, the buses are disconnected from the offsite power sources and connected to the onsite emergency diesel generator (EDG) power sources.

Each 4.16 kV emergency bus has its own independent LOP instrumentation. The voltage for each bus is monitored at two levels, which can be considered as two different undervoltage functions: loss of voltage and degraded voltage.

4.16 kV Emergency Bus Undervoltage (Loss of Voltage)

Loss of voltage on a 4.16 kV emergency bus indicates the inability of the power source to supply sufficient power for proper operation of the applicable equipment. Therefore, the power supply to the bus is transferred from offsite power to EDG power when the voltage on the bus drops below the loss of voltage function ALLOWABLE VALUES (loss of voltage with a short time delay). This assures that adequate power will be available to the required equipment.

4.16 kV Emergency Bus Undervoltage (Degraded Voltage)

A reduced voltage condition on a 4.16 kV emergency bus indicates that power may be insufficient for starting large ECCS motors without risking damage to the motors that could disable the ECCS function. Therefore, power supply to the bus is transferred from offsite power to onsite EDG power when the voltage on the bus drops below the degraded voltage trip function ALLOWABLE VALUES (degraded voltage with a time delay). This ensures that adequate power will be available to the required equipment.

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BASES (continued)

BACKGROUND (continued)

The LOP instrumentation is composed of two trip functions, which represent different voltage levels that cause various bus transfers and disconnects. Each function is monitored by two undervoltage relays whose output trip contacts are arranged in a two-out-of-two logic configuration.

The trip setpoints used in the voltage relays are based on the analytical limits presented in References 3. 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 Table 3.3.8.1-1 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 5. The actual nominal trip setpoint is normally still more conservative than that required by the plant-specific setpoint calculations. If the measured setpoint does not exceed the ALLOWABLE VALUE the voltage relay is considered OPERABLE.

Setpoints in accordance with the ALLOWABLE VALUE will assure that Safety Limits (SLs) are 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.

APPLICABLE SAFETY ANALYSES

The LOP instrumentation is required for engineered safety features (ESF) to function in any accident with a loss of offsite power. Its design basis is that of the ECCS. The required channels of LOP instrumentation, in conjunction with the safety systems powered from the EDGs, provide plant protection in the event of any of the Reference 1, 2, and 3 analyzed accidents in which a loss of offsite power is assumed.

Accident analyses credit the loading of the EDG based on the loss of offsite power during a loss-of-coolant accident (LOCA). 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

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

analysis assumes a non-mechanistic 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, low pressure coolant injection (LPCI), and high pressure coolant injection (HPCI) is [35] seconds. This delay time includes contributions from the EDG start, EDG loading, and ECCS component actuation. The response of the EDG to a loss of power must be demonstrated to fall within this response time when including the contributions of all portions of the delay.

The LOP instrumentation channels are required to meet the redundancy and testability requirements of GDC 21 in 10 CFR 50, Appendix A (Ref. 4).

The LOP instrumentation satisfies Criterion 3 of the NRC Interim Policy Statement.

LCO

The LCO requires OPERABILITY of sufficient LOP instrumentation to ensure 4.16 kV bus voltage protection. The OPERABILITY of the individual LOP instrumentation functions is confirmed through successful completion of required surveillance testing: CHANNEL CHECKS, CHANNEL FUNCTIONAL TESTS, LOGIC SYSTEM FUNCTION TESTS, and CHANNEL CALIBRATIONS. Individual measurement channels and the associated bistable trip units are considered OPERABLE when the following conditions are satisfied:

1. All channel components necessary to provide a start signal are functional and in service;

2. Channel measurement uncertainties are known via test, analysis, or design information to be within the assumptions of the setpoint calculations; and

3. Required surveillance testing is current and has demonstrated performance within each surveillance test's acceptance criteria.

Each function must have a minimum number of OPERABLE channels per 4.16 kV emergency bus, and each channel's setpoints must be within their ALLOWABLE VALUES. The actual

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BASES (continued)

LCO (continued)

setpoints are calibrated consistent with applicable setpoint methodology assumptions. Each channel must also respond within its specified delay time. [For this facility, the channels are configured in a one-out-of-two logic for actuation.]

Violation of this LCO could result in the delay of safety systems initiation when required. This could lead to the violation of the SLs during certain AOOs or to unacceptable consequences during accidents. During the loss of offsite power, the EDG powers the motor-driven LPCI and low pressure core spray (LPCS) pumps. Failure of these LPCI and LPCS pumps to start would leave turbine-driven reactor core isolation cooling and HPCI pumps to mitigate the effects of a LOCA under conditions with an increased potential for loss of decay heat removal capability.

The 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 calculations. 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.

The undervoltage bus ALLOWABLE VALUE is low enough to prevent inadvertent power supply transfer but high enough to ensure sufficient power is available to the required equipment. The time delay ALLOWABLE VALUE is long enough to provide time for the offsite power supply to recover to normal voltages but short enough to ensure sufficient power is available to the required equipment.

[For this facility, the relay configuration and trip function meets single failure criteria for single-phasing events for the following reasons:]

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BASES (continued)

LCO (continued)

[For this facility, the action of other relays (e.g., phase current differential) is required to ensure trip in the event of single phasing. The OPERABILITY of these other relays affects OPERABILITY of this function for the following reasons:]

[For this facility, the bases for ALLOWABLE VALUES is as follows:]

[For this facility, the bases for time-delay setpoints in the ALLOWABLE VALUES are as follows:]

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

[For this facility, the required support systems which, upon their failure, do not result in the LOP instrumentation being declared inoperable and their justification are as follows:]

[For this facility, the supported systems impacted by the inoperability of the LOP instrumentation and the justification of whether or not each supported system is declared inoperable are as follows:]

APPLICABILITY

The LOP instrumentation functions are required to be OPERABLE when the associated EDG is required to be OPERABLE. Thus, two channels per division of the loss of voltage function and two channels per division of the degraded voltage function are required to be OPERABLE in MODES 1, 2, and 3, because ECCS functions are required to provide protection in these modes. LOP instrumentation is required to be OPERABLE in MODE 4 and 5 when the associated EDG is required to be OPERABLE per LCO 3.8.2, so that the EDG can perform its function on a loss of power to the emergency bus.

ACTIONS

A protection function 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

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BASES (continued)

ACTIONS (continued)

section of the Bases. The most common cause of channel inoperability is outright failure or drift of the voltage relay setpoint 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 3.3.8.1-1, the channel must be declared inoperable immediately and the appropriate Conditions from Table 3.3.8.1-1 must 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 all affected functions provided by that channel must be declared inoperable and the LCO Condition entered for the particular protection function affected.

Condition A

Condition A applies to each one of the LOP instrumentation functions in Table 3.3.8.1-1. A Note is added to indicate that the Completion Time associated with Condition A is on a Condition basis for each function.

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

Required Action A.1 allows for the restoration of the inoperable LOP instrumentation channel(s) within 1 hour. If the channel cannot be restored to OPERABLE status in compliance with Required Action A.1, then Required Action A.2.1 requires that the channel be placed in trip. This places the system in a one-out-of-one logic for a limited amount of time. This Required Action is modified by a Note that Required Action A.2.1 is applicable only if placing the inoperable channel(s) in trip would not result in an initiation. Required Action A.2.1 is not intended to forces and unnecessary shutdown.

Required Action A.2.2 requires restoring the channel prior to the next CHANNEL FUNCTIONAL TEST which is performed on a

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BASES (continued)

ACTIONS (continued)

Surveillance Frequency of 31 days. The 31-day Completion Time for Required Action A.2.2 is sufficient to perform the Required Action. If after the required Completion Time has been exceeded, and the Required Action A.2.1 to trip the inoperable channel is completed, then the EDG may be declared OPERABLE, when determined to be so, and the remainder of Required Action A.2.2 is complied with.

The 1-hour Completion Time for Required Action A.1 and Required A.2.1 is sufficient for operations personnel to take corrective actions in an orderly manner and without challenging plant systems.

Condition B

Condition B applies to each one of the LOP instrumentation functions in Table 3.3.8.1-1.

B.1

If Condition A Required Actions cannot be completed within the required Completion Time, or if two required LOP instrumentation channels are inoperable for one or more functions in Table 3.3.8.1-1, then Required Action B.1 must be entered for each affected EDG. The affected EDGs and other associated supported systems are immediately declared inoperable and the corresponding LCOs are entered.

Prior to declaring the EDGs OPERABLE, it is not necessary to perform any Surveillance associated with LCO 3.8.1, "AC Sources—Operating," or LCO 3.8.2 "AC Sources—Shutdown," if the only reason for it was because of LOP instrumentation inoperability.

Condition C

Condition C is applicable to each one of the LOP instrumentation functions in Table 3.3.8.1-1.

C.1 and C.2

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

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BASES (continued)

ACTIONS (continued)

specified Completion Time is sufficient for plant operations personnel to make this verification.

Required Action C.1 ensures that those identified Required Actions associated with supported systems impacted by the inoperability of channel(s) associated with each LOP instrumentation 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 LOP instrumentation 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 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 EDG LOP instrumentation function are found in the SR column of Table 3.3.8.1-1. Most functions are subjected to CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION.

SR 3.3.8.1.1

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 indicated parameter output of the potential transformers that feed the LOP instrumentation relays. 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

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

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 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 the LCO required channels.]

SR 3.3.8.1.2

A CHANNEL FUNCTIONAL TEST is performed every 31 days to ensure that the entire channel will perform its intended function when needed.

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.

[For this facility, a CHANNEL FUNCTIONAL TEST constitutes the following:]

SR 3.3.8.1.3

SR 3.3.8.1.3 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

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

manufacturer. For OPERABLE channels, CHANNEL CALIBRATION shall find that measurement errors and 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. Repeated failures of the same channel over a relatively small number of test intervals, however, 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. For trips that take credit for undervoltage relay time voltage characteristics, CHANNEL CALIBRATION need only calibrate the time delay at one voltage. [For this facility, the time delay ALLOWABLE VALUE is specified for the following voltage:] 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 3.3.8.1.4

Performance of a LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required actuation logic for a specific channel. The LOGIC SYSTEM FUNCTIONAL TEST tests all logic components (i.e., all relays and contacts, all trip units, solid-state logic elements, etc.) of a logic circuit, from sensor up to the actuated device. For this facility, the system functional testing performed in LCO 3.8.1 and LCO 3.8.2 overlap this test to provide complete testing of the assumed safety functions. [For this facility, the Surveillance Frequency of 18 months is justified as follows:]

REFERENCES

1. [Unit Name] FSAR, Section [5.2], "[Title]."
2. [Unit Name] FSAR, Section [6.3], "[Title]."
3. [Unit Name] FSAR, Section [15], "[Title]."

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BASES (continued)

REFERENCES (continued)

4. Title 10, Code of Federal Regulations, Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants."

5. [Unit Name], "[Plant Protection System Selection of Trip Setpoint Values]."

B 3.3 INSTRUMENTATION

B 3.3.8.2 Reactor Protection System (RPS)-Electrical Power Monitoring (EPM)

BASES

BACKGROUND

RPS-EPM will detect any abnormal high or low voltage or low frequency condition in the outputs of the two motor-generator (MG) sets or the alternate power supply and will de-energize its respective RPS bus, thereby causing all safety functions normally powered by this bus to de-energize and, as a result, cause a half scram and half isolation.

RPS-EPM is provided to isolate the RPS bus power from the nonessential MG set or an alternate power source in the event of overvoltage, undervoltage, or underfrequency. This system protects the loads connected to the RPS bus against certain unacceptable voltage and frequency conditions as stated in Reference 1. It forms an important part of the primary success path of the essential safety circuits. Some of the essential equipment powered from the RPS buses includes the RPS logic, scram solenoids, and various isolation valve logic (e.g., residual heat removal (RHR) shutdown cooling valves logic).

In the event of failure of an RPS-EPM assembly, the RPS loads may experience unregulated power supply. Depending on the deviation from the nominal, the overvoltage, undervoltage, or underfrequency condition can cause potential damage to the scram solenoids and other Class 1E devices.

In the event of a low-voltage condition for a long period of time [e.g., 10 seconds], the scram solenoids can chatter and potentially lose the pneumatic control capability, resulting in loss of primary scram action. Under these conditions, in the event of a fault downstream of the circuit breakers, a low voltage will be present at the RPS bus, and the RPS assemblies EPM will open the circuit breaker and isolate the power source.

In the event of an overvoltage condition for a long period of time ([e.g., > 10 seconds]), the RPS logic relays and scram solenoids as well as the main steam isolation valve solenoids may experience a voltage higher than their design

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BASES (continued)

BACKGROUND (continued)

voltage. If this situation persists for a long time period ([e.g., > 10 seconds]) it may cause equipment degradation and the loss of plant safety function.

The output current of each MG set is controlled by two redundant Class 1E circuit breakers connected in series between the RPS bus and the MG set. Each of these circuit breakers is provided with an independent set of Class 1E overvoltage, undervoltage, and underfrequency sensing logic. If the output of the MG set exceeds predetermined limits of overvoltage, undervoltage, or underfrequency, a trip coil driven by this logic circuitry opens the circuit breaker.

A manual transfer scheme is also provided, which permits the energization of one RPS bus from an alternate source of power if its associated MG set is out of service. This allows both RPS buses to remain energized even though one of the MG sets may be out of service. The protective relaying scheme on the alternate source of power is identical to the two Class 1E protective schemes on the MG sets.

RPS-EPM supports the RPS design to conform with GDC 2, GDC 21, and GDC 23 (Ref. 2) and IEEE Standards 279 (Ref. 5) and 379 (Ref. 6). RPS-EPM provides protection to the RPS and other systems that receive power from the RPS buses, by acting to disconnect the RPS from the power-source circuits under specified conditions that could damage the RPS equipment.

APPLICABLE SAFETY ANALYSES

The RPS-EPM is necessary to meet the assumptions of the safety analyses and provide for the mitigation and in some cases termination of accident and transient conditions.

RPS-EPM satisfies Criterion 3 of the NRC Interim Policy Statement.

LCO

The LCO requires all instrumentation performing the RPS-EPM function to be OPERABLE. For this facility, there are [ ] channels of RPS electric power monitors per bus. Failure of

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BASES (continued)

LCO (continued)

any instrument renders the affected EPM assembly inoperable and reduces the reliability of the affected functions.

The RPS-EPM assemblies shall be OPERABLE to prevent the effects of specific abnormal voltage or frequency conditions on the system and components powered by the RPS power supplies from non-Class-1E sources. The two sets of Class 1E relays and circuit breakers in series between each MG set and its RPS bus, or an alternate source and its bus, provide a redundant protection function against the specified abnormal voltage or frequency conditions. Only the power monitoring system associated with the inservice power supply (MG set or alternate power supply) must be OPERABLE.

A channel is OPERABLE when the following conditions are satisfied:

1. All channel components necessary to provide a trip signal are functional and in service;

2. Channel measurement uncertainties are known via test, analysis, or design information to be within the assumptions of the setpoint calculations; and

3. Required surveillance testing is current and has demonstrated performance within each surveillance test's acceptance criteria.

[For this facility, the basis for overvoltage, undervoltage, and underfrequency ALLOWABLE VALUES are as follows:]

ALLOWABLE VALUES are specified in the LCO for the RPS-EPM. For this facility, nominal trip setpoints are specified in the plant-specific setpoint calculations provided in Reference 4. The nominal setpoints are selected to ensure that the setpoints measured by CHANNEL FUNCTIONAL TESTS do not exceed the ALLOWABLE VALUES. 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 transient and accident analysis in order to account for instrument uncertainties appropriate to the trip function.

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BASES (continued)

LCO (continued)

For this facility, these uncertainties are defined in the plant-specific setpoint methodology (Ref. 4).

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

[For this facility, those required support systems which, upon their failure, do not result in the RPS-EPM being declared inoperable and their justification are as follows:]

[For this facility, the supported systems impacted by the inoperability of the RPS-EPM and the justification of whether or not each supported system is declared inoperable are as follows:]

APPLICABILITY

The operation of the RPS-EPM assemblies is essential to disconnect the RPS-powered components from the MG set or alternate power source during the specified abnormal voltage or frequency conditions. Since the degradation of a non-Class-1E source supplying power to the RPS bus can occur as a result of any random single failure, the OPERABILITY of the RPS-EPM is required when the power source is in service, and the powered components are required to be OPERABLE. For this facility, this results in the RPS-EPM assemblies OPERABILITY requirements of MODES 1, 2, and 3, and MODE 4 or 5 with any control rod withdrawn from a core cell containing one or more fuel assemblies or RHR shutdown cooling isolation valves open.

For this LCO, a Note is added to indicate that Conditions A and B shall be treated as an entity with a single Completion Time.

ACTIONS

A protection function 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

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BASES (continued)

ACTIONS (continued)

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 SR 3.3.8.2.2, the channel must be declared inoperable immediately, and the appropriate Conditions from SR 3.3.8.2.2 must 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 bistable 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.

Condition A

A.1 and A.2

If one RPS-EPM assembly for an inservice power source (MG set or alternate) is inoperable, or one RPS-EPM assembly on each inservice power supply is inoperable, the OPERABLE assembly will still provide protection to the RPS equipment under degraded voltage or degraded frequency conditions. The reliability and redundancy of the RPS-EPM, however, is reduced. In this situation, 72 hours is allowed by Required Action A.1 to restore the inoperable assembly to OPERABLE status. If the inoperable assembly(s) cannot be made OPERABLE, the associated power supply must be taken out of service (Required Action A.2) within this 72-hour period. This places the RPS bus in a safe condition. An alternate power source with OPERABLE powering monitoring assemblies may then be used to power the RPS bus.

Required Action A.2 is modified by a Note to indicate that removal of the power source would only be applicable if it would not result in a scram or an isolation.

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BASES (continued)

ACTIONS (continued)

The 72-hour Completion Time takes into account the remaining OPERABLE power monitoring assembly and the low probability of an event requiring RPS-EPM Protection occurring during this period. It allows time for plant operations personnel to take corrective actions or to place the plant in the required condition in an orderly manner and without challenging plant systems.

Condition B

B.1 and B.2

If both power monitoring assemblies for an inservice power source (MG set or alternate) are inoperable or both power monitoring assemblies in each inservice power supply are inoperable, the system protective function is lost. In this situation, 1 hour is allowed by Required Action B.1 to restore at least one assembly to OPERABLE status for each inservice power source. Alternatively, if one inoperable assembly cannot be made OPERABLE, the associated power supplies must be taken out of service within 1 hour per Required Action B.2. An alternate power source with OPERABLE assemblies may then be used to power the one RPS bus. The 1-hour Completion Time is sufficient for the plant operations personnel to take corrective actions.

Required Action B.2 is modified by a Note to indicate that removal of the power source would only be applicable if it would result in a scram or an isolation.

Condition C

C.1 and C.2

If the Required Actions and associated Completion Times of Condition A or B are not met in MODE 1, 2, or 3, a controlled shutdown must be initiated. The controlled shutdown is accomplished by placing the plant in at least MODE 3 per Required Action C.1 within 12 hours and in MODE 4 per Required Action C.2 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required MODES from full power in an orderly manner and without challenging plant systems.

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BASES (continued)

ACTIONS (continued)

Condition D

D.1 and D.2

If the Required Actions and associated Completion Times of Condition A or B are not met in MODE 4 or MODE 5, or with any control rod withdrawn from a core cell containing one or more fuel assemblies or with RHR shutdown cooling valves open, the operator must take immediate corrective action. The Required Actions require operating personnel to initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies and to isolate the shutdown cooling line. These Required Actions result in the least reactive condition for the core and ensure that the safety function of the RPS (e.g., scram or insertion of control rods) and RHR shutdown cooling isolation valves are not required.

Condition E

E.1 and E.2

Condition E is applicable to each of the inservice RPS electrical power monitoring assemblies.

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 inservice RPS electric power monitoring assemblies within a Completion Time of 1 hour. The specified Completion Time is sufficient for plant operations personnel to make this verification.

Required Action E.2 ensures that those identified Required Actions associated with supported systems impacted by the inoperability of the inservice RPS electric power monitoring assemblies associated with 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 E is entered. [For this facility, the identified supported systems Required Actions associated with each inservice RPS electric power monitoring assembly is as follows:]

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BASES (continued)

ACTIONS (continued)

Required Action E verifies that all required support or supported features associated with the other redundant inservice RPS-EPM monitoring assembly are OPERABLE within

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 RPS EPM function are found in the SRs column of Table 3.3.8.2-1 for that function. Most functions are subject to CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION.

SR 3.3.8.2.1

A CHANNEL FUNCTIONAL TEST is performed to verify that the inservice RPS-EPM System actuates as required to protect the RPS equipment from a degraded power source. A CHANNEL FUNCTIONAL TEST verifies the function of the trip, interlock, and alarm functions of the channel. The test inserts simulated or actual signals as close to the sensor as practicable and verifies required trip, interlocks, and alarms function when the input is beyond the trip point. Where the design has made provisions for including sensors in the CFT, 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 VALUES 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.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

The Frequency allows the test to be performed while the plant is in a condition where the loss of the RPS bus (the power source must be removed from service to conduct the test) will not jeopardize steady-state power operation. The 24-hour condition is intended to indicate an outage of sufficient duration to allow for scheduling of the

Surveillance and proper performance of the test. A Note has been added that if the previous performance was within 184 days, the CHANNEL FUNCTIONAL TEST is not required in MODE 4 and is considered current. [For this facility, the basis for the 184-day Surveillance Frequency is justified as follows:]

SR 3.3.8.2.2

A CHANNEL CALIBRATION is performed to verify the trip setpoints for overvoltage, undervoltage, and underfrequency. The trip setpoints are based on [providing a nominal 120 volts AC, 60 Hz, at the RPS logic cabinets, which assures a minimum frequency of 57 Hz, and [108 to 132] VAC at the scram solenoid valves].

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. 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 setpoints 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 assumed by the setpoint analysis. However, repeated failures of the same channel over a relatively small number

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

of test intervals must be considered as potentially indicating a deterministic failure that cannot be corrected by recalibration.

[For this facility, the basis for the Surveillance Frequency is as follows:]

SR 3.3.8.2.3

Performance of a system functional test demonstrates the OPERABILITY of the required actuation logic for a specific channel. This tests all logic components (i.e., all relays and contacts, trip units, solid-state logic elements, etc.) of a logic circuit, from sensor up to the actuated device. The system functional test of the Class 1E circuit breakers is included as part of this test to provide complete testing of the safety function. If the breakers are incapable of operating, the associated channel would be inoperable. [For this facility, the basis for the Surveillance Frequency is as follows:]

REFERENCES

1. [Unit Name] FSAR, Section [ ], "[Title]."
2. Title 10, Code of Federal Regulations, Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants."
3. [Unit Name] FSAR, Section [ ], "[Title]."
4. [Unit Name], "[Plant Protection Selection of Trip Setpoint Values]."
5. Institute of Electrical and Electronic Engineers, IEEE-279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations."
6. Institute of Electrical and Electronic Engineers, IEEE-379, "[Title]."
7. [Unit Name] FSAR, Section [ ], "[Title]."

APPENDIX A

Acronyms

The following acronyms are used, but not defined, in the Standard Technical Specifications:

AC	alternating current
CFR	Code of Federal Regulations
DC	direct current
FSAR	Final Safety Analysis Report
LCO	Limiting Condition for Operation
SR	Surveillance Requirement
GDC	General Design Criteria or General Design Criterion

The following acronyms are used, with definitions, in the Standard Technical Specifications:

ACOT	ANALOG CHANNEL OPERATIONAL TEST
ADS	Automatic Depressurization System
ADV	atmospheric dump valve
AFD	axial flux difference
AFW	auxiliary feedwater
AIRP	air intake, recirculation, and purification
ALARA	as low as reasonably achievable
ANS	American Nuclear Society
ANSI	American National Standards Institute
AOO	anticipated operational occurrence
AOT	allowed outage time
APD	axial power distribution
APLHGR	AVERAGE PLANAR LINEAR HEAT GENERATION RATE
APRM	average power range monitor
APSR	axial power shaping rod
ARO	all rods out
ARC	auxiliary relay cabinets
ARS	Air Return System
ARTS	Anticipatory Reactor Trip System
ASGT	asymmetric steam generator transient
ASGTPTF	asymmetric steam generator transient protective trip function
ASI	axial shape index
ASME	American Society of Mechanical Engineers

(continued)

APPENDIX A (continued)

ASTM	American Society for Testing Materials
ATWS	anticipated transient without scram
ATWS-RPT	anticipated transient without scram recirculation pump trip
AVV	atmospheric vent valve
BAST	boric acid storage tank
BAT	boric acid tank
BDPS	Boron Dilution Protection System
BIST	boron injection surge tank
BIT	boron injection tank
BOC	beginning of cycle
BOP	balance of plant
BPWS	banked position withdrawal sequence
BWST	borated water storage tank
BTP	Branch Technical Position
CAD	containment atmosphere dilution
CAOC	constant axial offset control
CAS	Chemical Addition System
CCAS	containment cooling actuation signal
CCGC	containment combustible gas control
CCW	component cooling water
CEA	control element assembly
CEAC	control element assembly calculator
CEDM	control element drive mechanism
CFT	core flood tank
CIAS	containment isolation actuation signal
COLR	CORE OPERATING LIMITS REPORT
COLSS	Core Operating Limits Supervisory System
CPC	core protection calculator
CPR	critical power ratio
CRA	control rod assembly
CRD	control rod drive
CRDA	control rod drop accident
CRDM	control rod drive mechanism
CREHVAC	Control Room Emergency Air Temperature Control System
CREFS	Control Room Emergency Filtration System
CREVS	Control Room Emergency Ventilation System
CRFAS	Control Room Fresh Air System
CS	core spray
CSAS	containment spray actuation signal

(continued)

APPENDIX A (continued)

CST	condensate storage tank
CVCS	Chemical and Volume Control System
DBA	Design Basis Accident
DBE	Design Basis Event
DF	decontamination factor
DG	diesel generator
DIV	drywell isolation valve
DNB	departure from nucleate boiling
DNBR	departure from nucleate boiling ratio
DOP	dioctyl phthalate
DPIV	drywell purge isolation valve
DRPI	digital rod position indicator
EAB	exclusion area boundary
ECCS	Emergency Core Cooling System
ECW	essential chilled water
ECP	estimated critical position
EDG	emergency diesel generator
EFAS	Emergency Feedwater Actuation System
EFIC	emergency feedwater initiation and control
EFCV	excess flow check valve
EFPDs	effective full power days
EFPYs	effective full power years
EFW	emergency feedwater
EHC	electro-hydraulic control
EOC	end of cycle
EOC-RPT	end of cycle recirculation pump trip
ESF	engineered safety feature
ESFAS	Engineered Safety Feature Actuation System
ESW	essential service water
EVS	Emergency Ventilation System
FBACS	Fuel Building Air Cleanup System
FCV	flow control valve
FHAVS	Fuel Handling Area Ventilation System
FSPVS	Fuel Storage Pool Ventilation System
FRC	fractional relief capacity
FR	Federal Register
FTC	fuel temperature coefficient
FWLB	feedwater line break

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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	Iodine Cleanup System
IEEE	Institute of Electrical and Electronic Engineers
IGSCC	intergranular stress corrosion cracking
IRM	intermediate range monitor
ISLH	inservice leak and hydrostatic
ITC	isothermal temperature coefficient
K-relay	control relay
LCS	Leakage Control System
LEFM	linear elastic fracture mechanics
LER	Licensee Event Report
LHGR	LINEAR HEAT GENERATION RATE
LHR	linear heat rate
LLS	low-low set
LOCA	loss-of-coolant accident
LOCV	loss of condenser vacuum
LOMFW	loss of main feedwater
LOP	loss of power
LOPS	loss of power start
LOVS	loss of voltage start
LPCI	low pressure coolant injection
LPCS	low pressure core spray
LPD	local power density
LPI	low pressure injection
LPRM	local power range monitor
LPSI	low pressure safety injection
LPSP	low power setpoint

(continued)

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

(continued)

PCIV	primary containment isolation valve
PCHRS	Primary Containment Hydrogen Recombiner System
PCP	Process Control Program
PCPV	primary containment purge valve
PCT	peak cladding temperature
PDIL	power dependent insertion limit
PDL	power distribution limit
PF	position factor
PIP	position indication probe
PIV	pressure isolation valve
PORV	power-operated relief valve
PPS	Plant Protective System
PRA	probabilistic risk assessment
PREACS	Pump Room Exhaust Air Cleanup System; Penetration Room Exhaust Air Cleanup System
PSW	plant service water
P/T	pressure and temperature
PTE	PHYSICS TEST exception
PTLR	PRESSURE AND TEMPERATURE LIMITS REPORT
QA	quality assurance
QPT	quadrant [illegible]we[illegible] tilt
QPTR	quadrant [illegible]ver [illegible]il% ratio
QS	quench spr[illegible]
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)

APPENDIX A (continued)

RPIS	Rod Position Information System
RPS	Reactor Protection System
RPT	recirculation pump trip
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

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S/RV	safety/relief valve
S/RVDL	safety/relief valve discharge line
SSPS	Solid State Protection System
SSW	standby service water
SWS	Service Water System
STE	special test exception
STS	Standard Technical Specifications
TADOT	trip actuating device operational test
TCV	turbine control valve
TIP	transversing incore probe
TLD	thermoluminescent dosimeter
TM/LP	thermal margin/low pressure
TS	Technical Specifications
TSV	turbine stop valve
UHS	Ultimate Heat Sink
VCT	volume control tank
VFTP	Ventilation Filter Testing Program
VHPT	variable high power trip
v/o	volume percent
VS	vendor specific
ZPMB	zero power mode bypass

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10. SUPPLEMENTARY NOTES

11. ABSTRACT (200 words or less)

This draft report documents the results of the NRC staff review of new Standard Technical Specifications (STS) proposed by the BWR Owners Group for the BWR/4 design. 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 - 3.10 of the new STS.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)

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