NUREG–1431
Vol. 2

Standard Technical Specifications
Westinghouse Plants

Bases (Sections 2 0–3.3)

Draft Report for Comment

Issued by the
U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

January 1991

NUREG-1431
Vol. 2

Standard Technical Specifications
Westinghouse Plants

Bases (Sections 2.0–3.3)

Draft Report for Comment

Issued by the
U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

January 1991

AVAILABILITY NOTICE

Availability of Reference Materials Cited in NRC Publications

Most documents cited in NRC publications will be available from one of the following sources:

1. The NRC Public Document Room, 2120 L Street, NW, Lower Level, Washington, DC 20555
2. The Superintendent of Documents, U.S. Government Printing Office, P.O. Box 37082, Washington, DC 20013-7082
3. The National Technical Information Service, Springfield, VA 22161

Although the listing that follows represents the majority of documents cited in NRC publications, it is not intended to be exhaustive.

Referenced documents available for inspection and copying for a fee from the NRC Public Document Room include NRC correspondence and internal NRC memoranda; NRC Office of Inspection and Enforcement bulletins, circulars, information notices, inspection and investigation notices; Licensee Event Reports; vendor reports and correspondence; Commission papers; and applicant and licensee documents and correspondence.

The following documents in the NUREG series are available for purchase from the GPO Sales Program: formal NRC staff and contractor reports, NRC-sponsored conference proceedings, and NRC booklets and brochures. Also available are Regulatory Guides, NRC regulations in the Code of Federal Regulations, and Nuclear Regulatory Commission Issuances.

Documents available from the National Technical Information Service include NUREG series reports and technical reports prepared by other federal agencies and reports prepared by the Atomic Energy Commission, forerunner agency to the Nuclear Regulatory Commission.

Documents available from public and special technical libraries include all open literature items, such as books, journal and periodical articles, and transactions. Federal Register notices, federal and state legislation, and congressional reports can usually be obtained from these libraries.

Documents such as theses, dissertations, foreign reports and translations, and non-NRC conference proceedings are available for purchase from the organization sponsoring the publication cited.

Single copies of NRC draft reports are available free, to the extent of supply, upon written request to the Office of Information Resources Management, Distribution Section, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

Copies of industry codes and standards used in a substantive manner in the NRC regulatory process are maintained at the NRC Library, 7920 Norfolk Avenue, Bethesda, Maryland, and are available there for reference use by the public. Codes and standards are usually copyrighted and may be purchased from the originating organization or, if they are American National Standards, from the American National Standards Institute, 1430 Broadway, New York, NY 10018.

PREFACE

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

The new STS will be used as bases for developing improved plant-specific technical specifications by individual nuclear power plant owners that have PWRs designed by Westinghouse. 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-1431, "Standard Technical Specifications, Westinghouse Plants." Each proposed change should be numbered. Each proposed change should be accompanied with a separate technical justification, cross referenced to the applicable proposed change on the marked up pages.

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

(continued)

B 2.0 SAFETY LIMITS

B 2.1.1 Reactor Core Safety Limits (SLs)

BASES

BACKGROUND

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

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 is prevented by maintaining the steady-state peak linear heat rate (LHR) below the level at which centerline fuel melting occurs. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime, where the heat-transfer coefficient is large and the cladding-surface temperature is slightly above the coolant-saturation temperature.

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

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

(continued)

(continued)

BASES (continued)

BACKGROUND (continued)

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

APPLICABLE SAFETY ANALYSES

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

a. There must be at least 95% probability at a 95% confidence level that the hot fuel rod in the core does not experience DNB (this is referred to hereafter as the 95/95 DNB criterion); and

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

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

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

a. High pressurizer pressure reactor trip;

b. Low pressurizer pressure reactor trip;

c. Overtemperature ΔT reactor trip;

d. Overpower ΔT reactor trip;

e. Power range high neutron flux reactor trip; and

f. SG safety valves.

The limitation that the average enthalpy in the hot leg be less than or equal to the enthalpy of saturated liquid also ensures that the ΔT measured by instrumentation (used in the

(continued)

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

RPS design as a measure of core power) is proportional to core power.

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

SAFETY LIMITS

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

The curves are based on enthalpy hot channel factor limits provided in the CORE OPERATING LIMITS REPORT (COLR). The dashed line of Figure B 2.1.1-1 shows an example of a limit curve at 2235 psig. In addition, it illustrates the various RPS functions that are designed to prevent the unit from reaching the limit.

The SL is higher than the limit calculated when the AXIAL FLUX DIFFERENCE (AFD) is within the limits of the $F_1(\Delta I)$ function of the overtemperature ΔT reactor trip. When the AFD is not within the tolerance, the AFD effect on the overtemperature ΔT reactor trips will reduce the setpoints to provide protection consistent with the reactor core SLs (Refs. 3 and 4).

APPLICABILITY

SL 2.1.1 only applies in MODES 1 and 2 because these are the only modes in which the reactor is critical. Automatic protection functions are required to be OPERABLE during MODES 1 and 2 to ensure operation within the reactor core SLs. The SG safety valves or automatic protection actions serve to prevent RCS heatup to the reactor core SL conditions or to initiate a reactor trip function (which

(continued)

(continued)

BASES (continued)

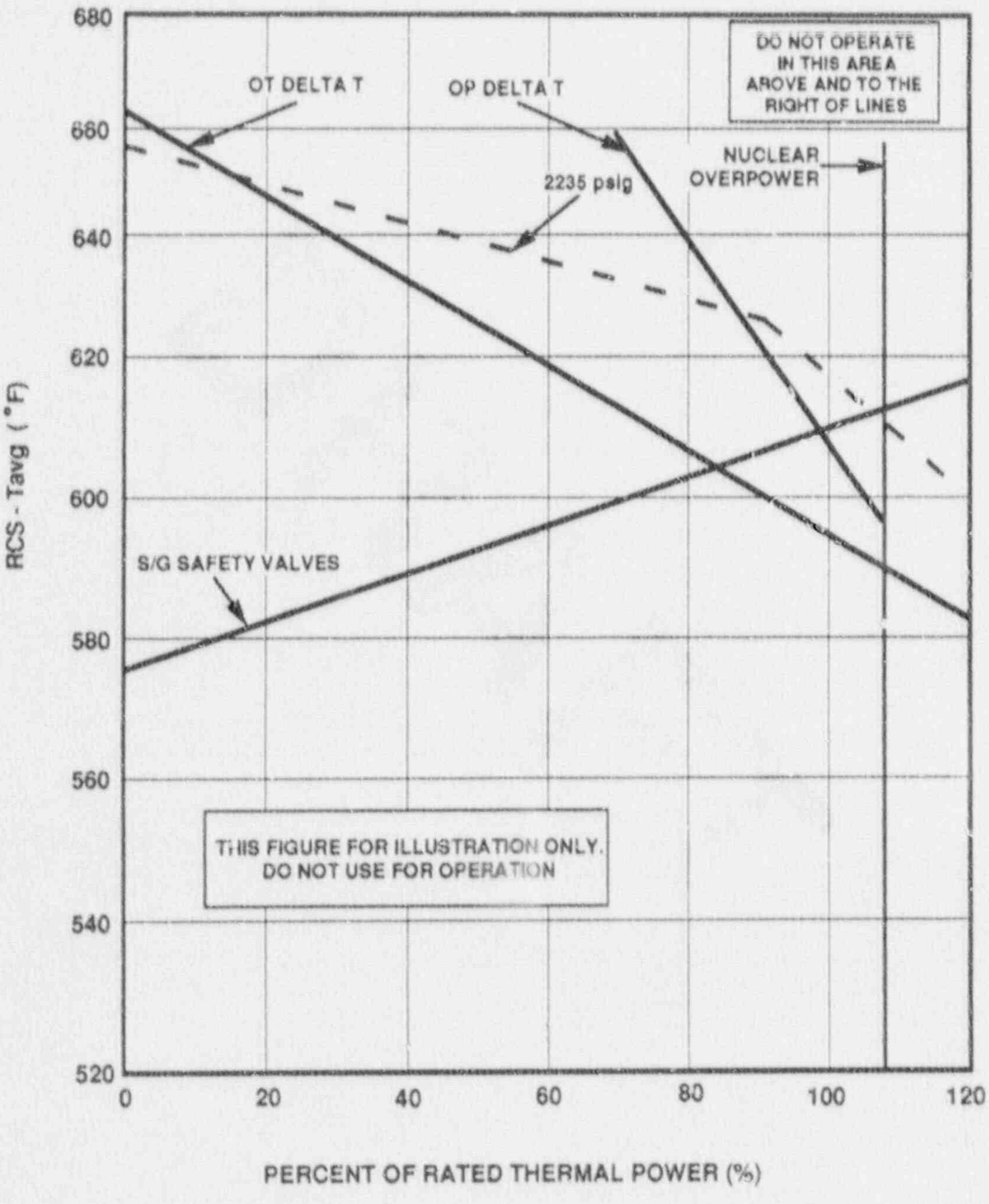


Figure B 2.1.1-1
Reactor Core Safety Limits vs Boundary of Protection

(continued)

(continued)

BASES (continued)

APPLICABILITY (continued)

forces the unit into MODE 3). Setpoints for the reactor trip functions are specified in LCO 3.3.1 and LCO 3.3.2. In MODES 3, 4, 5, and 6, Applicability is not required since the reactor is not generating significant THERMAL POWER.

SAFETY LIMIT VIOLATIONS

2.2.1

If SL 2.1.1 is violated, the requirement to go to MODE 3 places the plant in a MODE in which this SL is not applicable.

The allowed Completion Time of 1 hour recognizes the importance of bringing the plant to a mode of operation where this SL is not applicable. Also, the Completion Time of 1 hour ensures that the probability of an accident occurring during this period is minimal.

2.2.3

If SL 2.1.1 is violated, the NRC Operations Center must be notified within 1 hour. This is in accordance with 10 CFR 50.72 (Ref. 5).

2.2.4

If SL 2.1.1 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 senior management.

2.2.5

If SLs 2.1.1 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).

(continued)

(continued)

BASES (continued)

SAFETY LIMIT VIOLATIONS (continued)

2.2.6

If SL 2.1.1 is violated, restart of the unit shall not commence until authorized by the NRC. This requirement ensures the NRC that all necessary reviews, analyses, and actions are completed before the unit begins its restart to normal operation.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix A, General Design Criterion 10, "Reactor Design," 1988.

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

3. WCAP-8746-A, "Design Bases for the Overtemperature ΔT and the Overpower ΔT Trips," March 1977.

4. WCAP-9273-NP-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985.

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 Coolant System (RCS) Pressure Safety Limit (SL)

BASES

BACKGROUND

The SL on RCS pressure protects the integrity of the RCS against overpressurization. In the event of fuel-cladding failure, fission products are released into the reactor coolant. The RCS then serves as the primary barrier in preventing the release of fission products into the atmosphere. By establishing an upper limit on RCS pressure, the continued integrity of the RCS is ensured. According to 10 CFR 50, Appendix A, GDC 14, "Reactor Coolant Pressure Boundary," and GDC 15, "Reactor Coolant System Design" (Ref. 1), the reactor pressure coolant 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 2500 psia. During normal operation and AOOs, RCS pressure is limited from exceeding the design pressure by more than 10%, in accordance with Section III of the American Society of Mechanical Engineers (ASME) Code (Ref. 2). To ensure system integrity, all RCS components are hydrostatically tested at 125% of design, per the ASME Code requirements prior to initial operation when there is no fuel in the core. If repairs or replacements take place that would require a full hydrostatic test of the RCS, the fuel would have to be completely offloaded before exceeding the maximum pressure specified in this SL. Removing fuel from the vessel precludes fission products from entering the reactor coolant.

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

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

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

The RCS pressurizer safety valves are sized to prevent system pressure from exceeding the design pressure by more than 10%, as specified in Section III of the ASME Code for Nuclear Power Plant Components (Ref. 2). The transient that establishes the required relief capacity, and hence valve size requirements and lift settings, is a complete loss of external load without a direct reactor trip. During the transient, no control actions are assumed except that the safety valves on the secondary plant are assumed to open when the steam pressure reaches the secondary plant safety valve settings and nominal feedwater supply is maintained.

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

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

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

b. Steam line relief valve;

c. Steam Dump System;

d. RCS;

e. Pressurizer Level Control System; or

f. Pressurizer spray valve.

SAFETY LIMITS

The maximum transient pressure allowed in the RCS pressure vessel under the ASME Code, Section III, is 110% of design

(continued)

(continued)

BASES (continued)

SAFETY LIMITS

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

APPLICABILITY

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

SAFETY LIMIT VIOLATIONS

2.2.2.1

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

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

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

2.2.2.2

If the RCS pressure SL is exceeded in MODE 3, 4, or 5, RCS pressure must be restored to within the SL value within 5 minutes. Exceeding the RCS pressure SL in MODE 3, 4, or 5 is more severe than exceeding this SL in MODE 1 or 2, since the reactor vessel temperature may be lower and the vessel

(continued)

(continued)

BASES (continued)

SAFETY LIMIT VIOLATIONS (continued)

material, consequently, less ductile. As such, pressure must be reduced to less than the SL within 5 minutes. The action does not require reducing MODES, since this would require reducing temperature, which would compound the problem by adding thermal gradient stresses to the existing pressure stress.

2.2.3

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

2.2.4

If the RCS pressure 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 the plant operators and staff to take the appropriate immediate action and assess the condition of the plant before reporting to senior management.

2.2.5

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

2.2.6

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

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."

(continued)

(continued)

BASES (continued)

REFERENCES (continued)

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

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

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

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

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

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

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

B 3.0 APPLICABILITY

B 3.0 Limiting Conditions for Operation (LCO) Applicability

BASES

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

(continued)

(continued)

BASES (continued)

LCO 3.0.2 (continued)

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

This specification establishes that performance of the Required Actions within the specified Completion Times constitutes compliance with the TS. It also establishes, however, that completing the performance of the Required Actions is not required when an LCO is met within the associated Completion Time, unless otherwise 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

(continued)

(continued)

BASES (continued)

LCO 3.0.2 (continued)

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

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

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

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

(continued)

(continued)

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.

(continued)

(continued)

BASES (continued)

LCO 3.0.3 (continued)

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

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

a. The LCO is now met;

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

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

(continued)

(continued)

BASES (continued)

LCO 3.0.3 (continued)

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

The time limits of Specification 3.0.3 allow 37 hours for the facility to be in MODE 5 when a shutdown is required during MODE 1 operation. If the facility is in a higher-numbered MODE of operation when a shutdown is required, the time limit for reaching the next higher-numbered MODE applies. If a higher-numbered MODE is reached in less time than allowed, however, the total allowable time to reach MODE 5, or other applicable MODE, is not reduced. For example, if MODE 3 is reached in 2 hours, then the time allowed to reach MODE 4 is the next 11 hours, because the total time to reach MODE 4 is not reduced from the allowable limit of 13 hours. Therefore, if remedial measures are completed that would permit a return to MODE 1, a penalty is not incurred by having to reach a higher-numbered MODE of operation in less than the total time allowed.

In MODES 1, 2, 3, and 4, LCO 3.0.3 provides Required Actions for Conditions not stated in other specifications. The requirements of LCO 3.0.3 do not apply in MODES 5 and 6 because the facility is already in the most restrictive Condition in which that LCO 3.0.3 would require the facility to be placed. The requirements of LCO 3.0.3 do not apply in other specified conditions of the Applicability (unless in MODE 1, 2, 3, or 4) because the ACTIONS of individual specifications sufficiently define the remedial measures to be taken. [This must be verified by review of all LCOs when finalized.]

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

(continued)

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.

(continued)

BASES (continued)

LCO 3.0.5

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

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

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

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

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

(continued)

(continued)

BASES (continued)

LCO 3.0.5 (continued)

those Surveillances need not be performed unless specified by the ACTIONS or SRs of the STE LCO.

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

B 3.0 APPLICABILITY

B 3.0 Surveillance Requirement (SR) Applicability

BASES

SR 3.0.1, SR 3.0.2, SR 3.0.3, and SR 3.0.4	SR 3.0.1, 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 3.9.

SR 3.0.1

SR 3.0.1 establishes the requirement that SRs must be met during the MODES or other specified Conditions in the Applicability of the LCO, unless otherwise specified in the individual SRs. This specification ensures that Surveillances are performed to verify the OPERABILITY of systems and components, and that variables are within specified limits. Failure to meet an SR within the specified Frequency, in accordance with SR 3.0.2, constitutes a failure to meet an LCO.

Systems and components are assumed to be OPERABLE when the associated SRs have been met. Nothing in this specification, however, is to be construed as implying that systems or components are OPERABLE when:

a. The systems or components are known to be inoperable, although SRs are being met; or

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

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

(continued)

(continued)

BASES (continued)

SR 3.0.1 (continued)

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

Upon completion of maintenance, appropriate post-maintenance testing (which usually includes Surveillance testing) is required to declare equipment OPERABLE. Post-maintenance testing may not be possible in the MODE or Condition that the facility is in when the maintenance is completed because the necessary facility parameters have not been established. In these situations, proceeding to the appropriate 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).

(continued)

(continued)

BASES (continued)

SR 3.0.2 (continued)

The 25% extension does not significantly degrade the assurance of reliability obtained 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 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, becaus[illegible] NRC staff concluded that the removal would result in [illegible]ter benefit to safety. This line-item improvement t[illegible] STS did not extend the Applicability of the 25% exten[illegible]on 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

(continued)

(continued)

BASES (continued)

SR 3.0.2 (continued)

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.) 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.

(continued)

(continued)

BASES (continued)

SR 3.0.3 (continued)

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 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.

(continued)

(continued)

BASES (continued)

SR 3.0.3 (continued)

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.

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.

(continued)

(continued)

BASES (continued)

SR 3.0.4 (continued)

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 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.

(continued)

(continued)

BASES (continued)

SR 3.0.4 (continued)

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 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

(continued)

(continued)

BASES (continued)

SR 3.0.4 (continued)

(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.

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

(continued)

(continued)

BASES (continued)

SR 3.0.4 (continued)

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, 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 TS — 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 TS (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)—T_{avg} > 200°F

BASES

BACKGROUND

Per GDC 26 (Ref. 1), the Reactivity Control System must be redundant and capable of holding the reactor core subcritical when shutdown under cold conditions. 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 that would be obtained immediately following the insertion or scram of all shutdown and control rods, assuming that the single rod cluster assembly of highest reactivity worth is fully withdrawn. In MODES 3, 4, and 5, the specified SDM continues to provide for adequate shutdown capability and acceptable fuel design limits for potential accidents initiated from shutdown conditions.

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

During power operation, SDM control is ensured by operating with the shutdown banks fully withdrawn and the control banks within the limits of LCO 3.1.7, "Control Bank Insertion Limits." When in the shutdown and refueling MODES, the SDM requirements are met by adjustments to the RCS boron concentration.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The minimum required SDM is assumed as an initial condition in safety analyses. The safety analyses (Ref. 2) establish an 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.

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

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

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

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

The most limiting accident for the SDM requirements is based on a main steam line break (MSLB), as described in the accident analysis (Ref. 2). The increased steam flow resulting from a pipe break in the main steam system causes an increased energy removal from the affected steam generator (SG), and consequently the RCS. This results in a reduction of the reactor coolant temperature. The resultant coolant shrinkage causes a reduction in pressure. In the presence of a negative moderator temperature coefficient, this cooldown causes an increase in core reactivity. As RCS temperature decreases, the severity of an MSLB decreases until the MODE 5 value is reached. The most limiting MSLB, with respect to potential fuel damage before a reactor trip occurs, is a guillotine break of a main steam line inside containment initiated at the end of core life. The positive reactivity addition from the moderator temperature decrease will terminate when the affected SG boils dry, thus terminating RCS heat removal and cooldown. Following the MSLB, a post-trip return to power may occur; however, no fuel damage occurs as a result of the post-trip return to

(continued)

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

power, and the THERMAL POWER does not violate the Safety Limit requirement of SL 2.1.1.

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

a. Inadvertent boron dilution;

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

c. Startup of an inactive Reactor Coolant Pump (RCP); and

d. Rod ejection.

Each of these events is discussed below.

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

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

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

The withdrawal of CONTROL RODS from subcritical or low power conditions adds reactivity to the reactor core, causing both the core power level and heat flux to increase with

(continued)

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

corresponding increases in reactor coolant temperatures and pressure. The withdrawal of rods also produces a time-dependent redistribution of core power.

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

LCO

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

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

The MSLB (Ref. 2) and the boron dilution (Ref. 3) accidents are the most limiting analyses that establish the SDM value of the LCO. For MSLB accidents, if the LCO is violated, there is a potential to exceed the DNBR limit and to exceed 10 CFR 100, "Reactor Site Criteria," limits. For the boron dilution accident, if the LCO is violated, the minimum required time assumed for operator action to terminate dilution may no longer be applicable.

APPLICABILITY

In MODES 1, 2, 3, and 4, the SDM requirements are applicable to provide sufficient negative reactivity to meet the assumptions of the safety analyses discussed above. In MODE 5, SDM is addressed by LCO 3.1.2. In MODE 6, the shutdown reactivity requirements are given in LCO 3.9.1, "Boron Concentration."

(continued)

BASES (continued)

ACTIONS

A.1

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

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

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

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

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

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

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

e. Decay heat flow through decay heat pumps from the BWST: Decay heat pumps are rated at [3000] gpm at [100] psig.

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

(continued)

(continued)

BASES (continued)

ACTIONS (continued)

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

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

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1

In MODES 1 and 2, SDM is verified by observing that the requirements of LCO 3.1.6, "Shutdown Bank Insertion Limit," and LCO 3.1.7, "Control Bank Insertion Limit," are met. In the event that a rod is known to be untrippable, however, SDM verification must account for the worth of the untrippable rod as well as another rod of maximum worth.

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

a. RCS boron concentration;

b. Control bank position;

c. RCS average temperature;

d. Fuel burnup based on gross thermal energy generation;

e. Xenon concentration;

f. Samarium concentration; and

g. Isothermal Temperature Coefficient (ITC).

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

The Frequency of 24 hours is based on the low probability of an accident occurring between verifications and the generally slow change in required boron concentration. This

(continued)

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

allows time for the operator to collect the required data, including a boron concentration analysis, and complete the calculation.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix A, General Design Criterion 26, "Reactivity Control System Redundancy and Capability."
2. [Unit Name] FSAR, Section [ ], "[Title]."
3. [Unit Name] FSAR, Section [ ], "[Title]."

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.2 SHUTDOWN MARGIN (SDM)—$T_{avg} \leq 200°F$

BASES

BACKGROUND

Per 10 CFR 50, GDC 26, Ref. 1, the Reactivity Control System must be redundant and capable of holding the reactor core subcritical when shutdown under cold conditions. 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 shutdown and control rods, assuming the single rod cluster assembly of highest reactivity worth is fully withdrawn. In MODES 3, 4, and 5, the SDM specified continues to provide for adequate shutdown capability and acceptable fuel design limits for potential accidents initiated from shutdown conditions.

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

During power operation, SDM control is ensured by operating with the shutdown banks fully withdrawn and the control banks within the limits of LCO 3.1.7, "Control Bank Insertion Limits." When in the shutdown and refueling

(continued)

(continued)

BASES (continued)

BACKGROUND (continued)

MODES, the SDM requirements are met by adjustments to the RCS boron concentration.

APPLICABLE SAFETY ANALYSES

The minimum required SDM is assumed as an initial condition in the safety analysis. The safety analysis (Ref. 2) establishes 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, for MODE 5, the primary safety analysis that relies on the SDM limits is the boron dilution analysis.

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

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

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

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

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

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

(continued)

BASES (continued)

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.

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

The boron dilution accident (Ref. 2) is the most limiting analysis that establishes the SDM value of the LCO. For the boron dilution accident, if the LCO is violated, then the minimum required time assumed for operator action to terminate dilution may no longer be applicable.

APPLICABILITY

In MODE 5, the SDM requirements are applicable to provide sufficient negative reactivity to meet the assumptions of the safety analyses discussed above. In MODES 1, 2, 3, and 4, the SDM requirements are given in LCO 3.1.1, "SHUTDOWN MARGIN (SDM)." In MODE 6, the shutdown reactivity requirements are given in LCO 3.9.1, "Boron Concentration."

ACTIONS

A.1

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

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

Therefore, the operator should begin boration with the best source available for the plant conditions. Some of the

(continued)

(continued)

BASES (continued)

ACTIONS (continued)

possible sources of boron originate from either the boric acid storage tank (BAST), whose minimum concentration of boron is [11600] ppm or the borated water storage tank (BWST), whose minimum concentration of boron is [2270] ppm. These sources include:

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

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

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

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

e. Decay heat flow through decay heat pumps from BWST: Decay heat pumps are rated at [3000] gpm at [100] psig.

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

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

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

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1

In MODE 5, the SDM is verified by performing a reactivity balance calculation, considering the following reactivity effects:

(continued)

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

a. RCS boron concentration;

b. Control bank position;

c. RCS average temperature;

d. Fuel burnup based on gross thermal energy generation;

e. Xenon concentration;

f. Samarium concentration; and

g. Isothermal Temperature Coefficient (ITC).

Using the ITC accounts for Doppler reactivity in this calculation because the reactor is subcritical and the fuel temperature will be changing at the same rate as the RCS. The Frequency of 24 hours is based on the low probability of an accident occurring between verifications and the generally slow change in required boron concentration. This allows time enough for the operator to collect the required data, including a boron concentration analysis, and complete the calculation.

REFERENCES

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

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

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.3 Core Reactivity

BASES

BACKGROUND

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

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

(continued)

(continued)

BASES (continued)

BACKGROUND (continued)

calculation models used to generate the safety analysis are adequate.

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

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

APPLICABLE SAFETY ANALYSES

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

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

(continued)

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

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

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

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

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

LCO

This Specification is provided to ensure that core reactivity behaves as expected in the long term, and to ensure that significant reactivity anomalies will be investigated. Long-term core reactivity behavior is a result of the core physics design and cannot be easily controlled once the core design is fixed. During operation,

(continued)

(continued)

BASES (continued)

LCO (continued)

therefore, the Conditions of the LCO can only be ensured through measurement and tracking, and appropriate actions taken as necessary. Large differences between actual and predicted core reactivity may indicate that the assumptions of the design basis transient and accident analyses are no longer valid, or that the uncertainties in the Nuclear Method are larger than expected. A limit on the reactivity balance of ± 1% Δk/k has been established based on engineering judgment. A 1% deviation in reactivity from that predicted is larger than expected for normal operation and should therefore be evaluated.

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

APPLICABILITY

In MODE 1, most of the control rods are withdrawn and steady-state operation is typically achieved. Under these conditions, the comparison between predictions and measurements provides an effective measure of the reactivity balance. In MODE 2, control rods are typically being withdrawn during a startup. In MODES 3, 4, and 5, all control rods are fully inserted, and therefore the reactor is in the least reactive state where monitoring core reactivity is not necessary. In MODE 6, fuel loading results in a continually changing core reactivity. Boron concentration requirements (LCO 3.9.1) ensure that fuel movements are performed within the bounds of the safety analysis, and an SDM demonstration is required during the first startup following operations that could have altered core reactivity (e.g., fuel movement, control rod replacement, control rod shuffling).

(continued)

BASES (continued)

ACTIONS

A.1

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

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

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

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

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

d. OPERABILITY of all control rods is verified; and

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

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

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

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

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

d. The calculation models are reviewed to verify that they can adequately represent the core conditions.

(continued)

(continued)

BASES (continued)

ACTIONS (continued)

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

A.2

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

The required Completion Time of 72 hours is adequate to prepare whatever operating restrictions or Surveillances that may be required to allow continued reactor operation.

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% $\Delta k/k$ limit by the methods discussed in Required Actions A.1 and A.2 and their associated Completion Times. This is done by placing the unit in at least MODE 3 within

(continued)

(continued)

BASES (continued)

ACTIONS (continued)

6 hours. If the SHUTDOWN MARGIN for MODE 3 is not met, then the boration required by SR 3.1.1.1 would occur. The allowed Completion Time is reasonable, based on operating experience, related to the time required to reach the required plant conditions from full power in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.3.1

Core reactivity is verified by periodic comparisons of measured and predicted RCS boron concentrations. The comparison is made considering that other core conditions are fixed or stable, including control rod position, moderator temperature, fuel temperature, fuel depletion, xenon concentration, and samarium concentration. The Surveillance is performed prior to entering MODE 1 as an initial check on core conditions and design calculations at BOC. A Note is included in the SR to indicate that the normalization of predicted core reactivity to the measured value must take place within the first 60 effective full power days (EFPDs) after each fuel loading. This allows sufficient time for core conditions to reach steady state, but prevents operation for a large fraction of the fuel cycle without establishing a benchmark for the design calculations. The required subsequent Frequency of 31 EFPDs after the initial 60 EFPDs after entering MODE 1 is acceptable based on the slow rate of core changes due to fuel depletion and the presence of other indicators (QUADRANT POWER TILT RATIO, AXIAL FLUX DIFFERENCE, etc.) for prompt indication of an anomaly. Another note is included in SR to indicate that the provisions of SR 3.0.4 are not applicable for this SR for entering MODE 2.

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"; and General Design Criterion 29, "Protection Against Anticipated Operational Occurrences."

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

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.4 Moderator Temperature Coefficient (MTC)

BASES

BACKGROUND

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

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

MTC values are predicted at selected burnups during the safety evaluation analysis and are confirmed to be acceptable by measurements. Both initial and reload cores are designed so that the beginning of cycle (BOC) MTC is less than zero when THERMAL POWER is [95%] of RATED THERMAL POWER (RTP) or greater. The actual value of the MTC is dependent on core characteristics such as fuel loading and reactor coolant soluble boron concentration. The core design may require additional fixed distributed poisons (lumped burnable poison assemblies) to yield an MTC at BOC within the range analyzed in the plant accident analysis. The end of cycle (EOC) MTC is also limited by the requirements of the accident analysis. Fuel cycles designed to achieve high burnups or with changes to other characteristics are evaluated to ensure that the MTC does not exceed the EOC limit.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The acceptance criteria for the specified MTC are:

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

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

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

Accidents that cause core overheating (either decreased heat removal or increased power production) must be evaluated for results when the MTC is positive. Reactivity accidents that cause increased power production include the control rod withdrawal transient from either zero or full THERMAL POWER. The limiting overheating event relative to plant response is based on the maximum difference between core power and steam generator heat removal during a transient. The most limiting event with respect to positive MTC is a [rod withdrawal accident from zero power, also referred to as a startup accident (Ref. 4)].

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

(continued)

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

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

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

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

LCO

LCO 3.1.4 requires the MTC to be within specified limits of the CORE OPERATING LIMITS REPORT (COLR) (Ref. 5) to ensure that the core operates within the assumptions of the accident analysis. During the reload core safety evaluation, the MTC is analyzed to determine that its values remain within the bounds of the original accident analysis during operation. The limit of [+0.9E-4] (%Δk/k)/F on positive MTC when THERMAL POWER is less than [95%] of RTP assures that core overheating accidents will not violate the accident analysis assumptions. The requirement for a negative MTC when THERMAL POWER is [95%] of RTP or greater ensures that core operation will be stable. The negative MTC limit for EOC specified in the COLR ensures that core overcooling accidents will not violate the accident analysis assumptions. MTC is a core physics parameter determined by the fuel and fuel cycle design, and cannot be easily controlled once the core design is fixed. During operation, therefore, the conditions of the LCO can only be ensured through measurement. The surveillance checks at BOC and EOC on MTC provide confirmation that the MTC is behaving as anticipated so that the acceptance criteria are met.

APPLICABILITY

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

(continued)

(continued)

BASES (continued)

APPLICABILITY (continued)

analysis. In MODE 2, the limits must also be maintained to ensure that startup and subcritical accidents (such as the uncontrolled CONTROL ROD assembly or group withdrawal) will not violate the assumptions of the accident analysis. In MODES 3, 4, 5, and 6, this LCO is not applicable, since no Design Basis Accidents (DBAs) using the MTC as an analysis assumption are initiated from these MODES. However, the variation of MTC with temperature in MODES 3, 4, and 5 for DBAs initiated in MODES 1 and 2 is accounted for in the subject accident analysis. The variation of MTC with temperature assumed in the safety analysis is accepted as valid once the BOC and middle of cycle measurements are used for normalization.

ACTIONS

A.1

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

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

Required Action A.1 is modified by a Note that allows continued operation if A.1 is completed and the requirements of LCO 3.1.7 are met.

B.1

If the required administrative withdrawal limits at BOC are not established within 24 hours, the unit must be placed in MODE 2 with K_{eff} < 1.0 to prevent operation with an MTC which is more positive than that assumed in safety analyses.

(continued)

(continued)

BASES (continued)

ACTIONS (continued)

The allowed 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.

C.1

Exceeding the EOC MTC limit means that the safety analysis assumptions for the EOC accidents that use a bounding negative MTC value may be invalid. If the EOC MTC limit is exceeded, the plant must be placed in a MODE or Condition in which the LCO requirements are not applicable. This is done by placing the plant in at least MODE 4 within 12 hours.

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

SURVEILLANCE REQUIREMENTS

SR 3.1.4.1, SR 3.1.4.2 and SR 3.1.4.3 are modified by a Note identifying that the provisions of SR 3.0.4 are not applicable since the unit must be in MODE 1 or 2 to measure the MTC.

SR 3.1.4.1

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

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

(continued)

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.4.2 and SR 3.1.4.3

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

SR 3.1.4.3 is modified by a Note that includes the following requirements:

a. If the 300 ppm surveillance limit is exceeded, it is possible that the EOC limit on MTC could be reached before the planned end of the cycle. Because the MTC changes slowly with core depletion, the surveillance frequency of 14 effective full power days is sufficient to avoid exceeding the EOC limit.

b. The surveillance limit for RTP boron concentration of 60 ppm is conservative. If the measured MTC at 60 ppm is more positive than the 60 ppm surveillance limit, the EOC limit will not be exceeded because of the gradual manner in which MTC changes with core burnup.

REFERENCES

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

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

3. WCAP 9273-NP-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985.

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

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

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.5 Rod Group Alignment Limits

BASES

BACKGROUND

The OPERABILITY of the shutdown and control rods are initial assumptions in all safety analyses which assume rod insertion upon reactor trip. Maximum rod misalignment is an initial assumption in the safety analysis which directly affects core power distributions and assumptions of available SHUTDOWN MARGIN (SDM). The applicable criteria for these reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10, "Reactor Design," GDC 26, "Reactivity Limits" (Ref. 1), and 10 CFR Part 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plants" (Ref. 2).

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

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

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

The RCCAs are divided among control banks and shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control. A group consists of two or more RCCAs that are electrically paralleled to

(continued)

(continued)

BASES (continued)

BACKGROUND (continued)

step simultaneously. A bank of RCCAs consists of two groups that are moved in a staggered fashion, but always within one step of each other. All units have four control banks and at least two shutdown banks.

The shutdown banks are maintained either in the fully inserted or fully withdrawn position. The control banks are moved in an overlap pattern using the following withdrawal sequence: When control bank A reaches a predetermined height in the core, control bank B begins to move out with control bank A. Control bank A stops at the position of maximum withdrawal and control bank B continues to move out. When control bank B reaches a predetermined height, control bank C begins to move out with control bank B. This sequence continues until control banks A, B, and C are at the fully withdrawn position and control bank D is approximately halfway withdrawn. The insertion sequence is the opposite of the withdrawal sequence. The control rods are arranged in a radially symmetric pattern so that control bank motion does not introduce radial asymmetries in the core power distributions.

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

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

The DRPI System provides a highly accurate indication of actual control rod position, but at a lower precision than the step counters. This system is based on inductive analog signals from a series of coils spaced along a hollow tube

(continued)

(continued)

BASES (continued)

BACKGROUND (continued)	with a center-to-center distance of 3.75 inches, which is 6 steps. To increase the reliability of the system, the inductive coils are connected alternately to data system A or B. Thus, if one system fails, the DRPI will go on half-accuracy with an effective coil spacing of 7.5 inches, which is 12 steps. Therefore, the normal indication accuracy of the DRPI system is ± 6 steps (± 3.75 inches), and the maximum uncertainty is ± 12 steps (± 7.5 inches). With an indicated deviation of 12 steps between the group step counter and DRPI, the maximum deviation between actual rod position and the demand position could be 24 steps, or 15 inches.

APPLICABLE SAFETY ANALYSES

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

a. There be no violations of:

   1. specified acceptable fuel design limits,

   2. centerline fuel temperature, and

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

b. The core must remain subcritical after accident transients.

Two types of misalignment are distinguished. During movement of a control rod group, one rod may stop moving while the other rods in the group continue. This condition may cause excessive power peaking. The second type of misalignment occurs if one rod fails to insert upon a reactor trip and remains stuck fully withdrawn. This condition requires an evaluation to determine that sufficient reactivity worth is held in the control rods to meet the SDM requirement with the maximum worth rod stuck fully withdrawn.

(continued)

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

Two types of analysis are performed in regard to static rod misalignment (Ref. 4). With control banks at their insertion limits, one type of analysis considers the case when any one rod is completely inserted into the core. The second type of analysis considers the case of a completely withdrawn single rod from a bank inserted to its insertion limit. Satisfying limits on departure from nucleate boiling ratio in both of these cases bounds the situation when a rod is misaligned from its group by 12 steps.

Another type of misalignment occurs if one RCCA fails to insert upon a reactor trip and remains stuck fully withdrawn. This condition is assumed in the evaluation to determine that the required SDM is met with the maximum worth RCCA also fully withdrawn (Ref. 5).

The Required Actions in this LCO assure that either deviations from the alignment limits will be corrected or that THERMAL POWER will be adjusted so that excessive local linear heat rates (LHRs) will not occur, and that the requirements on SDM and ejected rod worth are preserved.

Continued operation of the reactor with a misaligned control rod is allowed if the heat flux hot channel factor ($F_Q(Z)$) and the nuclear enthalpy hot channel factor ($F^N_{\Delta H}$) are verified to be within their limits in the CORE OPERATING LIMITS REPORT, (Ref. 6) and the safety analysis is verified to remain valid. When a control rod is misaligned, the assumptions that are used to determine the rod insertion limits, AXIAL FLUX DIFFERENCE limits, and quadrant power tilt limits are not preserved. Therefore, the limits may not preserve the design peaking factors, and $F_Q(Z)$ and $F^N_{\Delta H}$ must be verified directly by incore mapping. Section B 3.2 (Power Distribution Limits) contains more complete discussions of the relation of $F_Q(Z)$ and $F^N_{\Delta H}$ to the operating limits.

Shutdown and control rod OPERABILITY and alignment are directly related to power distributions and SDM, which are initial conditions assumed in safety analyses. Therefore they satisfy Criterion 2 of the NRC Interim Policy Statement.

(continued)

BASES (continued)

LCO

The limits on shutdown or control rod alignments assure that the assumptions in the safety analysis will remain valid. The requirements on operability assure that upon reactor trip, the assumed reactivity will be available and will be inserted. The operability requirements also assure that the RCCAs and banks will move correctly upon command, to maintain the correct power distribution and rod alignment.

The requirement to maintain the rod alignment to within plus or minus 12 steps is conservative. The minimum misalignment assumed in safety analysis is 24 steps (15 inches), and in some cases a total misalignment from fully withdrawn to fully inserted is assumed.

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

[For this facility, an OPERABLE Bank Demand Position Indication System and Digital Rod Position Indication System constitute the following:]

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

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

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

APPLICABILITY

The requirements on RCCA OPERABILITY and alignment are applicable in MODES 1 and 2 because these are the only modes in which neutron (or fission) power is generated, and the OPERABILITY and alignment of rods has the potential to affect the safety of the plant. In MODES 3, 4, 5, and 6, the alignment limits do not apply because the control rods are bottomed and the reactor is shut down and not producing

(continued)

(continued)

BASES (continued)

APPLICABILITY (continued)

fission power. In the shutdown MODES, the OPERABILITY of the shutdown and control rods has the potential to affect the required SDM, but this effect can be compensated for by an increase in the boron concentration of the RCS. See LCO 3.1.1 for SDM in MODES 3, 4, and 5 and LCO 3.9.1 for boron concentration requirements during refueling.

ACTIONS

A.1.1 and A.1.2

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

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

A.2

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

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

B.1

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

(continued)

(continued)

BASES (continued)

ACTIONS (continued)

B.2

An alternative to realigning a single misaligned RCCA to the group average position is to align the remainder of the group to the position of the misaligned or inoperable RCCA. However, this must be done without violating the bank sequence, overlap, and insertion limits specified in LCO 3.1.6, "Shutdown Bank Insertion Limit," and LCO 3.1.7, "Control Bank Insertion Limits." The Completion Time of 1 hour gives the operator sufficient time to adjust the rod positions in an orderly manner.

B.3.1.1 and B.3.1.2

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

In many cases, realigning the remainder of the group to the misaligned rod may not be desirable. For example, realigning control bank B to a rod that is misaligned 15 steps from the top of the core would require a significant power reduction, since control bank D must be moved fully in and control bank C must be moved in to approximately 100 to 115 steps.

Power operation may continue with one RCCA trippable but inoperable or misaligned, provided that SDM is verified within 1 hour. The Completion Time of 1 hour represents the time necessary to determine the actual unit SDM and, if necessary, align and start the necessary systems and components to initiate boration.

B.3.2, B.3.3, B.3.4, B.3.5, B.3.6, and B.3.7

For continued operation with a misaligned rod, RATED THERMAL POWER (RTP) and associated trip setpoints must be reduced, SDM must periodically be verified within limits, hot channel factors (F_Q and $F^N_{\Delta H}$) must be verified within limits, and the safety analyses must be re-evaluated to confirm continued operation is permissible.

Reduction of power to 75% of RTP ensures that local LHR increases due to a misaligned RCCA will not cause the core design criteria to be exceeded (Ref. 7). The Completion

(continued)

(continued)

BASES (continued)

ACTIONS (continued)

Time of 2 hours gives the operator sufficient time to accomplish an orderly power reduction without challenging the Reactor Protection System.

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

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

Once current conditions have been verified acceptable, time is available to perform evaluations of accident analysis to determine that core limits will not be exceeded during a Design Basis Event for the duration of operation under these conditions. A Completion Time of 5 days is sufficient time to obtain the required input data and to perform the analysis.

C.1

If more than one rod is found to be misaligned or becomes misaligned because of bank movement, the unit conditions fall outside of the accident analysis assumptions. Since automatic bank sequencing would continue to cause misalignment, the unit must be placed in a MODE or Condition in which the LCO requirements are not applicable. This is done by placing the plant in at least MODE 3 within 6 hours.

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

(continued)

(continued)

BASES (continued)

ACTIONS (continued)

D.1

In most cases, when more than one rod is found to be trippable and aligned but immovable, the malfunction can be traced to the Rod Control System. Since the majority of Rod Control System malfunctions can be repaired without reactor shutdown, and since the unit conditions are not outside any accident analysis assumptions, the appropriate action is to locate the malfunction and restore the rods to an OPERABLE status. Maintaining the sequence, insertion, and power limits of LCO 3.1.6 and LCO 3.1.7 ensures that core design limits are not exceeded. Since a Completion Time of 72 hours provides adequate time to locate the malfunction as well to obtain parts and perform the repairs, if the malfunction is not corrected in 72 hours it would be indicative of additional problems and plant shutdown would be required.

E.1

When Required Actions B and D cannot be completed within their Completion Time, the unit must be placed in a MODE or Condition in which the LCO requirements are not applicable. This is done by placing the plant in at least MODE 3 within 6 hours, which obviates concerns about the development of undesirable xenon or power distributions. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging the plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.5.1

Verification that individual rod positions are within alignment limits at a Frequency of 12 hours provides a history that allows the operator to detect that a rod is beginning to deviate from its expected position. If the rod position deviation monitor is inoperable, a Frequency of 4 hours accomplishes the same goal. The specified Frequency takes into account other rod position information that is continuously available to the operator in the control room so that during actual rod motion, deviations can immediately be detected.

(continued)

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.5.2

Exercising rod groups at a frequency of 92 days allows the operator to determine that all rods continue to be OPERABLE, even if they are not regularly moved, as, for example, the shutdown banks. A movement of 10 steps is adequate to demonstrate motion without exceeding the alignment limit when only one rod is being moved. The 92-day Frequency takes into consideration other information available to the operator in the control room and other surveillances being performed more frequently which add to the determination of OPERABILITY of the rods.

SR 3.1.5.3

Verification of rod drop times allows the operator to determine that the maximum rod drop time permitted is consistent with the assumed rod drop time used in the safety analysis. Measuring rod drop times prior to reactor criticality after reactor vessel head removal assures that the reactor internals and rod drive mechanism will not interfere with rod motion or rod drop time. Also, every 18 months the rod drop times are verified to ensure that no degradation in these systems has occurred that would adversely affect control rod motion or drop time. Individual rods whose drop times are greater than safety analysis assumptions are not OPERABLE. Individual rods whose drop times are greater than safety analysis assumptions are not OPERABLE.

The 18-month Frequency was developed because it was considered prudent that this Surveillance only be performed during a plant outage. This is due to the plant conditions needed to perform the SR and the potential for an unplanned plant transient if the Surveillance is performed with the reactor at power. Operating experience has shown that these components usually pass this Surveillance when performed on the 18-month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

(continued)

BASES (continued)

REFERENCES

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

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

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

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

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

6. [Unit Name] CORE OPERATING LIMITS REPORT, "[Title]."

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

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.6 Shutdown Bank Insertion Limits

BASES

BACKGROUND

The insertion limits of the shutdown and control rods are initial assumptions in all safety analyses which assume rod insertion upon reactor trip. The insertion limits directly affect core power distributions and assumptions of available SHUTDOWN MARGIN (SDM) and initial reactivity insertion rate. The applicable criteria for these reactivity and power distribution design requirements are 10 CFR 50, Appendix A GDC 10, "Reactor Design," GDC 26, "Reactivity Limits" (Ref. 1), and 10 CFR Part 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 2). Limits on control rod insertion have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

The rod cluster control assemblies (RCCAs) are divided among control banks and shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control. A group consists of two or more RCCAs that are electrically paralleled to step simultaneously. A bank of RCCAs consists of two groups that are moved in a staggered fashion, but always within one step of each other. All plants have four control banks and at least two shutdown banks. See LCO 3.1.5 for control and shutdown rod OPERABILITY and alignment requirements, and LCO 3.1.8 for position indication requirements.

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

(continued)

(continued)

BASES (continued)

BACKGROUND (continued)

shutdown banks are used primarily to help ensure that the required SDM is maintained. The shutdown banks are controlled manually by the control room operator. During normal unit operation, the shutdown banks are either fully withdrawn or fully inserted. The shutdown banks must be completely withdrawn from the core prior to withdrawing any control banks during an approach to criticality. The shutdown banks are then left in this position until the reactor is shut down. They add negative reactivity to shut down the reactor upon receipt of a reactor trip signal.

APPLICABLE SAFETY ANALYSES

On a reactor trip, all RCCAs (shutdown banks and control banks), except the most reactive RCCA, are assumed to insert into the core. The shutdown banks shall be at or above their insertion limits and available to insert the maximum amount of negative reactivity on a reactor trip signal. The control banks may be partially inserted in the core as allowed by LCO 3.1.7, "Control Bank Insertion Limits." The shutdown bank insertion limit is established to ensure that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM (see LCOs 3.1.1 and 3.1.2) following a reactor trip from full power. The combination of control banks and shutdown banks (less the most reactive RCCA which is assumed to be fully withdrawn) is sufficient to take the reactor from full power conditions at rated temperature to zero power, and to maintain the required SDM at rated no-load temperature (Ref. 3). The shutdown bank insertion limit also limits the reactivity worth of an ejected shutdown rod.

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

a. There be no violations of:

   1. specified acceptable fuel design limits,

   2. centerline fuel temperature, and

(continued)

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

3. RCS pressure boundary damage; and

b. The core must remain subcritical after accident transients.

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

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

LCO

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

The shutdown bank insertion limits are defined in the CORE OPERATING LIMITS REPORT (Ref. 4).

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

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

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

APPLICABILITY

The shutdown banks must be within their insertion limits with the reactor in MODE 1 and MODE 2. The applicability in MODE 2 begins within 15 minutes prior to initial control bank withdrawal during an approach to criticality, and continues throughout MODE 2 until all control bank rods are again fully inserted by scram or during shutdown. This ensures that a sufficient amount of negative reactivity is

(continued)

(continued)

BASES (continued)

APPLICABILITY (continued)

available to shut down the reactor and maintain the required SDM following a reactor trip. The shutdown banks do not have to be within their insertion limits in MODE 3, unless an approach to criticality is being made. The reactor is not critical or approaching criticality in MODE 4, 5, or 6, and, therefore, the shutdown banks must be fully inserted.

The Applicability requirements have been modified by a Note that suspends the LCO requirement during SR 3.1.5.2, which assures the freedom of the rods to move. This SR requires the shutdown bank to move below the LCO limits, which would normally violate the LCO.

ACTIONS

A.1 and A.2

When one or more shutdown banks is not within insertion limits, 2 hours are allowed to restore the shutdown banks to within the insertion limits. This is necessary because the available SDM may be significantly reduced with one or more of the shutdown banks not within their insertion limits. Also, initiation of boration within 15 minutes is required since the SDM in MODES 1 and 2 is ensured by adhering to the control and shutdown bank insertion limits (see LCO 3.1.1).

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

In the event that the shutdown rod position indication systems are found to be inoperable, the shutdown rods are considered to be not within limits, and Required Action A.2 applies.

B.1

If the shutdown banks cannot be restored to within their insertion limits within 2 hours, the only other acceptable action is to place the unit in a MODE where the LCO is not applicable. The allowed Completion Time of 6 hours is reasonable based on operating experience to reach the required MODE in an orderly manner and without challenging plant systems.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1

Verification that the shutdown banks are within their insertion limits within 15 minutes prior to an approach to criticality ensures that when the reactor is critical, or being taken critical, the shutdown banks will be available to shut down the reactor, and the required SDM will be maintained following a reactor trip. This SR and Frequency ensure that the shutdown banks are withdrawn before the control banks are withdrawn during a unit startup.

Since the shutdown banks are positioned manually by the control room operator, a verification of shutdown bank position at a Frequency of 12 hours after the reactor is taken critical is adequate to ensure that they are within their insertion limits. Also, the 12-hour Frequency takes into account other information available to the operator in the control room that monitor the status of shutdown rods.

SR 3.1.6.1 is modified by a Note that allows exception to SR 3.0.4. SR 3.0.4 is not applicable before entering the Applicability Condition of "within 15 minutes prior to initial control bank withdrawal," because the SR is specifically selected to be concurrent with the Applicability.

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

REFERENCES

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

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

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

4. [Unit Name] COLR, "[Title]."

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.7 Control Bank Insertion Limits

BASES

BACKGROUND

The insertion limits of the shutdown and control rods are initial assumptions in all safety analyses which assume rod insertion upon reactor trip. The insertion limits directly affect core power distributions and assumptions of available SHUTDOWN MARGIN (SDM) and initial reactivity insertion rate. The applicable criteria for these reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10, "Reactor Design," GDC 26, "Reactivity Limits" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 2).

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

The rod cluster control assemblies (RCCAs) are divided among control banks and shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control. A group consists of two or more RCCAs that are electrically paralleled to step simultaneously. A bank of RCCAs consists of two groups that are moved in a staggered fashion, but always within 1 step of each other. All plants have four control banks and at least two shutdown banks. See LCO 3.1.5 for control and shutdown rod operability and alignment requirements, and LCO 3.1.8 for position indication requirements.

The control bank insertion limits are specified in the CORE OPERATING LIMITS REPORT (COLR). An example is provided for information only in Figure B 3.1.7-1. The control banks are required to be at or above the insertion limit lines.

Figure B 3.1.7-1 also indicates how the control banks are moved in an overlap pattern. Overlap is the distance travelled together by two control banks. The predetermined position of control bank C at which control bank D will

(continued)

(continued)

BASES (continued)

BACKGROUND (continued)

begin to move with bank C on a withdrawal will be at 118 steps for a fully withdrawn position of 231 steps. The fully withdrawn position is defined in the COLR.

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

The power density at any point in the core must be limited so that the fuel design criteria are maintained. Together, LCO 3.1.5, "Rod Group Alignment Limits," LCO 3.1.6, "Shutdown Bank Insertion Limits," LCO 3.1.7, "Control Bank Insertion Limits," LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)", provide limits on control component operation and on monitored process variables which ensure that the core operates within the fuel design criteria.

The shutdown and control bank insertion and alignment limits, AXIAL FLUX DIFFERENCE (AFD), and QUADRANT POWER TILT RATIO (QPTR) are process variables that together characterize and control the three-dimensional power distribution of the reactor core. Additionally, the control bank insertion limits control the reactivity that could be added in the event of a rod ejection accident, and the shutdown and control bank insertion limits assure the required SDM is maintained.

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

APPLICABLE SAFETY ANALYSES

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

(continued)

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

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

a. There be no violations of:

   1. specified acceptable fuel design limits,

   2. centerline fuel temperature, and

   3. Reactor Coolant System pressure boundary damage; and

b. The core must remain subcritical after accident transients.

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

The SDM requirement is assured by limiting the control and shutdown bank insertion limits so that allowable inserted worth of the RCCAs is such that sufficient reactivity is available in the rods to shut down the reactor to hot zero power with a reactivity margin which assumes the maximum worth RCCA remains fully withdrawn upon trip (Ref. 4).

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

The control and shutdown bank insertion limits ensure that safety analyses assumptions for SDM, ejected rod worth, and power distribution peaking factors are preserved (Ref. 5).

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

(continued)

BASES (continued)

LCO

The limits on shutdown and control banks sequence, overlap, and physical insertion as defined in the COLR (Ref. 6) must be maintained because they serve the function of preserving power distribution, ensuring that the SDM is maintained, ensuring that ejected rod worth is maintained, and ensuring adequate negative reactivity insertion on trip. The overlap between control banks provides more uniform rates of reactivity insertion and withdrawal and is imposed to maintain acceptable power peaking during control bank motion.

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

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

APPLICABILITY

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

This LCO has been modified by a Note that permits the requirements of this LCO to be not applicable during the performance of SR 3.1.5.2. This SR requires that the RCCAs be moved at least every 92 days to verify their OPERABILITY. The individual RCCAs are moved at least 10 steps and then returned to their original position.

ACTIONS

A.1, A.2, B.1, and B.2

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

(continued)

(continued)

BASES (continued)

ACTIONS (continued)

a. Reducing power to be consistent with rod position; or

b. Moving rods to be consistent with power.

Also, immediate initiation of boration to regain SDM is required since the SDM in MODES 1 and 2 normally ensured by adhering to the control and shutdown bank insertion limits (see LCO 3.1.1) has been upset.

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

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

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

In the event that a rod group is found to be inoperable, the rod group is considered to be not within limits, and Required Actions A.2 or B.2, and LCO 3.1.5 apply.

C.1

If Required Actions A.1 and A.2, or B.1 and B.2 cannot be completed within the associated Completion Times, the plant must be placed in a MODE where the LCO is not applicable. The allowed Completion Time of 6 hours is reasonable based on operating experience to reach the required MODE in an orderly manner and without challenging plant systems.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.1.7.1

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

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

SR 3.1.7.2

With an OPERABLE bank insertion limit monitor, verification of the control banks insertion limits at a Frequency of 12 hours is sufficient to ensure OPERABILITY of the bank insertion limit monitor and to detect control banks that may be approaching the control bank insertion limits since, normally, very little rod motion occurs in 12 hours. If the insertion limit monitor becomes inoperable, verification of the control bank position at a Frequency of 4 hours is sufficient to detect control banks that may be approaching the control bank insertion limits.

SR 3.1.7.2 is modified by a Note that allows exception to SR 3.0.4. SR 3.0.4 is not applicable, since the unit must be in the applicable MODES in order to perform Surveillances which demonstrate the LCO limits are met.

SR 3.1.7.3

When control banks are maintained within their insertion limits as checked by SR 3.1.7.2 above, it is unlikely that their sequence and overlap will not be in accordance with requirements provided in the COLR. A Surveillance Frequency of 12 hours is consistent with the insertion limit check above in SR 3.1.7.2.

(continued)

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)	SR 3.1.7.3 is modified by a Note that allows exception to SR 3.0.4. SR 3.0.4 is not applicable, since the unit must be in the applicable MODES in order to perform Surveillances which demonstrate the LCO limits are met.

REFERENCES

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

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

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

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

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

BASES (continued)

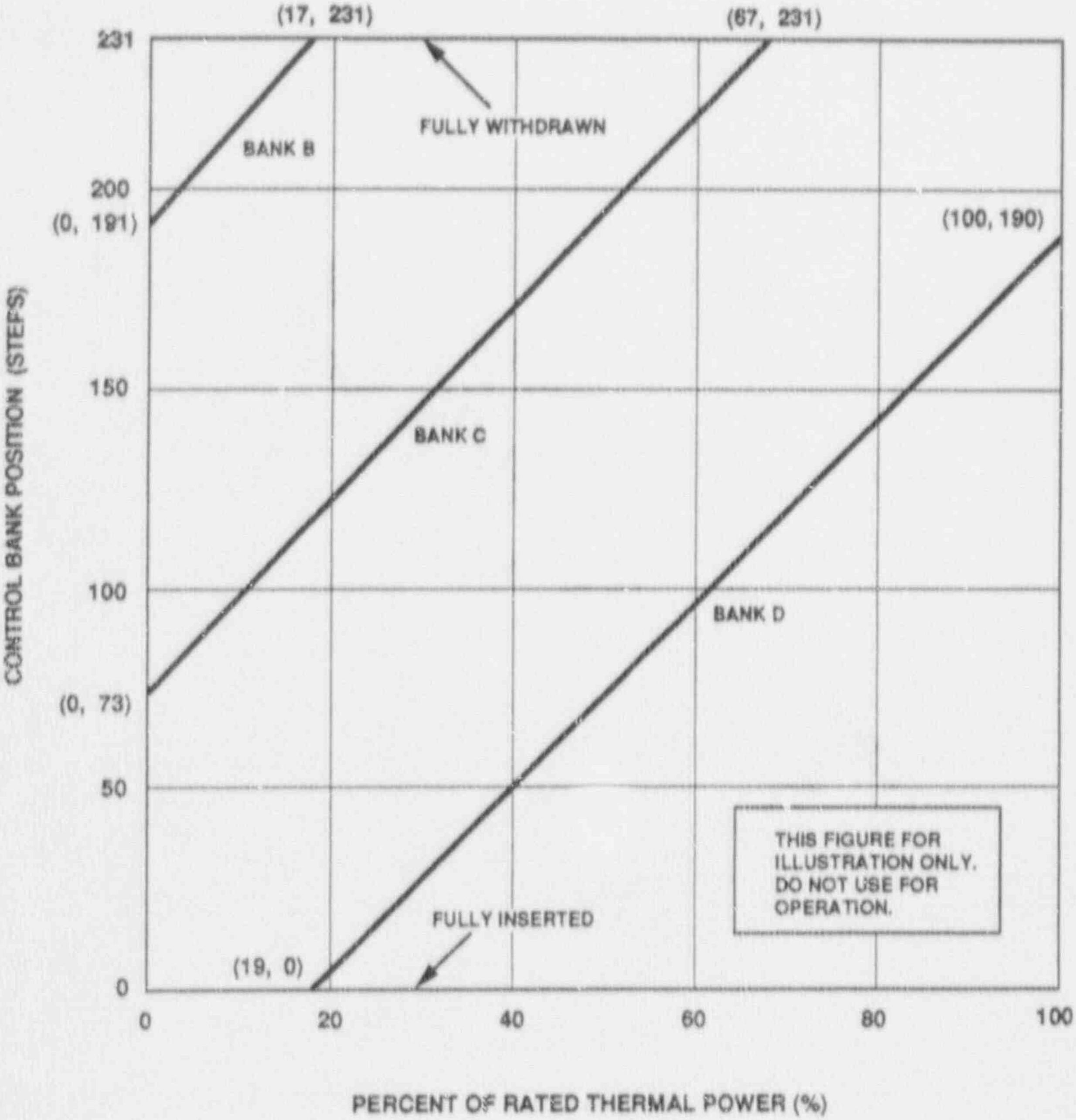


Figure B 3.1.7-1 (Page 1 of 1)

Control Bank Insertion vs. Percent RATED THERMAL POWER

B 3.1 REACTIVITY CONTROL SYSTEM

B 3.1.8 Rod Position Indication

BASES

BACKGROUND

According to GDC 13 (Ref. 1), instrumentation to monitor variables and systems over their operating ranges during normal operation, anticipated operational occurrences, and accident conditions must be OPERABLE. LCO 3.1.8 is required to ensure OPERABILITY of the control rod position indicators to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits.

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

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

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

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

(continued)

(continued)

BASES (continued)

BACKGROUND (continued)

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

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

The DRPI System provides a highly accurate indication of actual control rod position, but at a lower precision than the step counters. This system is based on inductive analog signals from a series of coils spaced along a hollow tube with a center-to-center distance of 3.75 inches, which is 6 steps. To increase the reliability of the system, the inductive coils are connected alternately to data system A or B. Thus, if one system fails, the DRPI will go on half-accuracy with an effective coil spacing of 7.5 inches, which is 12 steps. Therefore, the normal indication accuracy of the DRPI System is $\pm$ 6 steps ($\pm$ 3.75 inches), and the maximum uncertainty is $\pm$ 12 steps ($\pm$ 7.5 inches). With an indicated deviation of 12 steps between the group step counter and DRPI, the maximum deviation between actual rod position and the demand position could be 24 steps, or 15 inches.

APPLICABLE SAFETY ANALYSES

Control rod and shutdown position accuracy is essential during power operation. Power peaking, ejected rod worth, or SDM limits may be violated in the event of a Design Basis Accident (Ref. 2), with control or shutdown rods operating outside their limits undetected. Therefore, the acceptance criteria for rod position indication is that rod positions must be known with sufficient accuracy in order to verify the core is operating within the group sequence, overlap, design peaking limits, ejected rod worth, and with minimum SDM (LCO 3.1.6, "Shutdown Bank Insertion Limits,"

(continued)

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

LCO 3.1.7, "Control Bank Insertion Limits"). The rod positions must also be known in order to verify the verify the alignment limits are preserved (LCO 3.1.5. "Rod Group Alignment Limits"). Control rod positions are continuously monitored to provide operators with information that assures the plant is operating within the bounds of the accident analysis assumptions. The control rod position indicator channels satisfy Criterion 2 of the NRC Interim Policy Statement. The control rod position indicators monitor control rod position, which is an accident initial condition.

LCO

LCO 3.1.7 specifies that one DRPI System and one Bank Demand Position Indication System be OPERABLE for each control rod. For the control rod position indicators to be OPERABLE requires meeting the surveillance requirement of the LCO and the following:

a. The DRPI System has passed a CHANNEL FUNCTIONAL CHECK within the prescribed interval;

b. For the DRPI System there are no failed coils; and

c. The Bank Demand Indication System has been calibrated either in the fully inserted position or to the DRPI System.

The agreement between the Bank Demand Position Indication System and the DRPI System is within the limit, indicating that the Bank Demand Position Indication System is adequately calibrated for measurement of control rod bank position.

A deviation of less than the allowable limit given in the CORE OPERATING LIMITS REPORT in position indication for a single control rod ensures high confidence that the position uncertainty of the corresponding control rod group is within the assumed values used in the analysis that specified control rod group insertion limits.

(continued)

(continued)

BASES (continued)

LCO (continued)

These requirements prove adequate assurance that control rod position indication during power operation and PHYSICS TESTS is accurate, and that design assumptions are not challenged. OPERABILITY of the position indicator channels ensures that inoperable, misaligned, or mispositioned control rods can be detected. Therefore, power peaking, ejected rod worth, and SDM can be controlled within acceptable limits.

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

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

APPLICABILITY

The requirements on the DRPI and step counters are only applicable in MODES 1 and 2 (consistent with LCOs 3.1.5, 3.1.6, and 3.1.7), because these are the only modes in which power is generated, and the OPERABILITY and alignment of rods has the potential to affect the safety of the plant. In the shutdown MODES, the OPERABILITY of the shutdown and control banks has the potential to affect the required SDM, but this effect can be compensated for by an increase in the boron concentration of the Reactor Coolant System.

ACTIONS

A.1

When one DRPI channel per group fails, the position of the rod can still be determined by use of the incore movable detectors. Based on experience, normal power operation does not require excessive movement of banks. If a bank has been significantly moved, the Actions of B.1 or B.2 below are required. Therefore, verification of RCCA position within the Completion Time of 8 hours is adequate to allow continued full power operation, since the probability of simultaneously having a rod significantly out of position and an event sensitive to that rod position is small.

(continued)

(continued)

BASES (continued)

ACTIONS (continued)

A.2

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

The allowed Completion Time of 8 hours is reasonable, based on plant operating experience, to reduce power to ≤ 50% of RTP without challenging plant systems and allowing for rod position determination by A.1 above.

B.1 and B.2

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

If, within 8 hours, the rod positions have not been determined, THERMAL POWER must be reduced to less than 50% of RTP to avoid undesirable power distributions that could result from continued operation above 50% of RTP if one or more rods are misaligned by more than 24 steps. The allowed Completion Time of 8 hours provides an acceptable period of time to verify the rod positions.

C.1.1 and C.1.2

With one demand position indictor per bank inoperable, the rod positions can be determined by the digital rod position indication system. Since normal power operation does not require excessive movement of rods, verification that the rod position indicators are OPERABLE and the most withdrawn rod and the least withdrawn rod are ≤ 12 steps apart within the allowed Completion Time of once every 8 hours is adequate.

C.2

Reduction of THERMAL POWER to ≤ 50% of RTP puts the core into a condition where rod position is not significantly affecting core peaking factor limits (Ref. 3). The allowed

(continued)

(continued)

BASES (continued)

ACTIONS (continued)

Completion Time of 8 hours provides an acceptable period of time to verify the rod positions per C.1.1 and C.1.2 or reduce power to ≤ 50% of RTP.

D.1

If the Required Actions cannot be completed within the associated Completion Time, the plant must be placed in a MODE in which the requirement does not apply. This is done by placing the plant in MODE 3 within 6 hours. The allowed 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.

SURVEILLANCE REQUIREMENTS

SR 3.1.8.1

Verification that the DRPI agrees with the demand position within 12 steps provides assurance that the DRPI is operating correctly. Since the DRPI does not display the actual shutdown rod positions between 18 and 210 steps, only points within the indicated ranges are required in comparison.

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

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix A, General Design Criterion 13, "Instrumentation and Control."
2. [Unit Name] FSAR, Section [ ], "[Title]."
3. [Unit Name] FSAR, Section [ ], "[Title]."

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.9 MODE 1 PHYSICS TESTS Exceptions

BASES

BACKGROUND

The primary purpose of the MODE 1 PHYSICS TESTS Exceptions is to permit relaxations of existing LCO to allow the performance of instrumentation calibration tests and special PHYSICS TESTS. The exceptions to LCO 3.2.3, "AXIAL FLUX DIFFERENCE," and LCO 3.2.4, "QUADRANT POWER TILT RATIO," are most often appropriate for xenon stability tests. The exceptions to LCO 3.1.5, "Rod Group Alignment Limits," LCO 3.1.6, "Shutdown Bank Insertion Limit," and LCO 3.1.7, "Control Bank Insertion Limits," may be required in the event that it is necessary or desirable to do special PHYSICS TESTS involving abnormal rod or bank configurations.

Section XI of 10 CFR Part 50, Appendix B (Ref. 1), requires that a test program be established to ensure that structures, systems, and components will perform satisfactorily in service. All functions necessary to ensure that the specified design conditions are not exceeded during normal operation and anticipated operational occurrences must be tested. This testing is an integral part of the design, construction, and operation of the plant. Requirements for notification of the NRC for the purpose of conducting tests and experiments are specified in 10 CFR 50.59 (Ref. 2).

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

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

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

c. Verify the assumptions used to predict unit response;

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

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

(continued)

(continued)

BASES (continued)

BACKGROUND (continued)

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

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

The PHYSICS TESTS required for reload fuel cycles (Ref. 4) in MODE 1 are listed below:

a. Neutron Flux Symmetry;

b. Power Distribution—Intermediate Power;

c. Power Distribution—Full Power; and

d. Critical Boron Concentration—Full Power.

The first test can be performed in either MODE 1 or 2, and the last three tests are performed in MODE 1. These and other supplementary tests may be required to calibrate the nuclear instrumentation or to diagnose operational problems. These tests may cause the operating controls and process variables to deviate from their LCO requirements during their performance. The last two tests are performed at ≥ 90% of RATES THERMAL POWER (RTP).

[1. The Neutron Flux Symmetry Test measures the degree of azimuthal symmetry of the core neutron flux at as low a power level as practical, depending on the method used. The Flux Distribution Method uses incore flux detectors to measure the azimuthal flux distribution at selected locations with the core at ≤ 30% of RTP.]

(continued)

(continued)

BASES (continued)

BACKGROUND (continued)

[2. The Power Distribution—Intermediate Power Test measures the power distribution of the reactor core at intermediate power levels between 40% and 75% of RTP. This test uses the incore flux detectors to measure core power distribution.

3. The Power Distribution—Full Power Test measures the power distribution of the reactor core at ≥ 90% of RTP using incore flux detectors.

4. The Critical Boron Concentration—Full Power Test simply measures the critical boron concentration at greater than 90% of RTP, with all rods fully withdrawn, the lead control bank being at or near its fully withdrawn position, and with the core at equilibrium xenon conditions.]

For initial startups, there are two currently required tests which violate the referenced LCO. The pseudo-ejected rod test, performed at approximately 30% of RTP, and the pseudo-dropped rod test, performed at approximately 50% of RTP, require individual rod misalignments which exceed the limits specified in the relevant LCO.

APPLICABLE SAFETY ANALYSES

The fuel is protected by a Specification LCO, which preserves the initial conditions of the core assumed during the safety analyses. The methods for development of the LCO, which are superseded by this LCO, are described in the Westinghouse Reload Safety Evaluation Methodology Report (Ref. 5). The above-mentioned PHYSICS TESTS, and other tests that may be required to calibrate nuclear instrumentation or to diagnose operational problems, may require the operating controls or process variables to deviate from their LCO limitations.

Reference 6 defines requirements for initial testing of the facility, including PHYSICS TESTS. Tables [14.1-1 and 14.1-2] (Ref. 6) summarize the zero, low power, and power tests. Requirements for reload fuel cycle PHYSICS TESTS are defined in ANSI/ANS-19.6.1-1985 (Ref. 4). Although these

(continued)

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

PHYSICS TESTS are generally accomplished within the limits for all LCOs, conditions may occur when one or more LCOs must be suspended to make completion of PHYSICS TESTS possible or practical. This is acceptable as long as the fuel design criteria are not violated. When one or more of the requirements specified in

LCO 3.1.5	"Rod Group Alignment Limits,"
LCO 3.1.6	"Shutdown Bank Insertion Limits,"
LCO 3.1.7	"Control Bank Insertion Limits,"
LCO 3.2.3	"AXIAL FLUX DIFFERENCE (AFD)," or
LCO 3.2.4	"QUADRANT POWER TILT RATIO (QPTR)"

are suspended for PHYSICS TESTS, the fuel design criteria are preserved as long as the requirements of LCO 3.2.1, "Heat Flux Hot Channel Factor ($F_Q(Z)$)," and LCO 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)" are satisfied. Therefore, LCO 3.1.9 requires surveillance of the hot channel factors to verify that their limits are not being exceeded.

PHYSICS TESTS include measurements of core nuclear parameters or the exercise of control components that affect process variables. Among the process variables involved are AXIAL FLUX DIFFERENCE (AFD) and QUADRANT POWER TILT RATIO (QPTR) which represent initial conditions of the unit safety analyses. Also involved are the movable control components (control and shutdown rods) which are required to shut down the reactor. The limits for these variable are specified for each fuel cycle in the CORE OPERATING LIMITS REPORT (COLR) (Ref. 7).

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

Reference 8 allows special test exceptions to be included as part of the LCO that they affect. However, it was decided to retain this special test exception as a separate LCO, because it was less cumbersome and provided additional clarity.

(continued)

BASES (continued)

LCO

This LCO allows selected control rods and shutdown rods to be positioned outside their specified alignment limits and insertion limits to conduct PHYSICS TESTS in MODE 1, to verify certain core physics parameters. The power level is limited to ≤ 85% of RTP and the power range neutron flux trip setpoint is set at 10% of RTP above the PHYSICS TESTS power level with a maximum setting of 90% of RTP. Violation of LCO 3.1.5, "Rod Group Alignment Limits," LCO 3.1.6, "Shutdown Bank Insertion Limit," LCO 3.1.7, "Control Bank Insertion Limits," LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," or LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," during the performance of PHYSICS TESTS does not pose any threat to the integrity of the fuel as long as the requirements of LCO 3.2.1, "Heat Flux Hot Channel Factor," and LCO 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor," are satisfied.

The requirements of LCO 3.1.5, 3.1.6, 3.1.7, 3.2.3, and 3.2.4 may be suspended during the performance of PHYSICS TESTS provided:

a. THERMAL POWER is maintained ≤ 85% RTP; and

b. Power Range Neutron Flux—High trip setpoints are ≤ 10% RTP above the THERMAL POWER at which the test is performed, with a maximum setting of 90% RTP.

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

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

APPLICABILITY

This LCO is applicable in MODE 1 when performing PHYSICS TESTS. The applicable PHYSICS TESTS are performed at ≤ 85% of RTP. Other PHYSICS TESTS are performed at full power but do not require violation of any existing LCO, and therefore do not require a PHYSICS TESTS exception. The PHYSICS TESTS performed in MODE 2 are covered by LCO 3.1.10, "MODE 2 PHYSICS TESTS exceptions."

(continued)

BASES (continued)

ACTIONS

A.1 and A.2

When THERMAL POWER is > 85% of RTP, the only acceptable actions are to reduce THERMAL POWER to ≤ 85% of RTP or to suspend the PHYSICS TESTS exceptions. With the PHYSICS TEST exceptions suspended, the PHYSICS TESTS may proceed if all other LCO requirements are met. Fuel integrity may be challenged with control rods or shutdown rods misaligned and THERMAL POWER > 85% of RTP. The allowed Completion Time of 1 hour is reasonable, based on operating experience, to complete the Required Action in an orderly manner without challenging plant systems. This Completion Time is also consistent with the Required Actions of the LCO suspended by the PHYSICS TESTS. In the event that any withdrawn rod is found to be inoperable, the Required Actions A.1 and A.2, and LCO 3.1.5 apply.

B.1 and B.2

When the Power Range Neutron Flux—High trip setpoints are > 10% of RTP above the PHYSICS TESTS power level or > 90% of RTP, the Reactor Trip System (RTS) may not provide the required degree of core protection if the trip setpoint is greater than the specified value.

The only acceptable actions are to restore the trip setpoint to the allowed value or to suspend the performance of the PHYSICS TESTS exceptions. The Completion Time of 1 hour is based on the practical amount of time it may take to restore the Neutron Flux—High trip setpoints to the correct value, consistent with operating plant safety. This Completion Time is consistent with the Required Actions of the LCO suspended by the PHYSICS TESTS.

SURVEILLANCE REQUIREMENTS

SR 3.1.9.1

Verification that the THERMAL POWER level is ≤ 85% of RTP will ensure that the required core protection is provided during the performance of PHYSICS TESTS. Control of the reactor power level is a vital parameter and is closely monitored during the performance of PHYSICS TESTS. A Surveillance Frequency of 1 hour is sufficient to ensure that the power level does not exceed the limit.

(continued)

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.9.2

Verification of the Power Range Neutron Flux—High trip setpoints within 8 hours prior to initiation of the PHYSICS TESTS will ensure that the RTS is properly set to perform PHYSICS TESTS. Verifying the trip setpoint at a Frequency of 8 hours during the performance of the PHYSICS TESTS ensures that the RTS will provide the required core protection.

SR 3.1.9.3

The performance of SR 3.2.1.1 and SR 3.2.2.1 measures the core heat flux hot channel factor and the nuclear enthalpy rise hot channel factor, respectively. If the requirements of these LCO are met, the core has adequate protection from exceeding its design limits while other LCO requirements are suspended. The Frequency of 12 hours is based on operating experience and the practical amount of time that it may take to run an incore flux map and calculate the hot channel factors.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," 1988.

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

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

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

5. WCAP-9273-NP-A, "Westinghouse Reload Safety Evaluation Methodology Report," July 1985.

6. [Unit Name] FSAR, Section [14], "Initial Test and Operation."

(continued)

(continued)

BASES (continued)

REFERENCES (continued)

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

8. WCAP-11618, "MERITS Program — Phase II, Task 5, Criteria Application," including Addendum 1, April 1989.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.10 MODE 2 PHYSICS TESTS Exceptions

BASES

BACKGROUND

The primary purpose of the MODE 2 PHYSICS TESTS exceptions is to permit relaxations of existing LCOs to allow certain PHYSICS TESTS to be performed. Section XI of 10 CFR Part 50, Appendix B (Ref. 1), requires that a test program be established to ensure that structures, systems, and components will perform satisfactorily in service. All functions necessary to ensure that the specified design conditions are not exceeded during normal operation and anticipated operational occurrences must be tested. This testing is an integral part of the design, construction, and operation of the plant. Requirements for notification of the NRC for the purpose of conducting tests and experiments are specified in 10 CFR 50.59 (Ref. 2).

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

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

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

c. Verify the assumptions used to predict unit response;

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

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

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

(continued)

(continued)

BASES (continued)

BACKGROUND (continued)

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

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

a. Critical Boron Concentration—Control Rods Withdrawn;

b. Critical Boron Concentration—Control Rods Inserted;

c. Control Rod Group Worth;

d. Isothermal Temperature Coefficient (ITC); and

e. Neutron Flux Symmetry.

The first four tests are performed in MODE 2, and the last test can be performed in either MODE 1 or 2. These and other supplementary tests may be required to calibrate the nuclear instrumentation or to diagnose operational problems. These tests may cause the operating controls and process variables to deviate from their LCO requirements during their performance.

[The Critical Boron Concentration—Control Rods Withdrawn Test measures the critical boron concentration at hot zero power (HZP). With all rods out, the lead control bank is at or near its fully withdrawn position. HZP is where the core is critical ($k_{eff} = 1.0$), and the Reactor Coolant (RCS) is at design temperature and pressure for zero power. Performance of this test should not violate any of the referenced LCOs.

The Critical Boron Concentration—Control Rods Inserted Test measures the critical boron concentration at HZP with a bank having a worth of at least 1% $\Delta K/K$ when fully inserted into the core. This test is used to measure the boron reactivity coefficient. With the core at HZP and all banks fully withdrawn, the boron concentration of the reactor coolant is gradually lowered in continuous manner.]

(continued)

(continued)

BASES (continued)

BACKGROUND (continued)

The selected bank is then inserted to make up for the decreasing boron concentration until the selected bank has been moved over its entire range of travel. The reactivity resulting from each incremental bank movement is measured with a reactivity computer. The difference between the measured critical boron concentration with all rods fully withdrawn and with the bank inserted is determined. The boron reactivity coefficient is determined by dividing the measured bank worth by the measured boron concentration difference. Performance of this test could violate LCO 3.1.5, "Rod Group Alignment Limits," LCO 3.1.6, "Shutdown Bank Insertion Limit," or LCO 3.1.7, "Control Bank Insertion Limits."

The Control Rod Group Worth Test is used to measure the reactivity worth of selected control banks. This test is performed at HZP and has three alternative methods of performance. The first method, the Boron Exchange Method, varies the reactor coolant boron concentration and moves The selected control bank in response to the changing boron concentration. The reactivity changes are measured with a reactivity computer. This sequence is repeated for the remaining control banks. The second method, the Rod Swap Method, measures the worth of a predetermined reference bank using the Boron Exchange Method above. The reference bank is then nearly fully inserted into the core. The selected bank is then inserted into the core as the reference bank is withdrawn. The HZP critical conditions are then determined with the selected bank fully inserted into the core. The worth of the selected bank is inferred based on the position of the reference bank with respect to the selected bank. This sequence is repeated as necessary for the remaining control banks. The third method, the Boron Endpoint Method, moves the selected control bank over its entire length of travel and then varies the reactor coolant boron concentration to achieve HZP criticality again. The difference in boron concentration is the worth of the selected control bank. This sequence is repeated for the remaining control banks. Performance of this test could violate LCO 3.1.5, "Rod Group Alignment Limits," LCO 3.1.6, "Shutdown Bank Insertion Limit," or LCO 3.1.7, "Control Bank Insertion Limits."

(continued)

(continued)

BASES (continued)

BACKGROUND (continued)

[The ITC Test measures the ITC of the reactor. This test is performed at HZP and has two methods of performance. The first method, the Slope Method, varies RCS temperature in a slow and continuous manner. The reactivity change is measured with a reactivity computer as a function of the temperature change. The ITC is the slope of the reactivity versus the temperature plot. The test is repeated by reversing the direction of the temperature change and the final ITC is the average of the two calculated ITCs. The second method, the Endpoint Method, changes the RCS temperature and measures the reactivity at the beginning and end of the temperature change. The ITC is the total reactivity change divided by the total temperature change. The test is repeated by reversing the direction of the temperature change, and the final ITC is the average of the two calculated ITCs. Performance of this test could violate LCO 3.4.2, "Minimum Temperature for Criticality."

The Flux Symmetry Test measures the degree of azimuthal symmetry of the neutron flux at as low a power level as practical, depending on the test method employed. This test can be performed at HZP (Control Rod Worth Symmetry Method or at ≤ 30% of RATED THERMAL POWER (RTP) (Flux Distribution Method). The Control Rod Worth Symmetry Method inserts a control bank, which can then be withdrawn to compensate for the insertion of a single control rod from a symmetric set. The symmetric rods of each set are then tested to evaluate the symmetry of the control rod worth and neutron flux (power distribution). A reactivity computer is used to measure the control rod worths. Performance of this test could violate LCO 3.1.5, "Rod Group Alignment Limits," LCO 3.1.6, "Shutdown Bank Limit," Insertion Limit," or LCO 3.1.7, "Control Bank Insertion Limits." The Flux Distribution Method uses the incore flux detectors to measure the azimuthal flux distribution at selected locations with the core at ≤ 30% RTP.]

APPLICABLE SAFETY ANALYSES

The fuel is protected by Technical Specification LCOs that preserve the initial conditions of the core assumed during the safety analyses. The methods for development of the

(continued)

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

LCOs that are excepted by this LCO are described in the Westinghouse Reload Safety Evaluation Methodology report (Ref. 5). The above-mentioned PHYSICS TESTS, and other tests that may be required to calibrate nuclear instrumentation or to diagnose operational problems, may require the operating control or process variables to deviate from their LCO limitations.

Reference 6 defines requirements for initial testing of the facility, including PHYSICS TESTS. Tables [14.1-1 and 14.1-2] (Ref. 6) summarize the zero, low power, and power tests. Requirements for reload fuel cycle PHYSICS TESTS are defined in ANSI/ANS-19.6.1-1985 (Ref. 4). Although these PHYSICS TESTS are generally accomplished within the limits for all LCOs, conditions may occur when one or more LCOs must be suspended to make completion of PHYSICS TESTS possible or practical. This is acceptable as long as the fuel design criteria are not violated. When one or more of the requirements specified in

LCO 3.1.4	"Moderator Temperature Coefficient,"
LCO 3.1.5	"Rod Group Alignment Limits,"
LCO 3.1.6	"Shutdown Bank Insertion Limit,"
LCO 3.1.7	"Control Bank Insertion Limits," and
LCO [illegible].4.	"Minimum Temperature for Criticality,"

are suspended for PHYSICS TESTS, the fuel design criteria are preserved as long as the power level is limited to ≤ 5% RTP and the reactor coolant temperature is kept ≥ 531°F.

The PHYSICS TESTS include measurement of core nuclear parameters or the exercise of control components that affect process variables. Among the process variables involved are AXIAL FLUX DIFFERENCE and QUADRANT POWER TILT RATIO, which represent initial conditions of the unit safety analyses. Also involved are the movable control components (control and shutdown rods), which are required to shut down the reactor. The limits for these variables are specified for each fuel cycle in the CORE OPERATING LIMITS REPORT (COLR) (Ref. 6). PHYSICS TESTS meet the criteria for inclusion in the Technical Specifications, since the components and process variable LCOs suspended during PHYSICS TESTS meet criteria 1, 2, and 3 of the NRC Interim Policy Statement.

(continued)

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)	Reference 7 allows special test exceptions (STE) to be included as part of the LCO that they affect. It was decided, however, to retain this STE as a separate LCO because it was less cumbersome and provided additional clarity.
LCO	This LCO allows the reactor parameters of moderator temperature coefficient and minimum temperature for criticality to be outside their specified limits. In addition, it allows selected control and shutdown rods to be positioned outside of their specified alignment and insertion limits. Operation beyond specified limits is permitted for the purpose of performing PHYSICS TESTS and poses no threat to fuel integrity, provided the SRs are met. The requirements of LCO 3.1.4, 3.1.5, 3.1.6, 3.1.7, and 3.4.2 may be suspended during the performance of PHYSICS TESTS provided: a. THERMAL POWER is maintained ≤ 5% RTP, and b. RCS lowest loop average temperature is ≥ [531] °F. [For this facility, the following support systems are required to be OPERABLE to ensure that the LCO and SR conditions are met: [LIST]] [For this facility, the required support systems, which upon their failure do not result in the conditions of this LCO to not be met, and their justification are as follows: [list]]
APPLICABILITY	This LCO is applicable in MODE 2 when performing low power PHYSICS TESTS. The applicable PHYSICS TESTS are performed in MODE 2 at HZP. Other PHYSICS TESTS are performed in MODE 1 and are addressed in LCO 3.1.9, "MODE 1 PHYSICS TESTS Exceptions."

(continued)

BASES (continued)

ACTIONS

A.1

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

B.1 and B.2

When the RCS lowest T_{avg} is < 531°F, the appropriate action is to restore T_{avg} to within its specified limit. The allowed Completion Time of 15 minutes provides time to restore T_{avg} to within limits without allowing the plant to remain in an unacceptable condition for an extended period of time. Operation with the reactor critical and with temperature below 531°F could violate the assumptions for accidents analyzed in the safety analyses. If the Required Action cannot be completed within the associated Completion Time, the plant must be placed in a MODE in which the requirement does not apply. This is done by placing the plant in MODE 3 within an additional 15 minutes. The additional 15 minutes is reasonable, based on operating experience, to reach MODE 3 in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.10.1

The power-range and intermediate-range neutron detectors must be verified to be OPERABLE in MODE 2 by LCO 3.3.1. An ANALOG CHANNEL OPERATIONAL TEST is performed on each power-range and intermediate-range channel within 12 hours prior to initiation of the PHYSICS TESTS. This will ensure that the reactor trip system is properly aligned to provide the required degree of core protection during the performance of the PHYSICS TESTS. The 12-hour time limit is sufficient to ensure that the instrumentation is OPERABLE shortly before initiating PHYSICS TESTS.

(continued)

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.10.2

Verification that the RCS lowest loop T_{avg} is $\geq$ 531°F will ensure that the unit is not operating in a Condition that could invalidate the safety analyses. Verification of the RCS temperature at a Frequency of 30 minutes during the performance of the PHYSICS TESTS will provide assurance that the initial conditions of the safety analyses are not violated.

SR 3.1.10.3

Verification that the power level is $\leq$ 5% RTP will ensure that the fuel design criteria are not violated during the performance of the PHYSICS TESTS. The Frequency of once per hour is adequate to ensure that the power level does not exceed the limit. Unit operations are conducted slowly during the performance of PHYSICS TESTS and monitoring the power level at a Frequency of 1 hour is sufficient to ensure that the THERMAL POWER does not exceed the limit.

REFERENCES

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

2. Title 10, Code of Federal Regulations, Part 50.5, "Changes, Tests and Experiments."

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

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

5. WCAP-9273-NP-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985.

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

7. WCAP-11618, "MERITS Program—Phase II, Task 5, Criteria Application," including Addendum 1, April 1989.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.11 SHUTDOWN MARGIN (SDM) Test Exception

BASES

BACKGROUND

The primary purpose of the SDM Test exception is to permit relaxation of the requirements of LCO 3.1.1 during the measurement of control rod worths in MODE 2 during PHYSICS TESTS. Section XI of 10 CFR Part 50, Appendix B (Ref. 1), requires that a test program be established to ensure that structures, systems, and components will perform satisfactorily in service. All functions necessary to ensure that the specified design conditions are not exceeded during normal operation and anticipated operational occurrences must be tested. This testing is an integral part of the design, construction, and operation of the plant. Requirements for notification of the NRC for the purpose of conducting tests and experiments are specified in 10 CFR 50.59 (Ref. 2).

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

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

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

c. Verify the assumptions used to predict unit response;

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

e. Verify that operating and emergency procedures are adequate.

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

(continued)

(continued)

BASES (continued)

BACKGROUND (continued)

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

During the PHYSICS TESTS measurements of control rod worth, it may be necessary to align individual rods and banks in certain configurations and utilize boron concentrations which do not provide sufficient SDM to meet the requirements of LCO 3.1.1. In this situation, it is necessary to invoke special test exceptions (STEs) to allow the necessary PHYSICS TESTS to be completed.

APPLICABLE SAFETY ANALYSES

Special PHYSICS TESTS may require operating the core under controlled conditions for short periods of time with less than the normally required SDM. As such, these tests are not covered by any safety analysis calculations.

The acceptance criteria to allow suspension of certain LCOs for PHYSICS TESTS are that fuel damage criteria are not exceeded. Even if an accident occurs during PHYSICS TESTS with one or more LCOs suspended, fuel damage criteria are preserved because adequate limits on power distribution and shutdown capability are maintained during PHYSICS TESTS.

Reference 6 defines the requirements for initial testing of the facility, including PHYSICS TESTS. Requirements for reload fuel cycle PHYSICS TESTS are defined in ANSI/ANS-19.6.1-1985 (Ref. 4). PHYSICS TESTS for reload fuel cycles are given in Table 1 of ANSI/ANS 19-6.1-1985. Although these PHYSICS TESTS are generally accomplished within the limits of all LCOs, Conditions may occur when one or more LCOs must be suspended to make completion of PHYSICS TESTS possible or practical. This is acceptable as long as the fuel design criteria are not violated. As long as the linear heat rate remains within its limit, fuel design criteria are preserved.

PHYSICS TESTS meet the criteria for inclusion in Technical Specifications, since the components and process variable

(continued)

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)	LCOs suspended during PHYSICS TESTS meet Criteria 1, 2, and 3 of the NRC Interim Policy Statement.
LCO	This LCO provides an exemption to the requirements of LCO 3.1.1 under controlled conditions. These conditions require that at least the highest estimated control rod worths be available for trip insertion. It is assumed that this available negative reactivity will be sufficient to shut down the core if required, assuming there is not a concurrent boron dilution or cooldown event. This exemption is allowed even though there are no bounding safety analyses because the tests are performed under close supervision and provide valuable information on control rod worth and core SDM. [For this facility, the following support systems are required to be OPERABLE to ensure that the LCO and SR conditions are met:] [For this facility, the required support systems, which upon their failure do not result in the conditions of this LCO to not be met, and their justification are: [List]]
APPLICABILITY	This LCO is only applicable in MODE 2, and then only during actual measurement of control rod worths, because this is the only time the exception is required.
ACTIONS	<u>A.1</u> If one or more control rods are not fully inserted and the available trip reactivity from OPERABLE control rods is less than the highest estimated control rod worth, the SDM, assumed for the test conditions, may not be available. Under these conditions, it is necessary to promptly restore the SDM to within limits. The allowed Completion Time of 15 minutes ensures prompt action and provides an acceptable time to initiate boration

(continued)

(continued)

BASES (continued)

ACTIONS (continued)

to restore SDM, without allowing the core to remain in an unacceptable condition for an extended period of time.

B.1

If all control rods are fully inserted, and the reactor is subcritical by less than the highest estimated control rod worth, the SDM, assumed for the test conditions, may not be available. Under these conditions, it is necessary to promptly restore the SDM to within limits.

The allowed Completion Time of 15 minutes provides an acceptable time to initiate boration to restore SDM, without allowing the core to remain in an unacceptable condition for an extended period of time.

SURVEILLANCE REQUIREMENTS

SR 3.1.11.1

In order to establish an acceptable SDM during the measurement of control rod worths, it is necessary to know the position of each control rod. A test Frequency of 2 hours is reasonable, based on normal control rod motion during control rod worth measurements.

SR 3.1.11.1 has been modified by a Note that establishes that the position of only those control rods not fully inserted must be determined. It is assumed that the position and worth of fully inserted control rods is known.

SR 3.1.11.2

One of the assumptions made in granting an STE for LCO 3.1.1, SDM, is that all control rods not fully inserted will fully insert when tripped. This Surveillance is performed to verify that fact.

The Surveillance Frequency of 24 hours prior to reducing the plant SDM below the requirements of LCO 3.1.1 is acceptable, based on the assumption that the control rods will remain OPERABLE and trippable for 24 hours and during the performance of the test.

SR 3.1.11.2 has been modified by a Note that establishes that this surveillance is only required for control rods not

(continued)

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

fully inserted. During the performance of control rod worth measurements, certain control rods remain fully inserted. Since these rods are not relied on to trip, there is no need to demonstrate that they will fully insert when tripped.

REFERENCES

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

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

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

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

5. WCAP-11618, "MERITS Program—Phase II, Task 5, Criteria Application," including Addendum 1, April 1989.

6. [Unit Name] FSAR, Section [14], "Testing Requirements."

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1A Heat Flux Hot Channel Factor ($F_Q(Z)$) (F_{xy} Methodology) (Constant Axial Offset Control (CAOC) - AXIAL FLUX DIFFERENCE (AFD) Limits)

BASES

BACKGROUND

The purpose of the limits on the values of $F_Q(Z)$ is to limit the local (i.e. pellet) peak power density. The value of $F_Q(Z)$ varies along the axial height of the core (Z).

$F_Q(Z)$ is defined as the maximum local fuel rod linear power density divided by the average fuel rod linear power density, assuming nominal fuel pellet and fuel rod dimensions. Therefore, $F_Q(Z)$ is a measure of the peak pellet power within the reactor core.

During power operation, the global power distribution is limited by LCO 3.2.3, AXIAL FLUX DIFFERENCE (AFD), and LCO 3.2.4, QUADRANT POWER TILT RATIO (QPTR), which are directly and continuously measured process variables. Therefore, these LCOs preserve core limits on a continuous basis.

$F_Q(Z)$ varies with fuel loading patterns, control bank insertion, fuel burnup, and changes in axial power distribution.

$F_Q(Z)$ is measured periodically using the incore detector system, and measurements are generally taken with the core at or near steady-state conditions.

Using the measured three-dimensional power distributions, it is possible to determine a measured value for $F_Q(Z)$. However, since this value represents a steady-state condition, it will not include variations in the value of $F_Q(Z)$ which would be present during a nonequilibrium situation such as load following.

To account for these possible variations, the steady-state value of the fundamental radial peaking factor (F_{xy}) is adjusted by an elevation-dependent factor to account for the variations in $F_Q(Z)$ due to transient conditions.

(continued)

(continued)

BASES (continued)

BACKGROUND (continued)

Core monitoring and control under non-steady-state conditions are accomplished by operating the core within the limits of the appropriate LCOs, including the limits on AFD, QUADRANT POWER TILT RATIO (QPTR), and control rod insertion.

APPLICABLE SAFETY ANALYSES

This LCO precludes core power distributions from occurring which would violate the following fuel design criteria:

a. During a large-break loss-of-coolant accident (LOCA), the peak cladding temperature must not exceed a limit of 2200°F (10 CFR 50.46) (Ref. 1);

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

c. During an ejected rod accident, the fission energy input to the fuel must not exceed 280 cal/gm (Ref. 2); and

d. The control rods must be capable of shutting down the reactor with a minimum required SHUTDOWN MARGIN with the highest worth control rod stuck fully withdrawn (GDC 26) (Ref. 3).

Limits on $F_Q(Z)$ ensure that the value of the total peaking factor assumed as an initial condition in the accident analyses remains valid. Other criteria must also be met (e.g., maximum cladding oxidation, maximum hydrogen generation, coolable geometry, and long-term cooling). However, the peak cladding temperature is typically most limiting.

$F_Q(Z)$ limits assumed in the LOCA analysis are typically limiting (i.e., lower) relative to the $F_Q(Z)$ assumed in safety analyses for other accidents. Therefore, this LCO provides conservative limits for other accidents.

$F_Q(Z)$ satisfies Criterion 2 of the NRC Interim Policy Statement.

(continued)

BASES (continued)

LCO

The $F_Q(Z)$ shall be limited by the following relationships:

$$F_Q(Z) \leq \frac{CFQ}{P} * K(Z) \qquad \text{for } P > 0.5$$

$$F_Q(Z) \leq \frac{CFQ}{0.5} * K(Z) \qquad \text{for } P \leq 0.5$$

where: CFQ is the F_Q limit at RATED THERMAL POWER (RTP) provided in the CORE OPERATING LIMITS REPORT (COLR),

K(Z) is the normalized $F_Q(Z)$ as a function of core height provided in the COLR, and

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

For this facility, the actual values of CFQ and K(Z) are given in the COLR; however, CFQ is normally a number on the order of [2.32], and K(Z) is a function that looks like the one provided in Figure B 3.2.1A-1.

The $F_Q(Z)$ limits define limiting values for core power peaking that precludes peak cladding temperatures above 2200°F during either a large- or small-break LOCA.

This LCO requires operation within the bounds assumed in the safety analyses. Calculations are performed in the core design process to confirm that the core can be controlled in such a manner during operation that it can stay within the LOCA $F_Q(Z)$ limits. If $F_Q(Z)$ cannot be maintained within the LCO limits, reduction of the core power is required.

Violating the LCO limits for $F_Q(Z)$ could produce unacceptable consequences should a design basis event occur while $F_Q(Z)$ is outside its specified limits.

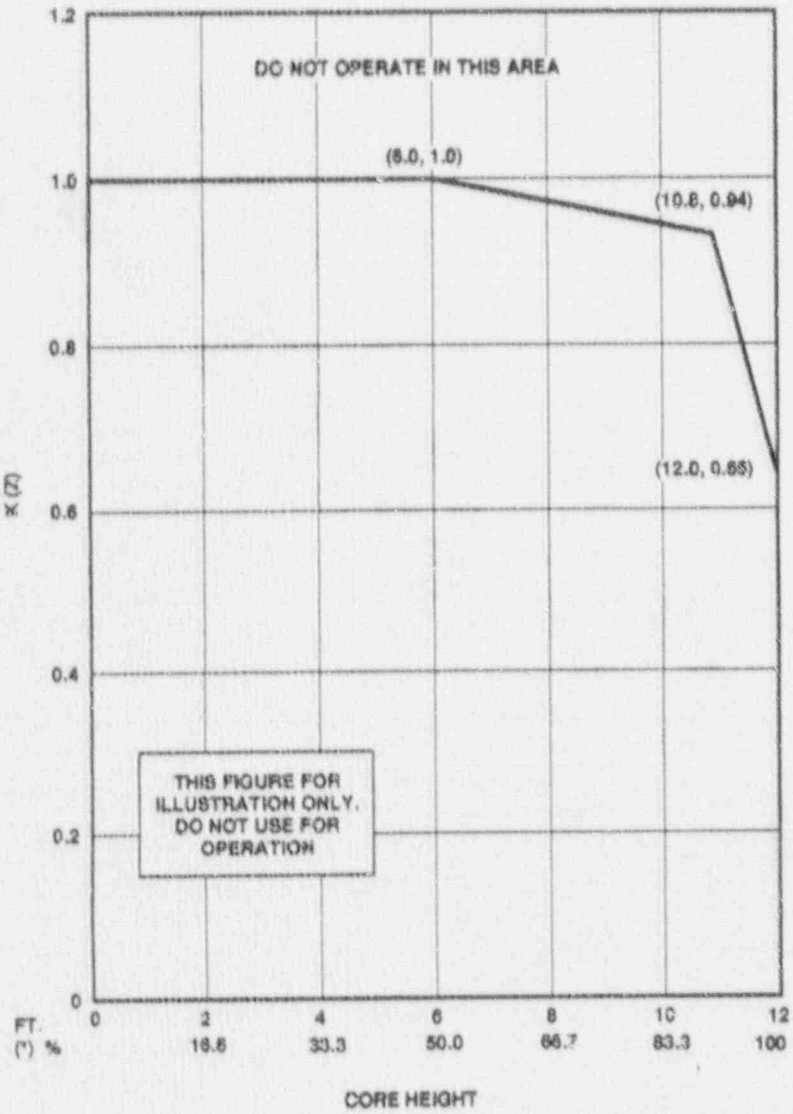
[For this facility, an OPERABLE [Incore Detector System] constitutes the following:]

[For this facility, the following support systems are required OPERABLE to ensure [Incore Detector System] OPERABILITY:]

(continued)

(continued)

BASES (continued)



* For core height of 12 feet

Figure B 3.2.1A-1 (Page 1 of 1)

K(Z) - Normalized $F_Q(Z)$ as a Function of Core Height

(continued)

BASES (continued)

LCO (continued)

[For this facility, those required support systems that upon their failure do not declare the [Incore Detector System] inoperable and their justification are as follows:]

APPLICABILITY

The $F_Q(Z)$ limits must be maintained while in MODE 1 to preclude core power distributions from exceeding the limits assumed in the safety analyses. Applicability in other MODES is not required because there is insufficient stored energy in the fuel or energy being transferred to the reactor coolant to require a limit on the distribution of core power.

ACTIONS

A.1

Reducing THERMAL POWER by at least 1% for each 1% by which $F_Q(Z)$ exceeds its limit maintains an acceptable absolute power density. [For this facility, THERMAL POWER is reduced by the following actions:] The Completion Time of 15 minutes provides an acceptable time to reduce power in an orderly manner and without allowing the plant to remain in an unacceptable condition for an extended period of time.

A.2

When core peaking factors are sufficiently high that LCO 3.2.3 does not permit operation at RTP, the Acceptable Operation Limits for AXIAL FLUX DIFFERENCE are scaled down. This percentage reduction is equal to the amount, expressed as a percentage, by which $F_Q(Z)$ exceeds its specified limit. This assures a near constant maximum linear heat rate in units of kW/ft at the acceptable operation limits. The Completion Time of 4 hours for the change in setpoints is sufficient considering the small likelihood of a severe transient in this relatively short time period and the preceding prompt reduction in THERMAL POWER in accordance with Required Action A.1.

A.3

A reduction of the Power Range Neutron—High Trip Setpoints by at least 1% for each 1% by which $F_Q(Z)$ exceeds its

(continued)

(continued)

BASES (continued)

ACTIONS (continued)

specified limit is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 8 hours is sufficient considering the small likelihood of a severe transient in this period and the preceding prompt reduction in THERMAL POWER in accordance with Required Action A.1.

A.4

Reduction in the Overpower ΔT Trip setpoints by 1% for each 1% by which $F_Q(Z)$ exceeds its limit is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 72 hours is sufficient considering the small likelihood of a severe transient in this period and the preceding prompt reduction in THERMAL POWER in accordance with Required Action A.1.

A.5

Verification that $F_Q(Z)$ has been restored to within its limit by performing SR 3.2.1.1 prior to increasing THERMAL POWER above the limit imposed by Required Action A.1, assures that core conditions during operation at higher power levels are consistent with safety analyses assumptions.

B.1

If the Required Actions of A.1 through A.4 cannot be met within their associated Completion Times, or if $F_Q(Z)$ cannot be determined because of incore detector system inoperability, the plant must be placed in a MODE or condition in which the LCO requirements are not applicable. This is done by placing the plant in at least MODE 2 within 6 hours.

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

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1

Verification that $F_Q(Z)$ is within its limit involves increasing the measured values of $F_Q(Z)$ to allow for manufacturing tolerance and measurement uncertainties and then making a comparison to the limits. These limits are provided in the COLR. Specifically, the measured value of the Heat Flux Hot Channel Factor (F_Q^M) is increased by 3% to account for fuel manufacturing tolerances and by 5% for flux map measurement uncertainty. This is equivalent to increasing the directly measured values of $F_Q(Z)$ by 1.0815% before comparing to LCO limits (Ref. 4).

SR 3.2.1.1 has been modified by a Note, which states that SR 3.0.4 is not applicable. The unit must be in MODE 1 before the surveillance can be performed.

The surveillance Frequency of 31 effective full power days (EFPDs) is adequate to monitor the change of power distribution with core burnup because the power distribution changes relatively slowly for this amount of fuel burnup. The surveillance may be done more frequently if required by the results of SR 3.2.1.2.

Performing the surveillance prior to exceeding 75% RTP after each refueling assures that the $F_Q(Z)$ limit will be met when RTP is achieved.

SR 3.2.1.2

The nuclear design includes calculations which predict that the core can be operated within the $F_Q(Z)$ limits. Since flux maps are taken at steady-state conditions, the axial variations in power distribution for normal operation maneuvers such as load following are not present in the flux map data. These axial variations are, however, conservatively calculated by considering, in the nuclear design process, a wide range of unit maneuvers in normal operation. $F_{XY}(Z)$ is the radial peaking factor, which is one component of $F_Q(Z)$, and should be consistent between the nuclear design values and the measured values ($F_{XY}(Z)$ multiplied by the normalized average axial power at elevation Z gives $F_Q(Z)$).

(continued)

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

The core plane regions applicable to an F_{XY} evaluation exclude the following, measured in percent of core height:

a. Lower core region, from 0% to 15% inclusive;

b. Upper core region, from 85% to 100% inclusive;

c. Grid plane regions, ± 2% inclusive; and

d. Core plane regions, within ± 2% of the bank demand position of the control banks.

The following terms are used in the F_{XY} evaluation:

F^M_{XY} = The measured value of F_{XY} obtained directly from the flux map results.

F^C_{XY} = The measured value, F^M_{XY} multiplied by 1.0815 to account for fuel manufacturing tolerances and flux map measurement uncertainty (Ref. 2).

F^{RTP}_{XY} = The lim t of F_{XY} at RTP.

F^L_{XY} = F^{RTP}_{XY} * [(1 + PFXY * (1 - P)]. (The limit of F_{XY} at the current THERMAL POWER level).

PFXY = The power factor multiplier for F_{XY}.

P = [The fraction of RTP at which F_{XY} was measured.]

F^{PR}_Q = The predicted value of the Heat Flux Hot Channel factor.

F^{RTP}_{XY} and PFXY are provided in the COLR. F^M_{XY} and F^C_{XY} are measured and calculated at discrete core elevations. Note that F_{XY} can be rewritten as $F_{XY}(Z)$ to indicate that F_{XY} varies along the axial height of the core. Flux map data will typically be taken for 30-75 core elevations.

(continued)

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

The top and bottom regions of the core are excluded from the F_{XY} evaluation because of the difficulty of making precise and meaningful measurements in these regions and also because of the low probability that these regions would be more limiting than the central 70% of the core in the accident analyses.

Grid plane regions and rod tip regions are also excluded because the flux data may give spurious values because of the difficulty in lining up flux traces accurately in regions of rapidly varying flux. In addition, these small portions of the core are reduced in local power density because of neutron absorption in the grids and control rods and, therefore, cannot be regions of peak linear power.

An evaluation of $F_{XY}(Z)$ is used to confirm that $F_Q(Z)$ is within its limits. If F^C_{XY} is less than F^{RTP}_{XY}, it is concluded that the LCO limit on $F_Q(Z)$ is met. This is true for flux maps taken at reduced power since the $F_{XY}(Z)$ value will inherently be less as THERMAL POWER is increased. The feedback from the Doppler coefficient and moderator effects will flatten the power distribution with increased THERMAL POWER.

The first Note of this surveillance provides the action to be taken, if F^C_{XY} is greater than F^L_{XY}. In this case, the $F_Q(Z)$ limit may be exceeded. Proportionally increasing the predicted $F^{PR}_Q(Z)$ by the amount that F^L_{XY} is exceeded gives an adjusted $F_Q(Z)$, which is compared with the $F_Q(Z)$ limit. If the adjusted $F_Q(Z)$ exceeds the LCO limit, the operator must perform Required Actions A.1 through A.5.

The second Note in this surveillance states that if F^C_{XY} is greater than F^{RTP}_{XY} but less than F^L_{XY}, then this surveillance shall be repeated within 24 hours after exceeding by $\geq$ 20% RTP the THERMAL POWER at which F^C_{XY} was last determined, so as to demonstrate that $F_{XY}(Z)$ is being sufficiently reduced as power increases. This reduction, because of feedback from the Doppler coefficient and moderator effects, should ensure that when RTP is attained, the measured $F^M_{XY}(Z)$ will be less than F^{RTP}_{XY}.

(continued)

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.2.1.2 has been modified by a third Note, which states that SR 3.0.4 is not applicable. The plant must be in MODE 1 before the surveillance can be performed.

The surveillance Frequency of 31 EFPD is adequate to monitor the change of power distribution with core burnup because the power distribution changes relatively slowly for this amount of fuel burnup. The surveillance may be done more frequently if required by the results of F_{xy} evaluations. Specifically, the F_{xy} evaluation is required by this surveillance if the evaluation shows that F_{xy}^{RTP} is < F_{xy}^{c} and to demonstrate that the LCO is met after being exceeded.

Performing the surveillance prior to exceeding 75% RTP after each refueling assures that the $F_Q(Z)$ limit will be met when RTP is achieved.

REFERENCES

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

2. Regulatory Guide 1.77, Rev. [ ], "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors," [date].

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

4. [WCAP-7308-L-P-A, Evaluation of Nuclear Hot Channel Factor Uncertainties, June 1988.]

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1B Heat Flux Hot Channel Factor ($F_Q(Z)$) (F_Q Methodology) (Relaxed Axial Offset Control (RAOC) - AXIAL FLUX DIFFERENCE (AFD))

BASES

BACKGROUND

The purpose of the limits on the values of $F_Q(Z)$ is to limit the local (i.e pellet) peak power density. The value of $F_Q(Z)$ varies along the axial height of the core (Z).

$F_Q(Z)$ is defined as the maximum local fuel rod linear power density divided by the average fuel rod linear power density, assuming nominal fuel pellet and fuel rod dimensions. Therefore, $F_Q(Z)$ is a measure of the peak fuel pellet power within the reactor core.

During power operation, the global power distribution is limited by LCO 3.2.3, AXIAL FLUX DIFFERENCE (AFD), and LCO 3.2.4, QUADRANT POWER TILT RATIO (QPTR), which are directly and continuously measured process variables. These LCOs along with LCO 3.1.7, Control Bank Insertion Limits, maintain the core limits on power distributions on a continuous basis.

$F_Q(Z)$ varies with fuel loading patterns, control bank insertion, fuel burnup, and changes in axial power distribution.

$F_Q(Z)$ is measured periodically using the incore detector system. These measurements are generally taken with the core at or near steady-state conditions.

Using the measured three-dimensional power distributions, it is possible to derive a measured value for $F_Q(Z)$. However, since this value represents a steady-state condition, it will not include the variations in the value of $F_Q(Z)$ which would be present during nonequilibrium situations, such as load following.

To account for these possible variations, the steady-state value of $F_Q(Z)$ is adjusted by an elevation-dependent factor, which accounts for the calculated worst-case transient conditions.

(continued)

(continued)

BASES (continued)

BACKGROUND (continued)

Core monitoring and control under non-steady-state conditions are accomplished by operating the core within the limits of the appropriate LCOs, including the limits on AFD, QUADRANT POWER TILT RATIO, and control rod insertion.

APPLICABLE SAFETY ANALYSES

This LCO precludes core power distributions from occurring which would violate the following fuel design criteria:

a. During a large-break loss-of-coolant accident (LOCA), the peak cladding temperature must not exceed a limit of 2200°F (10 CFR 50.46) (Ref. 1);

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

c. During an ejected rod accident, the fission energy input to the fuel must not exceed 280 cal/gm (Ref. 2); and

d. The control rods must be capable of shutting down the reactor with a minimum required SHUTDOWN MARGIN with the highest worth control rod stuck fully withdrawn (GDC 26) (Ref. 3).

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

$F_Q(Z)$ limits assumed in the LOCA analysis are typically limiting (i.e., lower) relative to the $F_Q(Z)$ assumed in safety analyses for other postulated accidents. Therefore, this LCO provides conservative limits for other postulated accidents.

$F_Q(Z)$ satisfies Criterion 2 of the NRC Interim Policy Statement.

(continued)

BASES (continued)

LCO

The Heat Flux Hot Channel Factor, $F_Q(Z)$, shall be limited by the following relationships:

$$F_Q(Z) \leq \frac{CFQ}{P} K(Z) \qquad \text{for } P > 0.5$$

$$F_Q(Z) \leq \frac{CFQ}{0.5} K(Z) \qquad \text{for } P \leq 0.5$$

where: CFQ is the $F_Q(Z)$ limit at RATED THERMAL POWER (RTP) provided in the CORE OPERATING LIMITS REPORT (COLR),

K(Z) is the normalized $F_Q(Z)$ as a function of core height provided in the COLR, and

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

For this facility, the actual values of CFQ and K(Z) are given in the COLR; however, CFQ is normally a number on the order of [2.32], and K(Z) is a function that looks like the one provided in Figure B.3.2.1B-1.

For RAOC operation, $F_Q(Z)$ is approximated by $F_Q^C(Z)$ and $F_Q^W(Z)$. Thus, both $F_Q^C(Z)$ and $F_Q^W(Z)$ must meet the above limits on $F_Q(Z)$.

An $F_Q^C(Z)$ evaluation requires obtaining an incore flux map in MODE 1. From the incore flux map results we obtain the measured value of $F_Q(Z)$, called $F_Q^M(Z)$. We then have:

$$F_Q^C(Z) = F_Q^M(Z) * [1.0815]$$

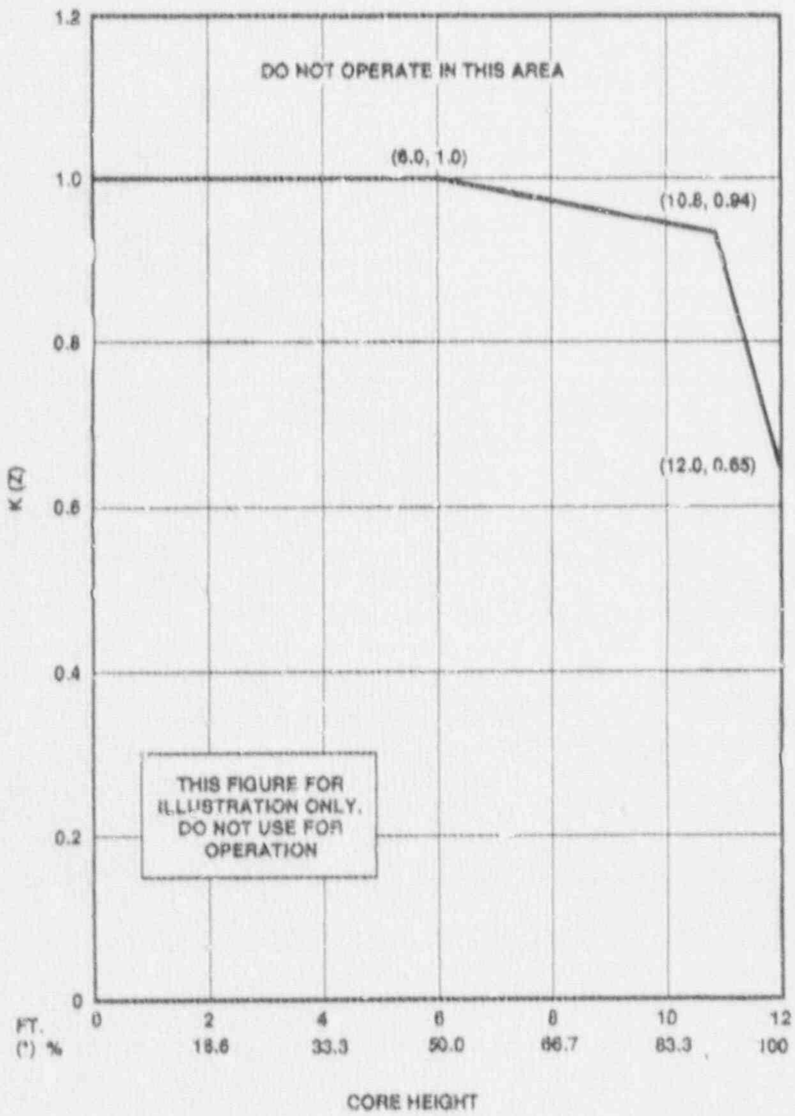
where [1.0815] is a factor that accounts for fuel manufacturing tolerances and flux map measurement uncertainty.

$F_Q^C(Z)$ is an excellent approximation for $F_Q(Z)$ when the reactor is at the steady-state power at which the incore flux map was taken.

(continued)

(continued)

BASES (continued)



* For core height of 12 feet

Figure B 3.2.1B-1 (Page 1 of 1)

K(Z) - Normalized $F_Q(Z)$ as a Function of Core Height

(continued)

BASES (continued)

LCO (continued)

To obtain $F_Q^W(Z)$ we multiply:

$F_Q^W(Z) = F_Q^C(Z) * W(Z)$

where W(Z) is a cycle-dependent function that accounts for power distribution transients encountered during normal operation. W(Z) is included in the COLR.

The $F_Q(Z)$ limits define limiting values for core power peaking that precludes peak cladding temperatures above 2200°F during either a large- or small-break LOCA.

This LCO requires operation within the bounds assumed in the safety analyses. Calculations are performed in the core design process to confirm that the core can be controlled in such a manner during operation that it can stay within the LOCA $F_Q(Z)$ limits. If $F_Q(Z)$ cannot be maintained within the LCO limits, reduction of the core power is required.

Violating the LCO limits for $F_Q(Z)$ could produce unacceptable consequences should a design basis event occur while $F_Q(Z)$ is outside its specified limits.

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

[For this facility, the following support systems are required OPERABLE to ensure [Incore Detector System] OPERABILITY:]

[For this facility, those required support systems that upon their failure do not declare the [Incore Detector System] inoperable and their justification are as follows:]

APPLICABILITY

The $F_Q(Z)$ limits must be maintained in MODE 1 to preclude core power distributions from exceeding the limits assumed in the safety analyses. Applicability in other MODES is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the reactor coolant to require a limit on the distribution of core power.

(continued)

BASES (continued)

ACTIONS

A.1

Reducing THERMAL POWER by at least 1% RTP for each 1% by which $F_Q^C(Z)$ exceeds its limit, maintains an acceptable absolute power density. [For this facility, THERMAL POWER is reduced by the following actions:] $F_Q^C(Z)$ is F_Q^M multiplied by a factor accounting for manufacturing tolerances and measurement uncertainties. $F_Q^M(Z)$ measured value of $F_Q(Z)$. The Completion Time of 15 minutes provides an acceptable time to reduce power in an orderly manner and without allowing the plant to remain in an unacceptable condition for an extended period of time.

A.2

A reduction of the Power Range Neutron Flux—High Trip setpoints by at least 1% for each 1% by which $F_Q^C(Z)$ exceeds its limit is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 8 hours is sufficient considering the small likelihood of a severe transient in this time period and the preceding prompt reduction in THERMAL POWER in accordance with Required Action A.1.

A.3

Reduction in the Overpower ΔT Trip setpoints by 1% for each 1% by which $F_Q^C(Z)$ exceeds its limit is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 72 hours is sufficient considering the small likelihood of a severe transient in this time period and the preceding prompt reduction in THERMAL POWER in accordance with Required Action A.1.

A.4

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

(continued)

(continued)

BASES (continued)

ACTIONS (continued)

B.1

If it is found that the maximum calculated value of $F_Q(Z)$ which can occur during normal maneuvers, $F_Q^W(Z)$, exceeds its specified limits, there exists a potential for $F_Q^C(Z)$ to become excessively high if a normal operational transient occurs. Reducing the AFD by at least 1% for each 1% by which $F_Q^W(Z)$ exceeds its limit within the allowed Completion Time of 2 hours restricts the axial flux distribution such that even if a transient occurred, core peaking factors would not be exceeded.

C.1

If Required Actions A.1 through A.4 or B.1 are not met within their associated Completion Times, or if $F_Q^C(Z)$ or $F_Q^W(Z)$ or both cannot be determined because of incore detector system inoperability, the plant must be placed in a mode or condition in which the LCO requirements are not applicable. This is done by placing the plant in at least MODE 2 within 6 hours.

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

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1

SR 3.2.1.1 is modified by two Notes. The first Note applies during the first power ascension after a refueling. It states that THERMAL POWER may be increased until a power level for extended operation has been achieved and a power distribution map has been obtained. This allowance is modified, however, by one of the Frequency conditions for SR 3.2.1.1 that requires verification that $F_Q^C(Z)$ is within its specified limits after a power rise of more than 10% of RTP over the THERMAL POWER at which $F_Q^C(Z)$ was last verified to be within its specified limits. Since $F_Q^C(Z)$ could not have previously been measured in this reload core, there is a second Frequency condition applicable for reload cores only that allows a first power ascension after a refueling

(continued)

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

up to 75% of RTP before the first determination of $F_Q^C(Z)$ is required. [For this facility, this is acceptable because:]

The second Note states that the provisions of SR 3.0.4 do not apply because the plant must be in MODE 1 before the surveillance can be performed.

Verification that $F_Q^C(Z)$ is within its specified limits involves increasing the measured values of $F_Q^C(Z)$ (i.e., $F_Q^M(Z)$) to allow for manufacturing tolerance and measurement uncertainties in order to obtain $F_Q^C(Z)$. Specifically, $F_Q^M(Z)$ = the measured value of $F_Q(Z)$ obtained from incore flux map results and $F_Q^C(Z) = F_Q^M(Z)$ * [1.0815] (Ref. 4). $F_Q^C(Z)$ is then compared to its specified limits.

The limit to which $F_Q^C(Z)$ is compared varies inversely with power and directly with a function called K(Z) provided in the COLR.

[For this facility, the surveillance Frequency of 31 effective full power days (EFPDs) is adequate to monitor the change of power distribution with core burnup because:]

[For this facility, performing this surveillance prior to exceeding 75% of RTP assures that the $F_Q^C(Z)$ limit will be met when RTP is achieved because:]

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

SR 3.2.1.2

The nuclear design process includes calculations which are performed to determine that the core can be operated within the $F_Q(Z)$ limits. Since flux maps are taken in steady-state conditions, the variations in power distribution resulting from normal operational maneuvers are not present in the flux map data. These variations are, however, conservatively calculated by considering a wide range of unit maneuvers in normal operation. The maximum peaking factor increase over steady-state values, calculated as a

(continued)

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

function of core elevation, Z, is called W(Z). Multiplying the measured total peaking factor, $F_Q^C(Z)$, by W(Z) gives the maximum $F_Q(Z)$ calculated to occur in normal operation, $F_Q^W(Z)$.

The limit to which $F_Q^W(Z)$ is compared varies inversely with power and directly with a function called K(Z) provided in the COLR.

The W(Z) curve is provided in the COLR for discrete core elevations. Flux map data will typically be taken for 30 to 75 core elevations. $F_Q^W(Z)$ evaluations are not applicable for the following axial core regions, measured in percent of core height:

a. Lower core region, from 0% to 15% inclusive; and

b. Upper core region, from 85% to 100% inclusive.

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

This Surveillance has been modified by Note 1, which may require that more frequent surveillances be performed. If $F_Q^W(Z)$ is evaluated and found to be within its limit, an evaluation of the expression below is required to account for any increase to $F_Q^M(Z)$ which could occur and cause the $F_Q(Z)$ limit to be exceeded before the next required $F_Q(Z)$ evaluation.

If the two most recent $F_Q(Z)$ evaluations show an increase in the expression

$$\underset{\text{over } Z}{\text{maximum}} \left[\frac{F_Q^C(Z)}{K(Z)} \right]$$

it is required to meet the $F_Q(Z)$ limit with the last $F_Q^W(Z)$ increased by a factor of [1.0815] or to evaluate $F_Q(Z)$ more frequently, each 7 EFPD. These alternative requirements

(continued)

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

will prevent $F_Q(Z)$ from exceeding its limit for any significant period of time without detection.

Based upon calculations, it is not expected that the $F^C_Q(Z)$ which is closest to the limit will increase by more than [ ]% in 31 EFPDs.

The surveillance Frequency of 31 EFPDs is adequate to monitor the change of power distribution with core burnup. The surveillance may be done more frequently if required by the results of $F_Q(Z)$ evaluations.

[For this facility, the surveillance Frequency of 31 EFPDs is adequate to monitor the change of power distribution because:]

$F_Q(Z)$ is verified at power levels greater than 10% of RTP above the THERMAL POWER of its last verification after achieving equilibrium conditions to assure that $F_Q(Z)$ will be within its limit at higher power levels.

[For this facility, performing the surveillance prior to exceeding 75% of RTP assures that the $F_Q(Z)$ limit will be met when RTP is achieved, because:]

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors," 1974.

2. Regulatory Guide 1.77, Rev. [ ], "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors," [date].

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

4. [WCAP-7308-L-P-A, Evaluation of Nuclear Hot Channel Factor Uncertainties, June 1988.]

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.2 Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)

BASES

BACKGROUND

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

$F^N_{\Delta H}$ is defined as the ratio of the integral of the linear power along the fuel rod with the highest integrated power to the average integrated fuel rod power. Therefore, $F^N_{\Delta H}$ is a measure of the maximum total power produced in a fuel rod.

$F^N_{\Delta H}$ is sensitive to fuel loading patterns, bank insertion and fuel burnup. $F^N_{\Delta H}$ typically increases with control bank insertion and typically decreases with fuel burnup.

$F^N_{\Delta H}$ is not directly measurable but is inferred from a power distribution map obtained with the movable incore detector system. Specifically, the results of the three-dimensional power distribution map is analyzed by a computer to determine $F^N_{\Delta H}$. This factor is calculated at least every 31 effective full power days (EFPDs). However, during power operation, the global power distribution is monitored by LCO 3.2.3, AXIAL FLUX DIFFERENCE (AFD), and LCO 3.2.4, QUADRANT POWER TILT RATIO (QPTR), which are directly and continuously measured process variables.

The CORE OPERATING LIMIT REPORT (COLR) provides peaking factor limits that ensure that the design basis value of the Departure from Nucleate Boiling (DNB) is met for normal operation, operational transients, and any transient condition arising from events of moderate frequency. The DNB design basis precludes DNB and is met by limiting the

(continued)

(continued)

BASES (continued)

BACKGROUND (continued)

minimum local DNB heat flux ratio to 1.3. All transient events that may be DNB limited are assumed to begin with a $F^N_{\Delta H}$ that satisfies the LCO requirements.

Operation outside the LCO limits could produce unacceptable consequences should a DNB limiting event occur. The DNB design basis ensures that there will be no overheating of the fuel, which may result in possible cladding perforation with the release of fission products to the reactor coolant.

APPLICABLE SAFETY ANALYSES

Limits on $F^N_{\Delta H}$ prevent core power distributions from occurring which would exceed the following fuel design limits:

a. There must be at least a 95% probability at a 95% confidence level that the hottest fuel rod in the core does not experience a DNB condition, hereafter referred to as the 95/95 DNB criterion;

b. During a large-break loss-of-coolant accident (LOCA), the peak cladding temperature (PCT) must not exceed a limit of 2200°F;

c. During an ejected rod accident, the fission energy input to the fuel must not exceed 280 cal/gm [Ref. 1]; and

d. The control rods must be capable of shutting down the reactor with a minimum required SHUTDOWN MARGIN with the highest worth control rod stuck fully withdrawn [GDC 26) (Ref. 2].

For transients that may be DNB limited, the Reactor Coolant System flow and the $F^N_{\Delta H}$ are the core parameters of most importance. The limits on $F^N_{\Delta H}$ ensure that the DNB design basis is met for normal operation, operational transients, and any transients arising from events of moderate frequency. The DNB design basis is met by limiting the minimum DNBR to the 95/95 DNB criterion of 1.3. This value provides a high degree of assurance that the hottest fuel rod in the core will not experience a DNB.

(continued)

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

The allowable $F^N_{\Delta H}$ limit increases with decreasing power level. This functionality in $F^N_{\Delta H}$ is included in the analyses that provide the Reactor Core Safety Limits (SLs) of SL 2.1.1. Therefore, any DNB events in which the calculation of the core limits is modeled implicitly use this variable value of $F^N_{\Delta H}$ in the analyses. Likewise, all transients that may be DNB limited are assumed to begin with an initial $F^N_{\Delta H}$ as a function of power level defined by the COLR limit equation.

The LOCA safety analysis indirectly models $F^N_{\Delta H}$ as an input parameter. The Nuclear Heat Flux Hot Channel Factor ($F_Q(Z)$) and the axial peaking factors are inserted directly into the LOCA safety analyses that verify the acceptability of the resulting peak cladding temperature [Ref. 3].

The fuel is protected in part by Technical Specifications, which provide assurance that the initial conditions assumed in the safety and accident analyses remain valid. The following LCOs assure this: LCO 3.2.3, AXIAL FLUX DIFFERENCE (AFD), LCO 3.2.4, QUADRANT POWER TILT RATIO (QPTR), LCO 3.1.7, Control Bank Insertion Limits, LCO 3.2.2, Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$), and LCO 3.2.1, Heat Flux Hot Channel Factor ($F_Q(Z)$).

$F^N_{\Delta H}$ and $F_Q(Z)$ are measured periodically using the movable incore detector system. Measurements are generally taken with the core at, or near, steady-state conditions. Core monitoring and control under transient conditions (Condition 1 events) are accomplished by operating the core within the limits of the LCOs on AXIAL FLUX DIFFERENCE, QUADRANT POWER TILT RATIO, and Bank Insertion Limits.

$F^N_{\Delta H}$ satisfies Criterion 2 of the NRC Interim Policy Statement.

LCO

$F^N_{\Delta H}$ shall be maintained within the limits of the relationship provided in the COLR.

The $F^N_{\Delta H}$ limit identifies the coolant flow channel with the maximum enthalpy rise. This channel has the least heat removal capability and thus the highest probability for a DNB.

(continued)

(continued)

BASES (continued)

LCO (continued)

The limiting value of $F^N_{\Delta H}$, described by the equation contained in the COLR, is the design radial peaking factor used in the unit safety analyses.

A power multiplication factor in this equation includes an additional margin for higher radial peaking from reduced thermal feedback and greater control rod insertion at low power levels. The limiting value of $F^N_{\Delta H}$ is allowed to increase 0.3% for every 1% of RATED THERMAL POWER (RTP) reduction in THERMAL POWER.

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

[For this facility, the following support systems are required OPERABLE to ensure [Incore Detector System] OPERABILITY:]

[For this facility, those required support systems that upon their failure do not declare the [Incore Detector System] inoperable and their justification are as follows:]

APPLICABILITY

The $F^N_{\Delta H}$ limits must be maintained in MODE 1 to preclude core power distributions from exceeding the fuel design limits for DNBR and PCT. Applicability in other modes is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the coolant to require a limit on the distribution of core power. Specifically, the design bases events that are sensitive to $F^N_{\Delta H}$ in other modes (MODES 2 through 5) have significant margin to DNB, and therefore, there is no need to restrict $F^N_{\Delta H}$ in these modes.

ACTIONS

A.1.1

With $F^N_{\Delta H}$ exceeding its limit, the unit is allowed 4 hours to restore $F^N_{\Delta H}$ to within its limits. This restoration could, for example, involve realigning any misaligned rods or reducing power enough to bring $F^N_{\Delta H}$ within its power-

(continued)

(continued)

BASES (continued)

ACTIONS (continued)

dependent limit. When the $F^N_{\Delta H}$ limits are exceeded, the DNBR limit would likely not be violated in steady-state operation, since events that could significantly perturb the $F^N_{\Delta H}$ value (e.g., static control rod misalignment) are considered in the safety analyses. However, the DNBR limit may be violated if a DNB limiting event were to occur. Thus, the allowed Completion Time of 4 hours provides an acceptable time to restore $F^N_{\Delta H}$ to within its limits without allowing the plant to remain in an unacceptable condition for an extended period of time.

[For this facility, the 4-hour Completion Time is acceptable because:]

Condition A is modified by a Note that requires that Required Actions A.2 and A.3 must be completed whenever Condition A is entered. Thus, if power were not reduced because this Required Action was completed within the 4-hour time period, Required Action A.2 would nevertheless require another measurement and calculation of $F^N_{\Delta H}$ within 24 hours in accordance with SR 3.2.2.1.

However, if power were reduced below 50% of RTP, Required Action A.3 would require that another determination of $F^N_{\Delta H}$ would be required prior to exceeding 50% of RTP, prior to exceeding 75% of RTP, and within 24 hours after reaching or exceeding 95% of RTP. In addition, Required Action A.2 would be performed if power ascension were delayed past 24 hours.

<u>A.1.2.1 and A.1.2.2</u>

If the value of $F^N_{\Delta H}$ is not restored to within its specified limit either by adjusting a misaligned rod or by reducing THERMAL POWER, the alternative option is to reduce THERMAL POWER to < 50% of RTP in accordance with Required Action A.1.2.1 and reduce the Power Range Neutron Flux—High ≤ 55% of RTP in accordance with Required Action A.1.2.2. The allowed Completion Time of 4 hours for Required Action A.1.2.1 is consistent with those allowed for in Required Action A.1.1 and provides an acceptable time to reach the required power level from full power operation without allowing the plant to remain in an unacceptable condition for an extended period of time because [ ]. The

(continued)

(continued)

BASES (continued)

ACTIONS (continued)

Completion Time of 4 hours for Required Actions A.1.1 and A.2.2.1 are not additive.

The allowed Completion Time of 8 hours to reset the trip setpoints per Required Action A.1.2.2 recognizes that, once power is reduced, the safety analysis assumptions are satisfied and there is no urgent need to reduce the trip setpoints. This is a sensitive operation which has a potential to inadvertently trip the Reactor Protection System.

A.2

Once the power level has been reduced to < 50% of RTP per Required Action A.1.1, an incore flux map must be obtained and the measured value of $F^N_{\Delta H}$ verified not to exceed the allowed limit at the lower power level. The unit is provided 20 additional hours to perform this task over and above the 4 hours allowed by either Action A.1.1 or Action A.1.2.1. The Completion Time of 24 hours is acceptable because of the increase in the DNB margin, which is obtained at lower power levels, and the low probability of having a DNB limiting event within this 24-hour period. Additionally, operating experience has indicated that this Completion Time is sufficient to obtain the incore flux map, perform the required calculations, and evaluate $F^N_{\Delta H}$.

A.3

Verification that $F^N_{\Delta H}$ is within its specified limits after an out-of-limit occurrence assures that the cause that led to the $F^N_{\Delta H}$ exceeding its limit has been corrected, and subsequent operation will proceed within the LCO limit. This demonstrates that the $F^N_{\Delta H}$ limit is within the LCO limits prior to exceeding 50% of RTP and again prior to exceeding 75% of RTP, and within 24 hours after THERMAL POWER is ≥ 95% of RTP.

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

(continued)

(continued)

BASES (continued)

ACTIONS (continued)

B.1

When Required Actions A.1.1 through A.3 cannot be completed within their required Completion Times, or if $F^N_{\Delta H}$ cannot be determined because of incore detector system inoperability, the plant must be placed in a mode in which the LCO requirements are not applicable. This is done by placing the plant in at least MODE 2 within 6 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience regarding the time required to reach MODE 2 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1

The value of $F^N_{\Delta H}$ is determined by using the movable incore detector system to obtain a flux distribution map. A data reduction computer program then calculates the maximum value of $F^N_{\Delta H}$ from the measured flux distributions. The measured value of $F^N_{\Delta H}$ must be multiplied by 1.04 to account for measurement uncertainty before making comparisons to the $F^N_{\Delta H}$ limit.

The 31 EFPDs surveillance Frequency is acceptable because the power distribution will change relatively slowly over this amount of fuel burnup. Accordingly, this Frequency is short enough so that the $F^N_{\Delta H}$ limit will not be exceeded for any significant period of operation.

After each refueling, $F^N_{\Delta H}$ must be determined prior to exceeding 75% RTP. This requirement ensures that $F^N_{\Delta H}$ limits are met at the beginning of each fuel cycle.

This surveillance is modified by a Note, which states that the requirements of SR 3.0.4 do not apply since the unit must be in MODE 1 to perform surveillances that demonstrate that the LCO is met.

(continued)

BASES (continued)

REFERENCES

1. Regulatory Guide 1.77, Rev [ ], "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors," [date].

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

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

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3A AXIAL FLUX DIFFERENCE (AFD) (Constant Axial Offset Control (CAOC) Methodology)

BASES

BACKGROUND

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

The operating scheme used to control the axial power distribution, called Constant Axial Offset Control (CAOC), involves maintaining the AFD within a tolerance band around a burnup-dependent target, known as the target flux difference, to minimize the variation of the axial peaking factor and axial xenon distribution during unit maneuvers.

The target flux difference is determined at equilibrium xenon conditions. The control banks must be positioned within the core in accordance with their insertion limits and Control Bank D should be inserted near its normal position (i.e., $\geq$ 210 steps withdrawn) for steady-state operation at high power levels. The power level should be as near RATED THERMAL POWER (RTP) as practical. The value of the target flux difference obtained under these conditions divided by the fraction of RTP is the target flux difference at RTP for the associated core burnup conditions. Target flux differences for other THERMAL POWER levels are obtained by multiplying the RTP value by the appropriate fractional THERMAL POWER level.

Periodic updating of the target flux difference value is necessary to follow the change of the flux difference at steady-state conditions with burnup.

The Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$) and QUADRANT POWER TILT RATIO (QPTR) LCOs limit the radial component of the peaking factors.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The AFD is a measure of axial power distribution skewing to the top or bottom half of the core. The AFD is very sensitive to many core-related parameters such as control bank positions, core power level, axial burnup, axial xenon distribution and, to a lesser extent, reactor coolant temperature and boron concentrations. The allowed range of the AFD is used in the nuclear design process to confirm that operation within these limits produces core peaking factors and axial power distributions that meet safety analysis requirements.

The CAOC methodology (Refs. 1, 2, and 3) entails:

a. Establishing an envelope of allowed power shapes and power densities;

b. Devising an operating strategy for the cycle which maximizes unit flexibility (maneuvering) and minimizes axial power shape changes;

c. Demonstrating that this strategy will not result in core conditions that violate the envelope of permissible core power characteristics; and

d. Demonstrating that this power distribution control scheme can be effectively supervised with excore detectors.

The limits on the AFD assure that the Heat Flux Hot Channel Factor ($F_Q(Z)$) is not exceeded during either normal operation or in the event of xenon redistribution following power changes. The limits on the AFD also limit the range of power distributions which are assumed as initial conditions in analyzing Conditions 2, 3, and 4 events. This ensures that fuel-cladding integrity is maintained for these postulated accidents. The most important Condition 4 event is the loss-of-coolant accident. The most significant Condition 3 event is the loss-of-flow accident. The most significant Condition 2 events are uncontrolled bank withdrawal and boration or dilution accidents. Condition 2 accidents, assumed to begin from within the AFD limits, are used to confirm the adequacy of Overpower ΔT and Overtemperature ΔT trip setpoints.

(continued)

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)	The limits on the AFD satisfy Criterion 2 of the NRC Interim Policy Statement.

LCO

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

Signals are available to the operator from the Nuclear Instrumentation System (NIS) excore neutron detectors (Ref. 4). Separate signals are taken from the top and bottom detectors. The AFD is defined as the difference in normalized flux signals between the top and bottom excore detector in each detector well. For convenience, this flux difference is converted to provide flux difference units expressed as a percentage and labeled as %Δ-flux or %ΔI.

Part A of this LCO is modified by a Note, which states the conditions necessary to declare the AFD outside of the target band. The required target band varies with axial burnup distribution which in turn varies with the core average accumulated burnup. The target band defined in the CORE OPERATING LIMITS REPORT (COLR) may provide one target band for the entire cycle or more than one band, each to be allowed for a specific range of cycle burnup.

With THERMAL POWER ≥ 90% of RTP, the AFD must be kept within the target band. With the AFD outside the target band with THERMAL POWER ≥ 90% of RTP, the assumptions of the accident analyses may be violated.

Parts B and C of this LCO are modified by Notes, which describe how the cumulative penalty deviation time is calculated. It is intended that the unit will be operated with the AFD within the target band about the target flux difference. However, during rapid THERMAL POWER reductions, control bank motion may cause the AFD to deviate outside of the target band at reduced THERMAL POWER levels. This

(continued)

(continued)

BASES (continued)

LCO (continued)

deviation will not affect the xenon distribution sufficiently to change the envelope of peaking factors which may be reached on a subsequent return to RTP with the AFD within the target band, provided the time duration of the deviation is limited. Accordingly, while THERMAL POWER is ≥ 50% of RTP and < 90% of RTP (i.e., Part B of this LCO), a 1-hour cumulative penalty deviation time limit, cumulative during the preceding 24 hours, is allowed during which the unit may be operated outside of the target band but within the acceptable operation limits provided in the COLR. This penalty time is accumulated at the rate of 1 minute for each minute of operating time within the power range of Part B of this LCO (i.e., power > 50% of RTP but < 90% of RTP). The cumulative penalty time is the sum of penalty times from Part B and Part C of this LCO.

For THERMAL POWER levels > 15% of RTP and < 50% of RTP (i.e., Part C of this LCO), deviations of the AFD outside of the target band are less significant. The accumulation of 1/2-minute penalty deviation time per 1 minute of actual time outside the target band reflects this reduced significance. With THERMAL POWER < 15% of RTP, AFD is not a significant parameter in the assumptions used in the safety analysis and, therefore, requires no limits. Because the xenon distribution produced at THERMAL POWER levels less than RTP does affect the power distribution as power is increased, unanalyzed xenon and power distribution is prevented by limiting the accumulated penalty deviation time.

The frequency of monitoring the AFD by the unit computer is once per minute providing an essentially continuous accumulation of penalty deviation time which allows the operator to accurately assess the status of the penalty deviation time.

Violating the LCO on the AFD could produce unacceptable consequences should a Condition 2, 3, or 4 event occur while the AFD is outside the limits.

Figure B 3.2.3A-1 shows a typical target band and typical AFD acceptable operation limits.

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

(continued)

(continued)

BASES (continued)

LCO (continued)

[For this facility, the following support systems are required OPERABLE to ensure [Excore Detector System] OPERABILITY:]

[For this facility, those required support systems that upon their failure do not declare the [Excore Detector System] inoperable and their justification are as follows:]

APPLICABILITY

AFD requirements are applicable in MODE 1 above 15% of RTP. Above 50% of RTP, the combination of THERMAL POWER and core peaking factors are the core parameters of primary importance in safety analyses (Ref. 1).

Between 15% and 90% of RTP, this LCO is applicable to assure that the distributions of xenon are consistent with safety analysis assumptions.

At or below 15% of RTP and for lower operating MODES, the stored energy in the fuel and the energy being transferred to the reactor coolant is low. The value of the AFD in these conditions does not affect the consequences of the design basis events.

For surveillance testing of the power range channels performed according to SR 3.3.1.6, deviation outside the target band is permitted for 16 hours and no penalty deviation time is accumulated. Some deviation in the AFD is required for doing the NIS calibration with the incore detector system. This calibration is performed every 92 days.

Low signal levels in the excore channels may preclude obtaining valid AFD signals below 15% of RTP.

ACTIONS

A.1

With the AFD outside the target band and THERMAL POWER ≥ 90% of RTP, the assumptions used in the accident analyses may be

(continued)

(continued)

BASES (continued)

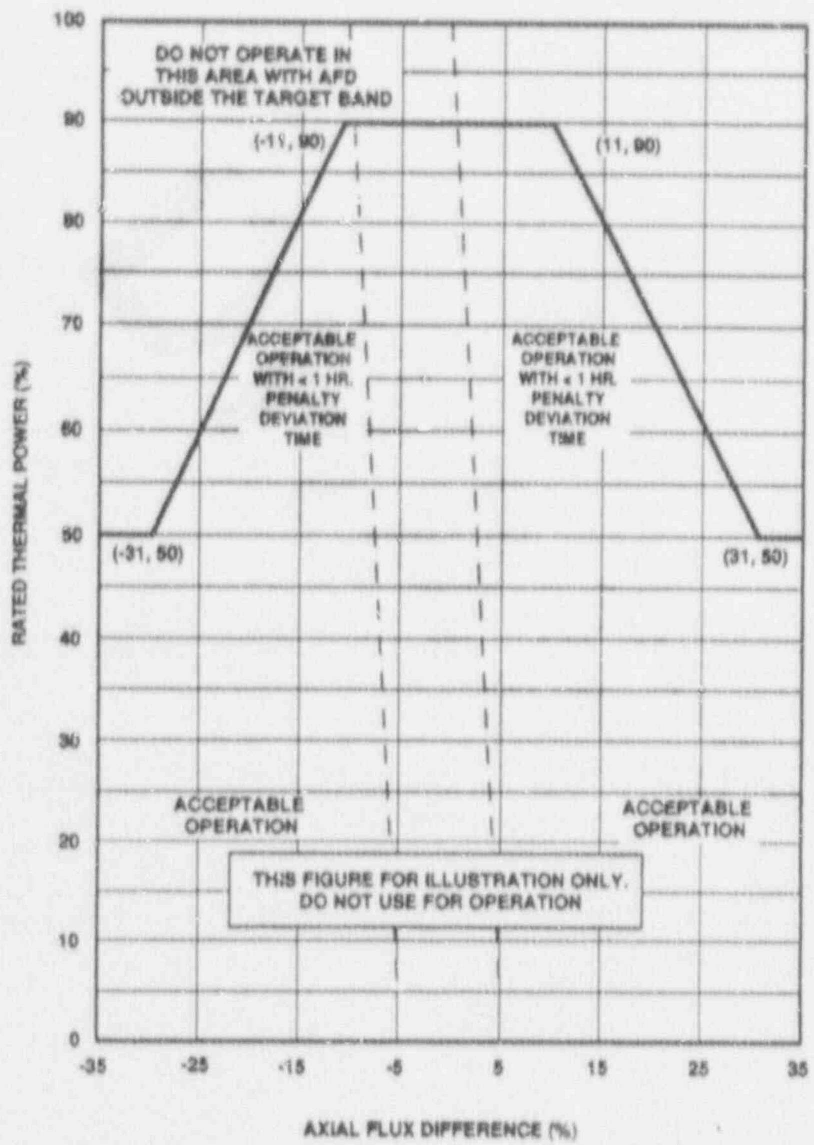


Figure B 3.2.3A-1 (Page 1 of 1)

AXIAL FLUX DIFFERENCE Acceptable Operation Limits
and Target Band Limits as a Function
of RATED THERMAL POWER

(continued)

BASES (continued)

ACTIONS (continued)

violated with respect to the maximum heat generation. Therefore, a Completion Time of 15 minutes is allowed to restore the AFD to within the target band because xenon distributions will change very little in this relatively short time.

A.2

If the AFD cannot be restored within the target band, then reducing THERMAL POWER to < 90% of RTP places the core in a condition which has been analyzed and found to be acceptable, provided that the AFD is within the acceptable operation limits provided in the COLR.

The allowed Completion Time of 15 minutes provides an acceptable time to either restore the AFD within its specified limits or reduce power to < 90% of RTP without allowing the plant to remain in an unanalyzed condition for an extended period of time.

B.1

If either Required Action A.1 or A.2 is not completed within their required Completion Times of 15 minutes, the axial xenon distribution will start to become skewed. In this situation, the assumption that, when the AFD is outside its target band for less than 1 hour with THERMAL POWER < 90% of RTP but ≥ 50% of RTP, this deviation will not significantly affect the xenon distribution, is no longer valid. Reducing the power level to < 50% of RTP within the Completion Time of 15 minutes and compliance with LCO requirements for subsequent increases in THERMAL POWER will assure that acceptable xenon distributions are restored.

The Completion Time of 15 minutes is acceptable because the xenon distributions will change very little in this relatively short time. In addition, Required Action B.2 is to reduce Power Range Neutron Flux—High trip setpoints to ≤ 55% of RTP within a Completion Time of 8 hours. [For this facility, this is an acceptable Required Action and Completion Time because:]

The second part of Condition B concerns excore detector system inoperability. If the AFD in Condition A cannot be determined because of excore detector system inoperability,

(continued)

(continued)

BASES (continued)

ACTIONS (continued)

then taking the Required Actions B.1 and B.2 is a conservative action because it amounts to taking the same actions one would have taken if the excore detector system was OPERABLE and it showed that the AFD was not within the target band.

C.1

With THERMAL POWER < 90% of RTP but ≥ 50% of RTP, operation with the AFD outside the target band is allowed for up to 1 hour if the AFD is within the acceptable operation limits provided in the COLR. With the AFD within these limits, the resulting axial power distribution will be acceptable as an initial condition for accident analyses assuming the then-existing xenon distributions. The 1-hour cumulative penalty deviation time restricts the extent of xenon redistribution which can occur. Without this limitation, unanalyzed xenon axial distributions could result from a different pattern of xenon buildup and decay. The reduction to a power level < 50% of RTP puts the reactor at a THERMAL POWER level where the AFD is not a significant accident analysis parameter.

If the indicated AFD is outside the target band and outside the acceptable operation limits provided in the COLR, the peaking factors assumed in accident analysis may be exceeded with the existing xenon condition. (Note that any AFD within the target band is acceptable regardless of its relationship to the acceptable operation limits.) The Completion Time of 30 minutes allows for a prompt, yet orderly reduction in power.

Condition C is modified by a Note which requires that Required Actions C.1 and C.2 must be completed whenever this Condition is entered.

C.2

If Required Action B.1 is not completed within its required Completion Time of 15 minutes, the axial xenon distribution will start to become significantly skewed with the THERMAL POWER ≥ 50% of RTP. In this situation, the assumption that a cumulative penalty deviation time of 1 hour or less during the previous 24 hours while the AFD is outside its target band is acceptable at < 50% of RTP, is no longer valid.

(continued)

(continued)

BASES (continued)

ACTIONS (continued)

Reducing the power level to < 15% of RTP within the Completion Time of 9 hours and compliance with LCO penalty deviation time requirements for subsequent increases in THERMAL POWER will assure that acceptable xenon conditions are restored.

This Required Action must also be implemented either if the cumulative penalty deviation time is > 1 hour during the previous 24 hours, or the AFD is not within the target band and not within the acceptable operation limits or if the ADF is still not able to be determined because of excore detector system inoperability.

SURVEILLANCE REQUIREMENTS

SR 3.2.3.1

The AFD is monitored on an automatic basis using the unit process computer which has an AFD monitor alarm. The computer determines the 1-minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFDs for two or more OPERABLE excore channels are outside the target band and the THERMAL POWER is > 90% of RTP. During operation at THERMAL POWER levels < 90% of RTP but > 15% of RTP, the computer sends an alarm message when the cumulative penalty deviation time is > 1 hour in the previous 24 hours.

This surveillance verifies that the AFD as indicated by the NIS excore channels is within the target band and consistent with the status of the AFD monitor alarm. The surveillance frequency of 7 days is adequate because the AFD is controlled by the operator and monitored by the process computer. Furthermore, any deviations of the AFD from the target band which is not alarmed should be readily noticed.

SR 3.2.3.2

With the AFD monitor alarm inoperable, the AFD is monitored to detect operation outside of the target band and to compute the penalty deviation time. During operation at ≥ 90% of RTP, the AFD is monitored at a surveillance Frequency of 15 minutes to provide a high level of assurance that the AFD is within its limits at high THERMAL POWER

(continued)

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

levels. At power levels < 90% of RTP, but > 15% of RTP, the surveillance Frequency is reduced to 1 hour since the AFD may deviate from the target band for up to 1 hour using the methodology of Parts B and C of this LCO to calculate the cumulative penalty deviation time before corrective action is required. This Completion Time of 1 hour is acceptable because [ ].

SR 3.2.3.2 is modified by a Note which states that monitored and logged values of the AFD are assumed to exist for the preceding 24-hour interval in order to compute the cumulative penalty deviation time. The AFD should be monitored and logged more frequently in periods of operation where the power level or control bank positions are changing to allow corrective measures when the AFD is more likely to move outside the target band.

SR 3.2.3.3

This surveillance requires that the target flux difference be updated at a Frequency of 31 effective full power days (EFPDs) to account for small changes which may occur in the target flux differences in that period due to burnup by performing SR 3.2.3.4.

Alternatively, linear interpolation between the most recent measurement of the target flux differences and a predicted end-of-cycle value provides a reasonable update since the AFD changes due to burnup will tend toward a zero AFD. Where the predicted end-of-cycle AFD from the cycle nuclear design is different from 0%, this latter value may be a better value for the interpolation because [ ].

SR 3.2.3.3 is modified by a Note which states that the provisions of SR 3.0.4 are not applicable since the unit must be in MODE 1 to perform SR 3.2.3.3.

SR 3.2.3.4

Measurement of the target flux difference is accomplished by taking a flux map when the core is at equilibrium xenon conditions, preferably at high power levels with the control banks nearly withdrawn. This flux map provides the equilibrium xenon axial power distribution from which the

(continued)

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

target value can be determined. The target flux difference will vary slowly with core burnup.

A frequency of 92 EFPD for remeasuring the target flux differences adjusts the target flux difference for each excore channel to the value measured at steady-state conditions. This is the basis for the CAOC. Remeasurement at this surveillance interval also establishes the AFD target flux difference values which will account for changes in incore-excore calibrations which may have occurred in the interim.

SR 3.2.3.4 is modified by a Note which indicates that the provisions of SR 3.0.4 are not applicable since the unit must be in MODE 1 to perform this surveillance.

REFERENCES

1. WCAP-8403 (nonproprietary), Power Distribution Control and Load Following Procedures, Westinghouse Electric Corporation, September 1974.

2. T.M. Anderson to K. Kniel (Chief of Core Performance Branch, NRC.), Attachment: Operation and Safety Analysis Aspects of an Improved Load Follow Package, January 31, 1980.

3. C. Eicheldinger to D.B. Vassallo (Chief of Light Water Reactors Branch, NRC), Letter NS-CE-687, July 16, 1975.

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

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3B AXIAL FLUX DIFFERENCE (AFD) (Relaxed Axial Offset Control (RAOC) Methodology)

BASES

BACKGROUND

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

ROAC is a calculational procedure which defines the allowed operational space of the AFD versus THERMAL POWER. The AFD limits are selected by considering a range of axial xenon distributions which could occur as a result of large variations of the AFD. Subsequently, power peaking factors and power distributions are examined to assure that the loss-of-coolant accident (LOCA), loss-of-flow accident, and anticipated transient limits are met. Violation of the AFD limits would invalidate the conclusions of the accident and transient analyses with regard to fuel-cladding integrity.

Although the RAOC defines limits which must be met to satisfy safety analyses, typically an operating scheme called Constant Axial Offset Control (CAOC) will be used to control axial power distribution in day-to-day operation (Ref. 1). CAOC requires that the AFD be controlled within a narrow tolerance band around a burnup-dependent target to minimize the variation of axial peaking factors and axial xenon distribution during unit maneuvers.

The CAOC operating space is typically smaller and lies within the RAOC operating space. Control within the CAOC operating space constrains the variation of axial xenon distributions and axial power distributions which can occur. RAOC calculations assume a wide range of xenon distributions and then confirm that the resulting power distributions will satisfy the requirements of the accident analyses.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

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

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

The RAOC methodology (Ref. 2) establishes a xenon distribution library with tentative, very wide AFD limits. These limits are labeled tentative because [ ]. One-dimensional axial power distribution calculations are then performed to demonstrate that normal operation power shapes are acceptable for the LOCA, the loss-of-flow accident, and for initial conditions of anticipated transients. The tentative limits are adjusted as necessary to meet the safety analysis requirements.

The limits on the AFD assure that the Heat Flux Hot Channel Factor ($F_Q(Z)$) is not exceeded during either normal operation or in the event of xenon redistribution following power changes. The limits on the AFD also restrict the range of power distributions which are used as initial conditions in the analyses of Condition 2, 3, or 4 events. This ensures that the fuel-cladding integrity is maintained for these postulated accidents. The most important Condition 4 event is the LOCA. The most important Condition 3 event is the loss-of-flow accident. The most important Condition 2 events are uncontrolled bank withdrawal and boration or dilution accidents. Condition 2 accidents simulated to begin from within the AFD limits are used to confirm the adequacy of the Overpower ΔT and Overtemperature ΔT trip setpoints.

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

LCO

The shape of the power profile in the axial (i.e., the vertical) direction is largely under the control of the

(continued)

(continued)

BASES (continued)

LCO (continued)

operator through either the manual operation of the control banks or automatic motion of control banks responding to temperature deviations resulting from either manual operation of the Chemical and Volume Control System to change boron concentration or from power level changes.

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

The AFD limits are provided in the CORE OPERATING LIMITS REPORT. Figure B 3.2.3B-1 shows typical RAOC AFD limits. The AFD limits for RAOC do not depend on the target flux difference. However, the target flux difference may be used to minimize changes in the axial power distribution.

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

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

[For this facility, the following support systems are required OPERABLE to ensure [Excore Detector System] OPERABILITY:]

[For this facility, those required support systems that upon their failure do not declare the [Excore Detector System] inoperable and their justification are as follows:]

APPLICABILITY

The AFD requirements are applicable in MODE 1 above 50% of RATED THERMAL POWER (RTP) where the combination of THERMAL POWER and core peaking factors are of primary importance in safety analysis.

(continued)

(continued)

BASES (continued)

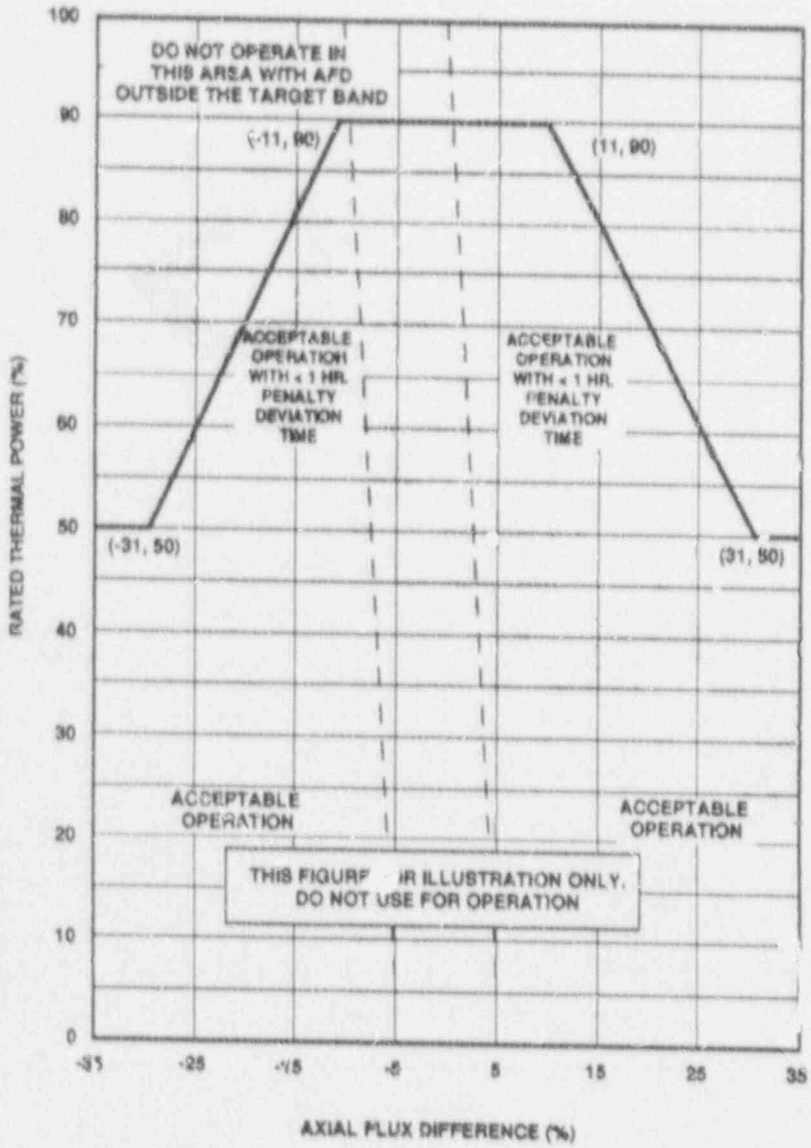


Figure B 3.2.3B-1 (Page 1 of 1)

Axial Flux Difference Limits as a Function of RATED THERMAL POWER

(continued)

(continued)

BASES (continued)

APPLICABILITY (continued)

For AFD limits developed using RAOC methodology, the value of the AFD does not affect the limiting accident consequences with THERMAL POWER < 50% of RTP and for lower operating power MODES.

ACTIONS

A.1

When the AFD exceeds its specified limits, the assumptions used in the accident analyses may not be valid. Therefore, the allowed Completion Time of 15 minutes provides an acceptable time to restore the AFD to within its limits without allowing the plant to remain in an unacceptable condition for an extended period of time.

A.2.1 and A.2.2

As an alternative to restoring the AFD to within its specified limits, Required Action A.2.1 requires a THERMAL POWER reduction to < 50% of RTP. This places the core in a condition where the value of the AFD is not important in the applicable safety analyses. A Completion Time of 30 minutes is reasonable, based on operating experience, to reach 50% of RTP without challenging plant systems.

Required Action A.2.2 requires a reduction in the Power Range Neutron Flux—High trip setpoints to ≤ 55% RTP within a Completion Time of 8 hours. [For this facility, this is an appropriate Required Action and Completion Time because:]

B.1.1 and B.1.2

Condition B is that the AFD cannot be determined because of excore detector system inoperability. Required Action B.1.1 is to reduce THERMAL POWER to < 50% RTP within a Completion Time of 30 minutes. This is a conservative action that is consistent with Required Action A.2.1. Similarly Required Action B.1.2 is to reduce the Power Range Neutron Flux—High trip setpoints to ≤ 55% RTP. This is a conservative action that is consistent with Required Action A.2.2. These are conservative actions because the plant is taken down to a power level < 50% RTP, which is outside the Applicability, in the same time as if the AFD had actually not been within limits (Condition A).

(continued)

(continued)

BASES (continued)

ACTIONS (continued)

Required Actions B.2.1 and B.2.2 are modified by a Note that states that Required Action B.1.2 must be completed if Required Action B.1.1 is completed. [For this facility, this is a requirement because:]

SURVEILLANCE REQUIREMENTS

SR 3.2.3.1

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

This surveillance verifies that the AFD, as indicated by the NIS excore channel, is within its specified limits and consistent with the status of the AFD monitor alarm. With the AFD monitor alarm inoperable, the AFD is monitored every hour to detect operation outside its limit. The Frequency of 1 hour is based on operating experience regarding the amount of time required to vary the AFD, and the fact that the AFD is closely monitored. With the AFD monitor alarm OPERABLE, the surveillance Frequency of 7 days is adequate considering that the AFD is monitored by a computer and any deviation from requirements will be alarmed.

REFERENCES

1. WCAP-8403 (nonproprietary), "Power Distribution Control and Load Following Procedures," Westinghouse Electric Corporation, September 1974.

2. R.W. Miller et al., "Relaxation of Constant Axial Offset Control: F_Q Surveillance Technical Specification," WCAP-10217(NP), June 1983.

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

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 QUADRANT POWER TILT RATIO (QPTR)

BASES

BACKGROUND	The QPTR limit assures that the gross radial power distribution remains consistent with the design values used in the safety analyses. Precise radial power distribution measurements are made during startup testing, after refueling, and periodically during power operation. The QPTR is a defined term in Section 1.1, Definitions. The power density at any point in the core must be limited so that the fuel design criteria are maintained. Together, LCO 3.2.3, AXIAL FLUX DIFFERENCE (AFD), LCO 3.2.4, QUADRANT POWER TILT RATIO (QPTR), and LCO 3.1.7, Control Rod Insertion Limits, provide limits on process variables that characterize and control the three-dimensional power distribution of the reactor core. Control of these variables ensures that the core operates within the fuel design criteria and that the power distribution remains within the bounds used in the safety analyses.
APPLICABLE SAFETY ANALYSES	This LCO precludes core power distributions from occurring which would violate the following fuel design criteria: a. During a large-break loss-of-coolant accident, the peak cladding temperature must not exceed a limit of 2200°F (10 CFR 50.46) (Ref. 1); b. During a loss-of-forced-reactor-coolant-flow accident, there must be at least a 95% probability at a 95% confidence level that the hot fuel rod in the core does not experience a Departure from Nucleate Boiling condition; c. During an ejected rod accident, the fission energy input to the fuel must not exceed 280 cal/gm (Ref. 2); and (continued)

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

d. The control rods must be capable of shutting down the reactor with a minimum required SHUTDOWN MARGIN with the highest worth control rod stuck fully withdrawn (GDC 26) (Ref. 3).

The LCO limits on the AFD, the QPTR, the Heat Flux Hot Channel Factor ($F_Q(Z)$), the Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$), and control bank insertion are established to preclude core power distributions from occurring which would exceed the safety analyses limits.

The QPTR limits assure that $F^N_{\Delta H}$ and $F_Q(Z)$ remain below their limiting values by preventing an undetected change in the gross radial power distribution.

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

The QPTR satisfies Criterion 2 of the NRC Interim Policy Statement.

LCO

The QPTR limit of 1.02, where corrective action is required, provides a margin of protection for both the Departure from Nucleate Boiling Ratio and linear heat generation rate (LHGR) contributing to excessive power peaks resulting from x-y plane power tilts. A limiting QPTR of 1.02 can be tolerated before the margin for uncertainty in $F_Q(Z)$ and ($F^N_{\Delta H}$) is possibly challenged.

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

[For this facility, the following support systems are required OPERABLE to ensure [Excore Detector System] OPERABILITY:]

[For this facility, those required suppor systems that upon their failure do not declare the [Excore tector System] inoperable and their justification are as llows:]

(continued)

BASES (continued)

APPLICABILITY

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

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

ACTIONS

A.1

With the QPTR exceeding its limit, a power level reduction of 3% of RTP for each 1% by which the QPTR exceeds 1.00 is a conservative tradeoff of total core power with peak linear power. The Completion Time of 2 hours allows sufficient time to identify the cause and correct the tilt. Note that the power reduction itself may cause a change in the tilted condition. Since the QPTR alarm would already be in its alarmed state, any additional changes in the QPTR will be detected by requiring a check of the QPTR once per 12 hours thereafter. If the QPTR were to continue to increase, THERMAL POWER would have to be reduced accordingly. [For this facility, this 12-hour Completion Time is acceptable because:]

A.2

Reduction in the Power Range Neutron Flux—High trip setpoints is a conservative action for protection against the consequences of transients with unanalyzed power distributions. The Completion Time of 8 hours is allowed since the reduction in THERMAL POWER is the principal compensation for the tilted condition. Since the QPTR alarm would already be in its alarmed state, additional changes in the QPTR would be detected by requiring a check of the QPTR once per 12 hours thereafter. If the QPTR has increased, the setpoints would have to be reduced accordingly. [For this facility, this 12-hour Completion Time is acceptable because:]

(continued)

(continued)

BASES (continued)

ACTIONS (continued)

A.3

The peaking factors $F^N_{\Delta H}$ and $F_Q(Z)$ are of primary importance in assuring that the power distribution remains consistent with the initial conditions used in the safety analyses. Performing SRs on $F^N_{\Delta H}$ and $F_Q(Z)$ within the Completion Time of 24 hours will assure that these primary indicators of power distribution are within their respective limits. [For this facility, this Completion Time of 24 hours is acceptable because:] If these peaking factors are not within their limits, the Required Actions of these surveillances will provide an appropriate response for the abnormal condition. If the QPTR remains above its specified limits, the peaking factor surveillances are required each 7 days thereafter to evaluate $F^N_{\Delta H}$ and $F_Q(Z)$ with changes in power distribution. Relatively small changes would be expected due to either burnup and xenon redistribution or correction of the cause for exceeding the QPTR limit.

A.4.1

Although $F^N_{\Delta H}$ and $F_Q(Z)$ are of primary importance as initial conditions in the safety analyses, other changes in the power distribution may occur as the QPTR limit is exceeded and may have an impact on the validity of the safety analysis. A change in the power distribution can affect such reactor parameters as bank worths and peaking factors for rod malfunction accidents. When the QPTR exceeds its limit, it does not necessarily mean a safety concern exists. It does mean that there is an indication of a change in the gross radial power distribution which requires an investigation and evaluation which is accomplished by examining the incore power distribution. Specifically, the core peaking factors and the quadrant tilt must be evaluated because they are the factors which best characterize the core power distribution. This reevaluation is required to assure that, prior to either continued operation at RTP, or an increase in THERMAL POWER to RTP, the reactor core conditions are consistent with the assumptions in the safety analyses.

(continued)

(continued)

BASES (continued)

ACTIONS (continued)

A.4.2

If the QPTR has exceeded the 1.02 limit and a reevaluation of the safety analysis is completed and shows that safety requirements will be met, the excore detectors should be recalibrated to show a zero QPTR prior to increasing THERMAL POWER. This is done to detect any subsequent significant changes in QPTR.

Required Action A.4.2 is modified by a Note which states that the quadrant power tilt (QPT) should not be zeroed out until after the reevaluation of the safety analysis has determined that core conditions at RTP will be within the safety analysis assumptions (i.e., Required Action A.4.1). This Note is intended to prevent any ambiguity about the required sequence of actions.

A.4.3

Once the flux tilt has been zeroed out (i.e., Required Action A.4.2), it is acceptable to return to full power operation. However, as an added check that the core power distribution at RTP is consistent with the safety analysis assumptions, Action A.3.3 requires verification that $F_Q(Z)$ and $F^N_{\Delta H}$ are within their specified limits within 24 hours of reaching RTP. As an added precaution, if the core power does not reach RTP within 24 hours, but is increased slowly, then the peaking factor surveillances must be performed within 48 hours of the time when the ascent to power was begun. These Completion Times are intended to allow adequate time to return the unit to its RTP but not permit the core to remain with unconfirmed power distributions for an extended period of time. [For this facility, these Completion Times are acceptable because.]

Action A.4.3 is modified by a Note which states that the peaking factor surveillances may only be done after the excore detectors have been calibrated to show zero tilt (i.e., Required Action A.4.2). The intent of this Note is to have the peaking factor surveillances performed at operating power levels, which can only be accomplished after the excore detectors are calibrated to show zero tilt and the core returned to power.

(continued)

(continued)

BASES (continued)

ACTIONS (continued)

B.1

If Required Actions A.1 through A.4.3 are not completed within their associated Completion Times or if the QPTR cannot be determined because of incore or excore detector system inoperability, the unit must be placed in a mode or condition in which the requirements do not apply. This is done by reducing the THERMAL POWER to < 50% of RTP within 4 hours. The allowed Completion Time of 4 hours is reasonable based on operating experience regarding the amount of time required to reach the reduced power level without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1

This QPTR surveillance verifies that the QPTR as indicated by the Nuclear Instrumentation System (NIS) excore channels is within its limits. The surveillance Frequency of 7 days when the QPTR alarm is OPERABLE is acceptable because of the low probability that this alarm would be inoperable without detection.

When the QPTR alarm is inoperable, the surveillance Frequency is increased to 12 hours. This frequency is adequate to detect any relatively slow changes in QPTR since for those causes of QPT which would occur quickly (e.g., a dropped rod), there will typically be other indications of abnormality such as a loss of reactivity which would prompt a verification of core power tilt.

SR 3.2.4.2

This surveillance is modified by Note 1, which states that it is required only when one power range channel is inoperable and the THERMAL POWER is ≥ 75% of RTP.

With an NIS power range channel inoperable, tilt monitoring for a portion of the reactor core becomes degraded. Large tilts would likely be detected with the remaining channels, but the capability for detection of small power tilts in some quadrants is decreased. Performing SR 3.2.4.2 at a Frequency of 12 hours provides an accurate alternative

(continued)

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

means for assuring that any tilt remains within its limits. [For this facility, this frequency is acceptable because.]

For purposes of monitoring the QPTR when one power range channel is inoperable, the moveable incore detectors are used to confirm that the normalized symmetric power distribution is consistent with the indicated QPTR and any previous data indicating a tilt. The incore detector monitoring is performed with a full incore flux map or two sets of four thimble locations with quarter-core symmetry. The two sets of four symmetric thimbles is a set of eight unique detector locations. These locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11, and N-8 for 3 and 4 loop cores.

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

REFERENCES

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

2. Regulatory Guide 1.77, Rev [ ], "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors," [date].

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

B 3.3 INSTRUMENTATION

B 3.3.1 Reactor Trip System (RTS) Instrumentation

BASES

BACKGROUND

The RTS initiates a plant shutdown, based upon 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 Features (ESF) Systems in mitigating accidents.

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

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

1. The Departure from Nucleate Boiling Ratio (DNBR) shall be maintained above the Safety Limit (SL) value to prevent departure from nuclear boiling;

2. Fuel centerline melt shall not occur; and

3. The RCS pressure SL of [2750] psia shall not be exceeded.

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

Accidents are events which are analyzed even though they are not expected to occur during the plant life. The acceptable limit during accidents is that the offsite dose shall be maintained within an acceptable fraction of 10 CFR 100 limits. Different accident categories are allowed a

(continued)

(continued)

BASES (continued)

BACKGROUND (continued)

different fraction of these limits, based on probability of occurrence. Meeting the acceptable dose limit for an accident category is considered having acceptable consequences for that event.

The RTS instrumentation is segmented into four distinct but interconnected modules as identified below:

1. Field transmitters or process sensors: provide a measurable electronic signal based upon the physical characteristics of the parameter being measured;

2. Signal Process Control and Protection System, including Analog Protection System, Nuclear Instrumentation System (NIS), field contacts, and protection channel sets: provides signal conditioning, bistable setpoint comparison, process algorithm actuation, compatible electrical signal output to protection system devices, and control board/control room/miscellaneous indications;

3. [Solid State Protection System (SSPS), including input, logic, and output bays: initiates proper plant shutdown and/or ESF actuation in accordance with the defined logic which is based on the bistable outputs from the signal process control and protection system]; and

4. Reactor trip switchgear, including Reactor Trip Breakers (RTBs) and bypass breakers: provides the means to interrupt power to the Control Rod Drive Mechanisms (CRDM) and allow the Rod Cluster Control Assemblies (RCCAs) or "rods" to fall into the core and shut down the reactor. The bypass breakers allow testing of the RTBs at power.

Field Transmitters or Process Sensors

In order to meet the sign demands for redundancy and reliability, more than one, and often as many as four, field transmitters or sensors are used to measure plant parameters. To account for the calibration tolerances and instrument drift, which are assumed to occur between and during calibrations, statistical allowances are provided in

(continued)

(continued)

BASES (continued)

BACKGROUND (continued)

the setpoint ALLOWABLE VALUES. The OPERABILITY of each transmitter or sensor can be evaluated when its "as found" calibration data are compared against its documented acceptance criteria.

Signal Process Control and Protection System

Generally, three or four channels of process control equipment are used for the signal processing of plant parameters measured by the field instruments. The process control equipment provides signal conditioning, comparable output signals for instruments located on the main control board, and comparison of measured input signals with setpoints established by safety analyses. These setpoints are defined in References 1, 2, and 3. If the measured value of a plant parameter exceeds the predetermined setpoint, an output from a bistable is forwarded to the [SSPS] for decision evaluation. Channel separation is maintained up to and through the input bays. However, not all plant parameters require four channels of sensor measurement and signal processing. Some plant parameters provide input only to the [SSPS] while others provide input to the [SSPS], main control board, plant computer, and one or more control systems.

Generally, if a parameter is used only for input to the protection circuits, three channels with a two-out-of-three logic are sufficient to provide the required reliability and redundancy. If one channel fails in the nonconservative direction, the function is still OPERABLE with a two-out-of-two logic. If one channel fails in the conservative direction, a trip will not occur because of the single failure; and the function is still OPERABLE with a one-out-of-two logic.

Generally, if a parameter is used for input to the [SSPS] and a control function, four channels with a two-out-of-four logic are sufficient to provide the required reliability and redundancy. The circuit must be able to withstand both an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation. Again, a single failure will neither cause nor

(continued)

(continued)

BASES (continued)

BACKGROUND (continued)

prevent the protection function actuation. These requirements are described in IEEE-279 (Ref. 4). The actual number of channels required for each plant parameter is specified in Reference 2.

Trip Setpoints and ALLOWABLE VALUES

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 2. The selection of these trip setpoints is such that adequate protection is provided when all sensor and processing time delays are taken into account. To allow for calibration tolerances, instrumentation uncertainties, instrument drift, and severe environment errors for those RTS channels which must function in harsh environments as defined by 10 CFR 50.49 (Ref. 6), ALLOWABLE VALUES specified in Table 3.3.1-1 are conservatively adjusted with respect to the analytical limits. A detailed description of the methodology used to calculate the trip setpoints, including their explicit uncertainties, is provided in Reference 5. The actual nominal trip setpoint entered into the bistable is more conservative than that specified by the ALLOWABLE VALUE, to account for changes in random measurement errors detectable by an ANALOG CHANNEL OPERATIONAL TEST. One example of such a change in measurement error is drift during the Surveillance interval. If the measured setpoint does not exceed the ALLOWABLE VALUE, the bistable is considered OPERABLE.

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

Note that in LCO 3.3.1, the ALLOWABLE VALUES of Table 3.3.1-1 are the LSSS. These ALLOWABLE VALUES are established to prevent violation of the SLs during normal plant operation and AOOs.

(continued)

(continued)

BASES (continued)

BACKGROUND (continued)

Each channel of the process control equipment can be tested on line to verify that the signal or setpoint accuracy is within the specified allowance requirements of Reference 2. 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. Surveillance Requirements for the channels are specified in the Surveillance Requirements section.

The ALLOWABLE VALUES listed in Table 3.3.1-1 are based upon the methodology described in Reference 5, 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.

[Solid State Protection System

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

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

(continued)

(continued)

BASES (continued)

BACKGROUND (continued)

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

Reactor Trip Switchgear

The RTBs are in the electrical power supply line from the control rod drive motor generator set power supply to the CRDMs. Opening of the RTBs interrupts the power to the CRDMs, which allows the shutdown rods and control rods to fall into the core by gravity. Each RTB is equipped with a bypass breaker to allow testing of the RTB while the plant is at power. During normal operation the output from the [SSPS] is a voltage signal which energizes the undervoltage coils in the RTBs, and bypass breakers if in use. When the required logic matrix combination is completed, the SSPS output voltage signal is removed, the undervoltage coils are de-energized, the breaker trip lever is actuated by the de-energized undervoltage coil, and the RTB and bypass breakers are tripped open. This allows the shutdown rods and control rods to fall into the core. In addition to the de-energization of the undervoltage coils, each breaker is also equipped with a shunt trip device which is energized to trip the breaker open upon receipt of a reactor trip signal from the [SSPS]. Either the undervoltage coil or the shunt trip mechanism is sufficient by itself, thus providing a diverse trip mechanism.

The decision logic matrix functions are described in the functional diagrams included in Reference 2. In addition to the reactor trip or ESF, these diagrams also describe the various "permissive interlocks" which are associated with plant conditions. Each train has a built-in testing device which can automatically test the decision logic matrix functions and the actuation devices while the plant is at power. When any one train is taken out of service for testing, the other train is capable of providing plant

(continued)

(continued)

BASES (continued)

BACKGROUND (continued)

monitoring and protection until the testing has been completed. The testing device is semiautomatic to minimize testing time.

APPLICABLE SAFETY ANALYSES

The RTS functions to maintain the SLs during all AOOs and mitigates the consequences of DBAs in all MODES in which the RTBs are closed.

Each of the analyzed accidents and transients can be detected by one of more RTS functions. The accident analysis described in Reference 3 takes credit for most RTS trip functions. RTS trip functions not specifically credited in the accident analysis were qualitatively credited in the safety analysis and the NRC staff-approved licensing basis for the plant. These RTS trip functions may provide protection for conditions which do not require dynamic transient analysis to demonstrate function performance. These RTS trip functions may also serve as backups to RTS trip functions that were credited in the accident analysis.

The safety analyses applicable to each RTS function are discussed below:

1. Manual Reactor Trip

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

2. Power Range Neutron Flux

The NIS power range detectors are located external to the reactor vessel and measure neutrons leaking from the core. The NIS power range detectors provide input to the Rod Control System and the Steam Generator (SG) Water Level Control System. Therefore, the actuation

(continued)

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

logic must be able to withstand an input failure to the control system (which may then require the protection function actuation) and a single failure in the other channels providing the protection function actuation. Note that this function also provides a signal to prevent automatic and manual rod withdrawal prior to initiating a reactor trip. Limiting further rod withdrawal may terminate the transient and eliminate the need to trip the reactor.

2.a. Power Range Neutron Flux--High Setpoint

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

2.b. Power Range Neutron Flux--Low Setpoint

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

2.c. Power Range Neutron Flux--f(ΔI)

The f(ΔI) function is used in the calculation of the Overtemperature ΔT trip. It is a function of the indicated difference between the upper and lower NIS power range detectors. This function measures the axial power distribution. The Overtemperature ΔT trip setpoint is varied to account for imbalances in the axial power distribution as detected by the NIS upper and lower power range detectors. If axial peaks are greater than the design limit, as indicated by the difference between the upper and lower NIS power range detectors, the trip setpoint is reduced in accordance with Note 1 of Table 3.3.1-1.

(continued)

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

3. Power Range Neutron Flux Rate

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

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

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

3.b. Power Range Neutron Flux--High Negative Rate

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

4. Intermediate Range Neutron Flux

The Intermediate Range Neutron Flux trip function ensures that protection is provided against an uncontrolled RCCA bank withdrawal accident from a subcritical condition during startup. This trip function provides redundant protection to the Power Range Neutron Flux--Low Setpoint trip function. The NIS intermediate range detectors are located external

(continued)

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

to the reactor vessel and measure neutrons leaking from the core. The NIS intermediate range detectors do not provide any input to control systems. Note that this function also provides a signal to prevent automatic and manual rod withdrawal prior to initiating a reactor trip. Limiting further rod withdrawal may terminate the transient and eliminate the need to trip the reactor.

5. Source Range Neutron Flux

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

6. Overtemperature ΔT

The Overtemperature ΔT trip function ensures that protection is provided to ensure that the design limit DNBR is met. This trip function also limits the range over which the Overpower ΔT trip function must provide protection. The inputs to the Overtemperature ΔT trip include all combinations of pressure, power, coolant temperature, and axial power distribution, assuming full reactor coolant flow. Protection from violating the DNBR limit is assured provided that the transient is slow with respect to delays from the core to the measurement system and pressure is between the Pressurizer Pressure--High and --Low trip setpoints. The Overtemperature ΔT trip function uses each loop's ΔT as a measure of reactor power and is automatically varied with the following parameters:

(continued)

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

- reactor coolant average temperature - the trip setpoint is varied to correct for changes in coolant density and specific heat capacity with changes in coolant temperature,
- pressurizer pressure - the trip setpoint is varied to correct for changes in system pressure,
- axial power distribution (discussed under Function 2.c, f(ΔI)), and
- dynamic compensation is included for system piping delays from the core to the temperature measurement system.

The Overtemperature ΔT trip function is calculated for each loop as described in Note 1 of Table 3.3.1-1. At some plants, the pressure and temperature signals are used for other control functions. For those plants, the actuation logic must be able to withstand an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation. Note that this function also provides a signal to generate a turbine runback prior to reaching the trip setpoint. A turbine runback will reduce turbine power and reactor power. A reduction in power will normally alleviate the Overtemperature ΔT condition and may prevent a reactor trip.

7. <u>Overpower ΔT</u>

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

(continued)

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

- reactor coolant average temperature - the trip setpoint is varied to correct for changes in coolant density and specific heat capacity with changes in coolant temperature, and
- the rate of change of reactor coolant average temperature and including dynamic compensation for the delays between the core and the temperature measurement system.

The Overpower ΔT trip function is calculated for each loop as per Note 2 of Table 3.3.1-1. At some plants, the temperature signals are used for other control functions. At those plants, the actuation logic must be able to withstand an input failure to the control system which may then require the protection function actuation and a single failure in the remaining channels providing the protection function actuation. Note that this function also provides a signal to generate a turbine runback prior to reaching the ALLOWABLE VALUE. A turbine runback will reduce turbine power and reactor power. A reduction in power will normally alleviate the Overpower ΔT condition and may prevent a reactor trip.

8. Pressurizer Pressure

The same sensors provide input to the Pressurizer Pressure--High and --Low Setpoint trips and the Overtemperature ΔT trip. At some plants, the Pressurizer Pressure channels are also used to provide input to the Pressurizer Pressure Control System. For those plants, the actuation logic must be able to withstand an input failure to the control system (which may then require the protection function actuation) and a single failure in the other channels providing the protection function actuation.

8.a. Pressurizer Pressure--Low Setpoint

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

(continued)

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

8.b. Pressurizer Pressure--High Setpoint

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

9. Pressurizer Water Level--High

The Pressurizer Water Level--High trip function provides a backup signal for the Pressurizer Pressure--High trip and also provides protection against water relief through the pressurizer safety valves. These valves are designed to pass steam in order to achieve their design energy removal rate. A reactor trip is actuated prior to the pressurizer becoming water solid. The pressurizer level channels are used as input to the Pressurizer Level Control System. However, the level channels do not actuate the safety valves.

10.a. Reactor Coolant Flow--Low (Single Loop)

The Reactor Coolant Flow--Low (Single Loop) trip function ensures that protection is provided against violating the DNBR limit due to low flow in one or more RCS loops. Above the P-8 setpoint, which is approximately 48% of RATED THERMAL POWER (RTP), a loss of flow in any RCS loop will actuate a reactor trip. Each RCS loop has three flow detectors to monitor flow. The flow signals are not used for any control system input.

10.b. Reactor Coolant Flow--Low (Two Loops)

The Reactor Coolant Flow--Low (Two Loops) trip function ensures that protection is provided against violating the DNBR limit due to low flow in two or more RCS loops. Above the P-7 setpoint and below the P-8 setpoint, a loss of flow in two or more loops will initiate a reactor trip. Each loop has three flow detectors to monitor flow. The flow signals are not used for any control system input.

(continued)

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

11. Reactor Coolant Pump (RCP) Breaker Open

Both RCP Breaker Open trip functions operate together on two sets of auxiliary contacts, with one set on each RCP breaker. These functions anticipate the Reactor Coolant Flow--Low trips to avoid RCS heat up that would occur before the low flow trip actuates.

11.a. RCP Breaker Open (Single Loop)

The RCP Breaker Open (Single Loop) reactor trip function ensures that protection is provided against violating the DNBR limit due to a loss of flow in one RCS loop. The position of each RCP breaker is monitored. If one RCP breaker is open above the P-8 setpoint, a reactor trip is initiated. This trip function will generate a reactor trip before the Reactor Coolant Flow--Low (Single Loop) trip setpoint is reached.

11.b. RCP Breaker Open (Two Loops)

The RCP Breaker Open (Two Loops) reactor trip function ensures that protection is provided against violating the DNBR limit due to a loss of flow in two or more RCS loops. The position of each RCP breaker is monitored. Above the P-7 setpoint and below the P-8 setpoint, a loss of flow in two or more loops will initiate a reactor trip. This trip function will generate a reactor trip before the Reactor Coolant Flow--Low (Two Loops) trip setpoint is reached.

12. Undervoltage Reactor Coolant Pumps

The Undervoltage RCPs reactor trip function ensures that protection is provided against violating the DNBR limit due to a loss of flow in two or more RCS loops. The voltage to each RCP is monitored. Above the P-7 setpoint and below the P-8 setpoint, a loss of voltage detected on two or more RCP buses will initiate a reactor trip. This trip function will generate a reactor trip before the Reactor Coolant Flow--Low (Two Loops) trip setpoint is reached. Time delays are

(continued)

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

incorporated into the Undervoltage RCPs channels to prevent reactor trips due to momentary electrical power transients.

13. Underfrequency Reactor Coolant Pumps

The Underfrequency RCPs reactor trip function ensures that protection is provided against violating the DNBR limit due to a loss of two or more RCS loops. The frequency of each RCP bus is monitored. Above the P-7 setpoint and below P-8 setpoint, a loss of frequency detected on two or more RCP buses will initiate a reactor trip. This trip function will generate a reactor trip before the Reactor Coolant Flow--Low (Two Loops) trip setpoint is reached. Time delays are incorporated into the Underfrequency RCPs channels to prevent reactor trips due to momentary electrical power transients.

14. Steam Generator Water Level--Low Low

The SG Water Level--Low Low trip function ensures that protection is provided against a loss of heat sink and actuates the Auxiliary Feedwater (AFW) System prior to uncovering the steam generator tubes. The steam generators are the heat sink for the reactor. In order to act as a heat sink, the steam generators must contain a minimum amount of water. A narrow range low-low level in any steam generator is indicative of a loss of heat sink for the reactor. The level transmitters provide input to the SG Level Control System. Therefore, the actuation logic must be able to withstand an input failure to the control system which may then require the protection function actuation and a single failure in the other channels providing the protection function actuation. This function also performs the Engineered Safety Feature Actuation System (ESFAS) function of starting the AFW pumps on low low SG level.

15. Steam Generator Water Level--Low, Coincident With Steam Flow/Feedwater Flow Mismatch

SG Water Level--Low, in conjunction with the Steam Flow/Feedwater Flow Mismatch, ensures that protection

(continued)

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

is provided against a loss of heat sink and actuates the AFW System prior to uncovering the SG tubes. In addition to a decreasing water level in the SG, the difference between feedwater flow and steam flow is evaluated to determine if feedwater flow is significantly less than steam flow. With less feedwater flow than steam flow, SG level will decrease at a rate dependent upon the magnitude of the difference in flow rates. There are two SG level channels and two Steam Flow/Feedwater Flow Mismatch channels per SG. One narrow range level channel sensing a low level coincident with one Steam Flow/Feedwater Flow Mismatch channel sensing flow mismatch (steam flow greater than feed flow) will actuate a reactor trip.

16.a. Turbine Trip--Low Fluid Oil Pressure

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

16.b. Turbine Trip--Turbine Stop Valve Closure

The Turbine Trip--Turbine Stop Valve Closure trip function is anticipatory for the loss of heat removal capabilities of the secondary system following a

(continued)

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

turbine trip from a power level below the P-9 setpoint, approximately 50% power, will not actuate a reactor trip. The trip function anticipates the loss of secondary heat removal capability that occurs when the stop valves close. Tripping the reactor in anticipation of loss of secondary heat removal acts to minimize the pressure and temperature transient on the reactor. This trip function will not and is not required to operate in the presence of a single channel failure. The plant is designed to withstand a complete loss of load and not sustain core damage or challenge the RCS pressure limitations. Core protection is provided by the Pressurizer Pressure--High trip function, and RCS integrity is ensured by the pressurizer safety valves. This trip function is diverse to the Turbine Trip--Low Fluid Oil Pressure trip function. Each turbine stop valve is equipped with one limit switch that inputs to the RTS. If all four limit switches indicate that the stop valves are all closed, a reactor trip is initiated.

17. Safety Injection (SI) Input from Engineered Safety Feature Actuation System

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

18. Reactor Trip System Interlocks

Reactor protection interlocks are provided to ensure reactor trips are in the correct configuration for the

(continued)

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

current plant status. They back up operator actions to ensure protection system functions are not bypassed during plant conditions under which the safety analysis assumes the functions are not bypassed. These are:

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

The Intermediate Range Neutron Flux, P-6 interlock is actuated when any NIS intermediate range channel goes approximately one decade above the minimum channel reading. The LCO requirement for the P-6 interlock ensures that the following functions are performed:

- on increasing power, the P-6 interlock allows the manual block of the NIS intermediate range channel approximately one decade above the minimum channel reading,
- on decreasing power, the P-6 interlock automatically energizes the NIS source ranges detectors and enables the NIS Source Range Neutron Flux reactor trip, and
- on increasing power, the P-6 interlock provides a backup block signal to the source range flux doubling circuit. Normally, this function is manually blocked by the control room operator during the reactor startup.

18.b. Low Power Reactor Trips Block, P-7

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

- on increasing power, the P-7 interlock automatically enables reactor trips on the following functions:

  Pressurizer Pressure--Low,

(continued)

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

Pressurizer Water Level--High,

Reactor Coolant Flow--Low (Two Loops),

RCPs Breaker Open (Two Loops),

Undervoltage RCPs, and

Underfrequency RCPs.

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

- on decreasing power, the P-7 interlock automatically blocks reactor trips on the following functions:

  Pressurizer Pressure--Low,

  Pressurizer Water Level--High,

  Reactor Coolant Flow--Low (Two Loops),

  Reactor Coolant Pump Breaker Open (Two Loops),

  Undervoltage RCPs, and

  Underfrequency RCPs.

18.c. <u>Power Range Neutron Flux, P-8</u>

The Power Range Neutron Flux, P-8 interlock is actuated at approximately 48% power as determined by two-out-of-four NIS power range detectors. The P-8 interlock automatically enables the Reactor Coolant Flow--Low (Single Loop) and RCP Breaker Open (Single Loop) reactor trips on low flow in one or more RCS loops on increasing power. The LCO requirement for this trip

(continued)

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

function ensures that protection is provided against a loss of flow in any RCS loop that could result in DNB conditions in the core when greater than approximately 48% power. On decreasing power, the reactor trip on low flow in any loop is automatically blocked.

18.d. Power Range Neutron Flux, P-9

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

18.e. Power Range Neutron Flux, P-10

The Power Range Neutron Flux, P-10 interlock is actuated at approximately 10% power as determined by two-out-of-four NIS power-range detectors. The LCO requirement for the P-10 interlock ensures that the following functions are performed:

- on increasing power, the P-10 interlock allows the operator to manually block the Intermediate Range Neutron Flux reactor trip. Note that blocking the reactor trip also blocks the signal to prevent automatic and manual rod withdrawal,
- on increasing power, the P-10 interlock allows the operator to manually block the Power Range Neutron Flux--Low Setpoint reactor trip,

(continued)

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

- on increasing power, the P-10 interlock automatically provides a backup block signal to the Source Range Neutron Flux reactor trip and also to de-energize the NIS source range detectors,
- the P-10 interlock provides one of the two inputs to the P-7 interlock, and
- on decreasing power, the P-10 interlock automatically enables the Power Range Neutron Flux--Low Setpoint reactor trip and the Intermediate Range Neutron Flux reactor trip (and rod stop).

In MODE 1, when the reactor is at power, the Power Range Neutron Flux, P-10, interlock must be OPERABLE. This function must also be OPERABLE in MODE 2 to ensure that core protection is provided during a startup or shutdown by the Power Range Neutron Flux--Low Setpoint and Intermediate Range Neutron Flux reactor trips. In MODES 3, 4, 5, and 6, this function does not have to be OPERABLE because the reactor is not at power and the Source Range Neutron Flux reactor trip provides core protection.

18.f. Turbine Impulse Pressure, P-13

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

19. Reactor Trip Breakers

20. Reactor Trip Breaker Undervoltage and Shunt Trip Mechanisms

(continued)

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

21. Automatic Trip Logic

The LCO requirement for the RTBs (Functions 19 and 20 and Automatic Trip Logic (Function 26) ensures that means are provided to interrupt the power to the control rod drive mechanisms and allow the rods to fall into the reactor core. Each RTB is equipped with an undervoltage coil and a shunt trip coil to trip the breaker open when needed. Each RTB is equipped with a bypass breaker to allow testing of the trip breaker while the plant is at power. The reactor trip signals generated by the RTS Automatic Trip Logic cause the RTBs and associated bypass breakers to open and shutdown the reactor.

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

LCO

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

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

(continued)

(continued)

BASES (continued)

LCO (continued)

logic channels are required to ensure no single random failure of a logic channel will disable the RTS. The logic channels are designed such that testing required while the reactor is at power may be accomplished without causing trip. Provisions to allow removing logic channels from service during maintenance are unnecessary because of the logic systems designed reliability. Specific exceptions to the above general philosophy exist and are discussed below.

Only the ALLOWABLE VALUES are specified for each RTS trip function in the LCO. Nominal trip setpoints are specified in the plant-specific setpoint calculations. The nominal setpoints are selected to ensure the setpoint measured by ANALOG CHANNEL OPERATIONAL 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 each RTS trip function. These uncertainties are defined in the "Plant-specific RTS/ESFAS Setpoint Methodology" (Ref. 5).

Reactor Trip System Functions

The trip function channels specified in Table 3.3.1-1 ar considered OPERABLE when the following numbered items of applicable RTS trip function criteria are met:

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

2. Channel measurement uncertainties and trip setpoints are known via test, analysis, or design information 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

(continued)

(continued)

BASES (continued)

LCO (continued)

4. Associated operational bypasses are not enabled except under the conditions specified by the LCO Applicability statement for the function.

The following bases for each trip function identify the items above that are applicable to establish each trip function OPERABILITY:

1. Manual Reactor Trip

The LCO requires one Manual Reactor Trip channel per train to be OPERABLE in MODE 1 or 2 and in MODE 3, 4, or 5 with RTBs closed and Control Rod Drive System capable of rod withdrawal. Manual Reactor Trip is OPERABLE when it satisfies OPERABILITY requirements 1 and 3. Two independent channels are required to be OPERABLE so that no single random failure will disable the manual reactor trip function.

2.a. Power Range Neutron Flux--High Setpoint

The LCO requires all four of the Power Range Neutron Flux--High channels to be OPERABLE in MODE 1 or 2. Power Range Neutron Flux--High Setpoint is OPERABLE when it satisfies OPERABILITY requirements 1, 2, and 3.

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

2.b. Power Range Neutron Flux--Low Setpoint

The LCO requires all four of the Power Range Neutron Flux--Low channels to be OPERABLE in MODE 1 below Power Range Neutron Flux, P-10 or MODE 2. Power Range Neutron Flux--Low Setpoint is OPERABLE when it satisfies OPERABILITY requirements 1, 2, 3, and 4.

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

2.c. Power Range Neutron Flux--f(ΔI)

The LCO requires all four channels of f(ΔI) to be OPERABLE in MODE 1 or 2. Power Range Neutron

(continued)

(continued)

BASES (continued)

LCO (continued)

Flux--f(ΔI) is OPERABLE when it satisfies OPERABILITY requirements 1, 2, and 3.

The LCO requirement ensures that if axial peaks are greater than the design limit, as indicated by the difference between the upper and lower NIS power range detectors, the Overtemperature ΔT trip setpoint is reduced in accordance with Note 1 of Table 3.3.1-1. Therefore, no trip setpoint ALLOWABLE VALUE is specifically applied to the f(ΔI) trip function.

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

The LCO requires all four of the Power Range Neutron Flux--High Positive Rate channels to be OPERABLE. Power Range Neutron Flux--High Positive Rate is OPERABLE when it satisfies OPERABILITY requirements 1, 2, and 3.

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

3.b. Power Range Neutron Flux--High Negative Rate

The LCO requires all four Power Range Neutro Flux--High Negative Rate channels to the OPE ABLE. In MODE 1 or 2, Power Range Neutron Flux--Hi h Negative Rate is OPERABLE when it satisfies OPERABIL..Y requirements 1, 2, and 3.

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

4. Intermediate Range Neutron Flux

The LCO requires two channels of Intermediate Range Neutron Flux to be OPERABLE. Two OPERABLE channels are sufficient to ensure no single random failure will disable this trip function. Intermediate Range Neutron Flux is OPERABLE when it satisfies OPERABILITY requirements 1, 2, 3, and 4.

(continued)

(continued)

BASES (continued)

LCO (continued)

Because this trip function is important only during startup, there is generally no need to disable channels for testing while the function is required to be OPERABLE.

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

5. Source Range Neutron Flux

The LCO requires two channels of Source Range Neutron Flux to be OPERABLE in MODE 2 below P6 and in MODE 3, 4, or 5 with RTBs closed and Control Rod Drive System capable of rod withdrawal. Two OPERABLE channels are sufficient to ensure no single random failure will disable this trip function. The LCO requires two channels for the Source Range Neutron Flux to be OPERABLE in MODES 3, 4, or 5 with RTBs open. In this case, the source range function is to provide input and indication to the LCO 3.3.5, "Boron Dilution Protection System (BDPS)." Source Range Neutron Flux is OPERABLE when it satisfies OPERABILITY requirements 1, 2, and 3. An additional requirement is that the channel's control room readouts are functional and in service.

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

6. Overtemperature ΔT

The LCO requires all four channels of the Overtemperature ΔT trip function to be OPERABLE in MODE 1 or 2 for two and four loop plants. LCO requires all three channels on the Overtemperature ΔT trip function to be OPERABLE in MODE 1 or 2 for three loop plants. Overtemperature ΔT is OPERABLE when it satisfies OPERABILITY requirements 1, 2, and 3. Note that the Overtemperature ΔT function receives input from channels shared with other RTS functions. Failures that affect multiple functions require entry into the Conditions applicable to all affected functions.

The Overtemperature ΔT trip function is calculated for each loop as described in Note 1 of Table 3.3.1-1.

(continued)

(continued)

BASES (continued)

LCO (continued)

Trip occurs if Overtemperature ΔT is indicated in two loops.

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

7. Overpower ΔT

The LCO requires four channels (two- and four-loop plants) and three channels (three-loop plants) of the Overpower ΔT trip function to be OPERABLE in MODE 1 or 2. Overpower ΔT is OPERABLE when it satisfies OPERABILITY requirements 1, 2, and 3. Note that the Overpower ΔT function receives input from channels shared with other RTS functions. Failures that affect multiple functions require entry into the Conditions applicable to all affected functions.

The Overpower ΔT trip function is calculated for each loop as per Note 2 of Table 3.3.1-1. Trip occurs if Overpower ΔT is indicated in two loops.

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

8.a. Pressurizer Pressure--Low Setpoint

The LCO requires all four channels (two- and four-loop plants) and three channels (three-loop plants) of Pressurizer Pressure--Low Setpoint to be OPERABLE in MODE 1 above P-7. Pressurizer Pressure--Low Setpoint is OPERABLE when it satisfies OPERABILITY requirements 1, 2, 3, and 4. This trip function ensures that protection is provided against violating the DNBR limit due to low pressure.

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

8.b. Pressurizer Pressure--High Setpoint

The LCO requires all four channels (two- and four-loop plants) and three channels (three loop-plants) of Pressurizer Pressure--High Setpoint to be OPERABLE in

(continued)

(continued)

BASES (continued)

LCO (continued)

MODE 1 or 2. Pressurizer Pressure--High Setpoint is OPERABLE when it satisfies OPERABILITY requirements 1, 2, and 3. This function ensures that protection is provided against overpressurizing the RCS.

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

9. Pressurizer Water Level--High

The LCO requires three channels of Pressurizer Water Level--High to be OPERABLE in MODE 1 above P-7. Pressurizer Water Level--High is OPERABLE when it satisfies OPERABILITY requirements 1, 2, 3, and 4. This trip function provides a backup signal for the Pressurizer Pressure--High trip and also provides protection against water relief through the pressurizer safety valves.

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

10.a. Reactor Coolant Flow--Low (Single Loop)

The LCO requires three Reactor Coolant Flow--Low channels per loop to be OPERABLE in MODE 1 above P-8. Reactor coolant flow is OPERABLE when it satisfies OPERABILITY requirements 1, 2, 3, and 4. This trip function ensures that protection is provided against violating the DNBR limit due to low flow in one or more RCS loops. Above the P-8 setpoint, which is approximately 48% of RTP, a loss of flow in any RCS loop will actuate a reactor trip. Each RCS loop has three flow detectors to monitor flow.

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

10.b. Reactor Coolant Flow--Low (Two Loops)

The LCO requires three Reactor Coolant Flow--Low channels per loop to be OPERABLE in MODE 1 above P-7 and below P-8. Reactor Coolant Flow--Low is OPERABLE when it satisfies OPERABILITY requirements 1, 2, 3, and 4. This trip function ensures that protection is

(continued)

(continued)

BASES (continued)

LCO (continued)

provided against violating the DNBR limit due to low flow in two or more RCS loops. Above the P-7 setpoint and below the P-8 setpoint, a loss of flow in two or more loops will initiate a reactor trip. Each loop has three flow detectors to monitor flow.

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

11.a. Reactor Coolant Pump Breaker Open (Single Loop)

The LCO requires one RCP Breaker Open channel per RCP to be OPERABLE in MODE 1 above P-8. RCP Breakers Open is OPERABLE when it satisfies OPERABILITY requirements 1, 3, and 4. One OPERABLE channel is sufficient for this trip function because the RCS Flow--Low trip alone provides sufficient protection of plant SLs for loss-of-flow events. The RCP Breaker Open trip serves only to anticipate the low flow trip, minimizing the thermal transient associated with loss of a pump.

The RCP Breaker Open (Single Loop) reactor trip function ensures that protection is provided against violating the DNBR limit due to a loss of flow in one RCS loop. The position of each RCP breaker is monitored. If one RCP breaker is open above the P-8 setpoint, a reactor trip is initiated. This trip function will generate a reactor trip before the Reactor Coolant Flow--Low (Single Loop) trip setpoint is reached. This function measures only the discrete position (open or closed) of the RCP breaker using a position switch. Therefore, the function has no adjustable trip setpoint with which to associate an ALLOWABLE VALUE.

11.b. Reactor Coolant Pump Breaker Open (Two Loops)

The LCO requires one RCP Breaker Open channel per RCP to be OPERABLE in MODE 1 above P-7 and below P-8. The RCP Breaker Open is OPERABLE when it satisfies OPERABILITY requirements 1, 3, and 4. One OPERABLE channel is sufficient for this function because the RCS Flow--Low trip alone provides sufficient protection of plant SLs for loss-of-flow events. The RCP Breaker

(continued)

(continued)

LCO
(continued)

Open trip serves only to anticipate the low flow trip, minimizing the thermal transient associated with loss of an RCP.

The LCO requirement for the RCP Breaker Open (Two Loops) reactor trip function ensures that protection is provided against violating the DNBR limit due to a loss of flow in two or more RCS loops. The position of each RCP breaker is monitored. Above the P-7 setpoint and below the P-8 setpoint, a loss of flow in two or more loops will initiate a reactor trip. This trip function will generate a reactor trip before the Reactor Coolant Flow--Low (Two Loops) trip setpoint is reached. This function measures only the discrete position (open or closed) of the RCP breaker using a position switch. Therefore, the function has no adjustable trip setpoint with which to associate an ALLOWABLE VALUE.

12. Undervoltage Reactor Coolant Pumps

The LCO requires three Undervoltage RCPs channels per bus to be OPERABLE in MODE 1 above P-7. An Undervoltage RCPs channel is OPERABLE when it satisfies OPERABILITY requirements 1, 2, and 3. The Undervoltage RCPs reactor trip function ensures that protection is provided against violating the DNBR limit due to a loss of flow in two or more RCS loops.

Above the P-7 setpoint, a loss of voltage detected in two or more loops will initiate a reactor trip. This trip function will generate a reactor trip before the Reactor Coolant Flow--Low (Two Loops) trip setpoint is reached. [For this facility, describe relay configuration and how the trip meets single failure criterion for single-phasing events. If action of other relays, eg., phase differential current, is required to ensure trip in the event of single phasing, discuss how OPERABILITY of these other relays affects OPERABILITY of this trip function and identify LCO covering the other relays.] Time delays are incorporated into the Undervoltage RCPs channels to prevent reactor trips due to momentary electrical power transients. [For this facility, the time delay

(continued)

(continued)

BASES (continued)

LCO (continued)

setpoint and justification for not including time delay setpoint ALLOWABLE VALUE in the LCO table are as follows:]

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

13. Underfrequency Reactor Coolant Pumps

The LCO requires three Underfrequency RCPs channels to be OPERABLE in MODE 1 above P7. An Underfrequency RCPs channel is OPERABLE when it satisfies OPERABILITY requirements 1, 2, 3, and 4. The Underfrequency RCPs reactor trip function ensures that protection is provided against violating the DNBR limit due to the loss of two or more RCS loops.

Above P-7 setpoint, a loss of frequency detected in two or more RCP buses will initiate a reactor trip. This trip function will generate a reactor trip before the Reactor Coolant Flow--Low trip setpoint is reached. Time delays are incorporated into the Underfrequency RCPs channels to prevent reactor trips due to momentary electrical power transients.

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

14. Steam Generator Water Level--Low Low

The LCO requires four channels of SG Water Level--Low Low per SG to be OPERABLE in MODE 1 or 2 for four-loop plants in which these channels are shared between protection and control. In two-, three-, and four-loop plants where three SG Water Levels are dedicated to the RTS, only three channels per SG are required to be OPERABLE. SG Water Level--Low Low is OPERABLE when it satisfies OPERABILITY requirements 1, 2, and 3. The SG Water Level--Low Low trip function ensures that protection is provided against a loss of heat sink and actuates the AFW System prior to uncovering the SG tubes.

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

(continued)

(continued)

BASES (continued)

LCO (continued)

15. Steam Cenerator Water Level--Low, Coincident with Steam Flow/Feedwater Flow Mismatch

The LCO requires two channels of SG Water Level--Low coincident with Steam Flow/Feedwater Flow Mismatch. This trip function is OPERABLE when it satisfies OPERABILITY requirements 1, 2, and 3.

SG Water Level--Low, in conjunction with Steam Flow/Feedwater Flow Mismatch, ensures that protection is provided against a loss of heat sink and actuates the AFW System prior to uncovoring the SG tubes.

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

16.a. Turbine Trip--Low Fluid Oil Pressure

The LCO requires three channels of Turbine Trip--Low Fluid Oil Pressure to be OPERABLE in MODE 1 above P-9. Turbine Trip--Low Fluid Oil Pressure is OPERABLE when it satisfies OPERABILITY requirements 1, 2, 3, and 4. This trip function is anticipatory for the loss-of-heat-removal capabilities of the secondary system following a turbine trip. This trip function acts to minimize the pressure/temperature transient on the reactor. A low pressure sensed by two-out-of-three of the pressure switches will actuate a reactor trip.

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

16.b. Turbine Trip--Turbine Stop Valve Closure

The LCO requires four Turbine Trip--Turbine Stop Valve Closure channels, one per valve, to be OPERABLE in MODE 1 above P-9. All four channels must trip to cause reactor trip. Turbine trip stop valve closure is OPERABLE when it satisfies the OPERABILITY requirements 1, 2, 3, and 4. One OPERABLE channel per valve in a four-out-of-four logic configuration is acceptable because the plant is designed to withstand stop valve closure and not sustain core damage or challenge RCS pressure limitations. This trip function is anticipatory for the loss-of-heat-removal capabilities

(continued)

(continued)

BASES (continued)

LCO (continued)

of the secondary system following a turbine trip. This trip function acts to minimize the pressure/temperature transient on the reactor.

The ALLOWABLE VALUE for this function is set to assure channel trip occurs when the associated stop valve is completely closed.

17. Safety Injection Input from Engineering Safety Feature Actuation System

The LCO requires two trains of SI Input from ESFAS to be OPERABLE in MODE 1 or 2.

[For this facility, an OPERABLE SI Input from ESFAS constitutes the following:]

This function ensures that, if a reactor trip has not already been generated by the RTS, the ESFAS automatic actuation logic will initiate a reactor trip upon any signal which initiates SI.

18. Reactor Trip System Interlocks

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

The LCO requires two channels of Intermediate Range Neutron Flux, P-6 interlock to be OPERABLE in MODE 2 when below the P-6 interlock setpoint. Intermediate Range Neutron Flux, P-6 interlock is OPERABLE when it satisfies the OPERABILITY requirements 1, 2, and 3.

[For this facility, the basis for the required number of channels and the ALLOWABLE VALUE is as follows:]

18.b. Low Power Reactor Trips Block, P-7

The LCO requires one channel per train of Low Power Reactor Trips Block, P-7 interlock to be OPERABLE in MODE 1. Low Power Reactor Trips Block, P-7 interlock is OPERABLE when it satisfies the OPERABILITY requirements 1 and 3.

(continued)

(continued)

BASES (continued)

LCO
(continued)

[For this facility, the basis for the required number of channels and the ALLOWABLE VALUE is as follows:]

18.c. Power Range Neutron Flux, P-8

The LCO requires four channels of Power Range Neutron Flux, P-8 interlock to be OPERABLE in MODE 1. Power Range Neutron Flux, P-8 interlock is OPERABLE when it satisfies the OPERABILITY requirements 1, 2, and 3.

[For this facility, the basis for the required number of channels and the ALLOWABLE VALUE are as follows:]

18.d. Power Range Neutron Flux, P-9

The LCO requires four channels of Power Range Neutron Flux, P-9 interlock to be OPERABLE in MODE 1. Power Range Neutron Flux, P-9 interlock is OPERABLE when it satisfies the OPERABILITY requirements 1, 2 and 3.

[For this facility, the basis for the required number of channels and the ALLOWABLE VALUE are as follows:]

18.e. Power Range Neutron Flux, P-10

The LCO requires four channels of Power Range Neutron Flux, P-10 interlock to be OPERABLE in MODE 1 or 2. Power Range Neutron Flux, P-10 interlock is OPERABLE when it satisfies the OPERABILITY requirements 1, 2, and 3.

[For this facility, the basis for the required number of channels and the ALLOWABLE VALUE are as follows:]

18.f. Turbine Impulse Pressure, P-13

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

(continued)

(continued)

BASES (continued)

LCO (continued)

MODE 1. Turbine Impulse Pressure, P-13 is OPERABLE when it satisfies the OPERABILITY requirements 1, 2, and 3.

[For this facility, the basis for the required number of channels and the ALLOWABLE VALUE are as follows:]

19. Reactor Trip Breakers

This trip function applies to the RTBs exclusive of individual trip mechanisms. The LCO requires two OPERABLE trains of trip breakers. A trip breaker train consists of all trip breakers associated with a single RTS logic train that are racked in, closed, and capable of supplying power to the Control Rod Drive (CRD) System. Thus the train may consist of the main breaker, bypass breaker, or main breaker and bypass breaker, depending upon the system configuration. Two OPERABLE trains ensure no single random failure can disable the RTS trip capability. A trip breaker train is considered OPERABLE when requirements 1 and 3 are met.

20. Reactor Trip Breaker Undervoltage and Shunt Trip Mechanisms

The LCO requires both the undervoltage and shunt trip mechanisms to be OPERABLE for each RTB that is in service. The trip mechanisms are not required to be OPERABLE for trip breakers that are open, racked out, incapable of supplying power to the CRD System, or declared inoperable under function 19 above. RTB trip mechanisms are considered OPERABLE when requirements 1 and 3 are met. OPERABILITY of both trip mechanisms on each breaker ensures that no single trip mechanism failure will prevent opening the breakers on a valid signal.

21. Automatic Trip Logic

The LCO requires two channels of RTS Automatic Trip Logic to be OPERABLE. Automatic Trip Logic channels are considered OPERABLE when requirements 1 and 3 are met. Having two OPERABLE channels ensures that random

(continued)

(continued)

BASES (continued)

LCO (continued)

failure of a single logic channel will not prevent reactor trip.

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

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

[For this facility, the supported systems impacted by the inoperability of the RTS instrumentation and the justification of whether or not each supported system is declared inoperable are as follows:]

It should be noted that LCO 3.3.1 may need to be augmented with additional Conditions, if it is determined that the RTS provides support to other systems included in the STS.

APPLICABILITY

1. Manual Reactor Trip

In MODE 1 or 2, manual initiation of a reactor trip must be OPERABLE. These are the MODES in which the shutdown rods and/or control rods are partially or fully withdrawn from the core. In MODE 3, 4, or 5, the manual initiation function must also be OPERABLE if the shutdown rods or control rods are withdrawn or the CRD System is capable of withdrawing the shutdown rods or the control rods. In MODE 3, 4, or 5, manual initiation of a reactor trip does not have to be OPERABLE if the CRD System is not capable of withdrawing the shutdown rods or control rods. If the rods cannot be withdrawn from the core, there is no need to be able to trip the reactor because all of the rods are inserted. In MODE 6, neither the shutdown rods nor the control rods are permitted to be withdrawn and the CRDMs are disconnected from the control rods and shutdown rods. Therefore, the manual initiation function does not have to be OPERABLE.

(continued)

(continued)

BASES (continued)

APPLICABILITY (continued)

2.a. Power Range Neutron Flux--High Setpoint

In MODE 1 or 2, when a positive reactivity excursion could occur, the Power Range Neutron Flux--High setpoint trip must be OPERABLE. This function will terminate the reactivity excursion and shutdown the reactor prior to reaching a power level that could damage the fuel. In MODE 3, 4, 5, or 6, the Power Range Neutron Flux--High Setpoint does not have to be OPERABLE because the reactor is shutdown and a reactivity excursion in the power range cannot occur. Other RTS functions and administrative controls provide protection against reactivity additions when in MODE 3, 4, 5, or 6. In addition, the NIS power range detectors cannot detect neutron levels in this range.

2.b. Power Range Neutron Flux--Low Setpoint

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

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

2.c. Power Range Neutron Flux--f(ΔI)

In MODE 1 or 2, when the Overtemperature ΔT trip is required to be OPERABLE, the f(ΔI) function must be OPERABLE as the f(ΔI) function provides one of the inputs to the Overtemperature ΔT trip.

(continued)

(continued)

BASES (continued)

APPLICABILITY (continued)

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

In MODE 1 or 2, when there is a potential to add a large amount of positive reactivity from a rod ejection accident (REA), the Power Range Neutron Flux--High Positive Rate trip must be OPERABLE. In MODE 3, 4, 5, or 6, the Power Range Neutron Flux--High Positive Rate trip function does not have to be OPERABLE because other RTS trip functions and administrative controls will provide protection against positive reactivity additions. Also, since only the shutdown banks may be withdrawn in MODE 3, 4, or 5, the remaining complement of control bank worth ensures a sufficient degree of SHUTDOWN MARGIN in the event of an REA. In MODE 6, no rods are withdrawn and the SHUTDOWN MARGIN is increased during refueling operations. The reactor vessel head is also removed or the closure bolts are detensioned preventing any pressure buildup. In addition, the NIS power range detectors cannot detect neutron levels present in this mode.

3.b. Power Range Neutron Flux--High Negative Rate

In MODE 1 or 2, when there is a potential for a multiple rod drop accident to occur, the Power Range Neutron Flux--High Negative Rate trip must be OPERABLE. In MODE 3, 4, 5, or 6, the Power Range Neutron Flux--High Negative Rate trip function does not have to be OPERABLE because the core is not critical and DNBR is not a concern. Also, since only the shutdown banks may be withdrawn in MODE 3, 4, or 5, the remaining complement of control bank worth ensures a sufficient degree of SHUTDOWN MARGIN in the event of an REA. In MODE 6, no rods are withdrawn and the required SHUTDOWN MARGIN is increased during refueling operations. In addition, the NIS power range detectors cannot detect neutron levels present in this mode.

4. Intermediate Range Neutron Flux

In MODE 1 below the P-10 setpoint and in MODE 2 when there is a potential for an uncontrolled rod withdrawal accident during reactor startup, the Intermediate Range Neutron Flux trip must be OPERABLE. Above the P-10 setpoint, the Power Range Neutron Flux--High Setpoint

(continued)

(continued)

BASES (continued)

APPLICABILITY (continued)

trip and the Power Range Neutron Flux--High Positive Rate trip provide core protection for a rod withdrawal accident. In MODE 3, 4, or 5, the Intermediate Range Neutron Flux trip does not have to be OPERABLE because the control rods must be fully inserted and only the shutdown rods may be withdrawn. The reactor cannot be started up in this condition. The core also has the required SHUTDOWN MARGIN to mitigate the consequences of a positive reactivity addition accident. In MODE 6, all rods are fully inserted and the core has a required increased SHUTDOWN MARGIN. Also, the NIS intermediate range detectors cannot detect neutron levels present in this mode.

5. <u>Source Range Neutron Flux</u>

In MODE 2 when below the P-6 setpoint during a reactor startup, the Source Range Neutron Flux trip must be OPERABLE. Above the P-6 setpoint, the Intermediate Range Neutron Flux trip and the Power Range Neutron Flux--Low Setpoint trip will provide core protection for reactivity accidents. Above the P-6 setpoint, the NIS source range detectors are de-energized and inoperable.

In MODE 3, 4, or 5 with the reactor shut down, the Source Range Neutron Flux trip function must also be OPERABLE. If the CRD System is capable of rod withdrawal, the Source Range Neutron Flux trip must be OPERABLE to provide core protection against a rod withdrawal accident. If the CRD System is not capable of rod withdrawal, the source range detectors are not required to trip the reactor. However, their monitoring function must be OPERABLE to monitor core neutron levels and provide indication of reactivity changes that may occur as a result of events like a boron dilution. These inputs are provided to LCO 3.3.5, "Boron Dilution Protection System." The requirements for the NIS source range detectors in MODE 6 are addressed in LCO 3.9.3.

6. <u>Overtemperature ΔT</u>

In MODE 1 or 2, the Overtemperature ΔT trip must be OPERABLE to prevent DNB. In MODE 3, 4, 5, or 6, this

(continued)

(continued)

BASES (continued)

APPLICABILITY (continued)

trip function does not have to be OPERABLE because the reactor is not operating and there is insufficient heat production to be concerned about DNB.

7. Overpower ΔT

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

8.a. Pressurizer Pressure--Low Setpoint

In MODE 1, when DNB is a major concern, the Pressurizer Pressure--Low Setpoint trip must be OPERABLE. This trip function is automatically enabled on increasing power by the P-7 interlock (NIS power range P-10 or turbine impulse pressure greater than approximately 10% of full power equivalent (P-13)). On decreasing power, this trip function is automatically blocked below P-7. Below the P-7 setpoint, no conceivable power distributions can occur that would cause DNB concerns.

8.b. Pressurizer Pressure--High Setpoint

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

(continued)

(continued)

BASES (continued)

APPLICABILITY (continued)

9. Pressurizer Water Level--High

In MODE 1 when there is a potential for overfilling the pressurizer, the Pressurizer Water Level--High trip must be OPERABLE. This trip function is automatically enabled on increasing power by the P-7 interlock. On decreasing power, this trip function is automatically blocked below P-7. Below the P-7 setpoint, transients which could raise the pressurizer water level will be slow and the operator will have sufficient time to evaluate plant conditions and take corrective actions.

10.a. Reactor Coolant Flow--Low (Single Loop)

In MODE 1 above the P-8 setpoint, when a loss of flow in one RCS loop could result in DNB conditions in the core, the Reactor Coolant Flow--Low (Single Loop) trip must be OPERABLE. In MODE 1 below the P-8 setpoint, a loss of flow in two or more loops is required to actuate a reactor trip (Function 10.b) because of the lower power level and the greater margin to the design limit DNBR.

10.b. Reactor Coolant Flow--Low (Two Loops)

In MODE 1 above the P-7 setpoint and below the P-8 setpoint, the Reactor Coolant Flow--Low (Two Loops) trip must be OPERABLE. Below the P-7 setpoint, all reactor trips on low flow are automatically blocked since no conceivable power distributions could occur that would cause a DNB concern at this low power level. Above the P-7 setpoint, the reactor trip on low flow in two or more RCS loops is automatically enabled. Above the P-8 setpoint, a loss of flow in any one loop will actuate a reactor trip because of the higher power level and the reduced margin to the design limit DNBR.

11.a. Reactor Coolant Pump Breaker Open (Single Loop)

In MODE 1 above the P-8 setpoint, when a loss of flow in any RCS loop could result in DNB conditions in the core, the RCP Breaker Open (Single Loop) trip must be OPERABLE. In MODE 1 below the P-8 setpoint, a loss of flow in two or more loops is required to actuate a

(continued)

(continued)

BASES (continued)

APPLICABILITY (continued)

reactor trip because of the lower power level and the greater margin to the design limit DNBR.

11.b. Reactor Coolant Pump Breaker Open (Two Loops)

In MODE 1 above the P-7 setpoint and below the P-8 setpoint, the RCP Breaker Open (Two Loops) trip must be OPERABLE. Below the P-7 setpoint, all reactor trips on loss of flow are automatically blocked since no conceivable power distributions could occur that would cause a DNB concern at this low power level. Above the P-7 setpoint, the reactor trip on loss of flow in two RCS loops is automatically enabled. Above the P-8 setpoint, a loss of flow in any one loop will actuate a reactor trip because of the higher power level and the reduced margin to the design limit DNBR.

12. Undervoltage Reactor Coolant Pumps

In MODE 1 above the P-7 setpoint, the Undervoltage RCP trip must be OPERABLE. Below the P-7 setpoint, all reactor trips on loss of flow are automatically blocked since no conceivable power distributions could occur that would cause a DNB concern at this low power level. Above the P-7 setpoint, the reactor trip on loss of flow in two or more RCS loops is automatically enabled. This function uses the same relays as the ESFAS Function 7.f, "Undervoltage Reactor Coolant Pump (RCP)" start of the AFW pumps.

13. Underfrequency Reactor Coolant Pumps

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

14. Steam Generator Water Level--Low Low

In MODE 1 or 2, when the reactor requires a heat sink, the SG Water Level--Low Low trip must be OPERABLE. The normal source of water for the SGs is the Main

(continued)

(continued)

BASES (continued)

APPLICABILITY (continued)

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

15. Steam Generator Water Level--Low, Coincident with Steam Flow/Feedwater Flow Mismatch

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

16.a. Turbine Trip--Low Fluid Oil Pressure

In MODE 1 above the P-9 setpoint, the Turbine Trip--Low Fluid Oil Pressure trip must be OPERABLE. Below the P-9 setpoint, a turbine trip does not actuate a reactor trip. In MODE 2, 3, 4, 5, or 6, there is no potential for a turbine trip; and the Turbine Trip--Low Fluid Oil Pressure trip function does not need to be OPERABLE.

(continued)

(continued)

BASES (continued)

APPLICABILITY (continued)

16.b. Turbine Trip--Turbine Stop Valve Closure

In MODE 1 above the P-9 setpoint, the Turbine Trip--Turbine Stop Valve Closure trip must be OPERABLE. Below the P-9 setpoint, a load rejection can be accommodated by the Steam Dump System. In MODE 2, 3, 4, 5, or 6, there is no potential for a load rejection; and the Turbine Trip--Turbine Stop Valve Closure trip function does not need to be OPERABLE.

17. Safety Injection Input from Engineering Safety Features Actuation System

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

18. Reactor Trip System Interlocks

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

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

In MODE 2, when below the P-6 interlock setpoint, the P-6 interlock must be operable. Above the P-6 interlock setpoint, the NIS Source Range Neutron Flux reactor trip will be blocked; and this function will no longer be necessary. In MODE 3, 4, 5, or 6, the P-6 interlock does not have to be OPERABLE because the NIS Source Range is providing core protection.

(continued)

(continued)

BASES (continued)

APPLICABILITY (continued)

18.b. Low Power Reactor Trips Block, P-7

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

18.c. Power Range Neutron Flux, P-8

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

18.d. Power Range Neutron Flux, P-9

A reactor trip is automatically initiated on a turbine trip when above the P-9 setpoint to minimize the transient on the reactor. In MODE 1, when a turbine trip could cause a load rejection beyond the capacity of the Steam Dump System, the Power Range Neutron Flux interlock must be OPERABLE. In MODE 2, 3, 4, 5, or 6, this function does not have to be OPERABLE because the reactor is not at a power level sufficient to have a load rejection beyond the capacity of the Steam Dump System.

18.e. Power Range Neutron Flux, P-10

In MODE 1, when the reactor is at power, the Power Range Neutron Flux, P-10 interlock must be OPERABLE. This function must be OPERABLE in MODE 2 to ensure that core protection is provided during a startup or shutdown by the Power Range Neutron Flux--Low Setpoint and Intermediate Range Neutron Flux reactor trips. In MODE 3, 4, 5, or 6, this function does not have to be OPERABLE

(continued)

(continued)

BASES (continued)

APPLICABILITY (continued)

because the reactor is not a power and the Source Range Neutron Flux reactor trip provides core protection.

18.f. Turbine Impulse Chamber Pressure, P-13

In MODE 1, the Turbine Impulse Chamber Pressure, P-13 interlock must be OPERABLE when the turbine generator is operating. The interlock function is not required OPERABLE in MODE 2, 3, 4, 5, or 6 because the turbine generator is not operating.

19. Reactor Trip Breakers (RTBs)

These trip functions must be OPERABLE in MODE 1 or 2 when the reactor is critical. In MODE 3, 4, or 5, these RTS trip functions must be OPERABLE when the RTBs and associated bypass breakers are in use, are closed, and the CRD System is capable of rod withdrawal.

20. Reactor Trip Breaker Undervoltage and Shunt Trip Mechanisms

These trip functions must be OPERABLE in MODE 1 or when the reactor is critical. In MODE 3, 4, or 5, these RTS trip functions must be OPERABLE when the RTBs and associated bypass breakers are in use, are closed, and the CRD System is capable of rod withdrawal.

21. Automatic Trip Logic

These trip functions must be OPERABLE in MODE 1 or 2 when the reactor is critical. In MODE 3, 4, or 5, these RTS trip functions must be OPERABLE when the RTBs and associated bypass breakers are in use, are closed, and the CRD System is capable of rod withdrawal.

A Note has been added to provide clarification that, for this LCO, each function specified in Table 3.3.1-1 is treated as an independent entity with an independent Completion Time.

(continued)

BASES (continued)

ACTIONS

In order for a facility to take credit for topical reports as the basis for justifying Completion Times, topical reports must 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 functions channels. 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 (but it is still within the time period assumed in the safety analysis) rather than a total loss of function. This determination is generally made during the performance of an ANALOG CHANNEL OPERATIONAL TEST, when the process instrumentation is set up for adjustment to bring it within specification. If the trip setpoint is less conservative than the ALLOWABLE VALUE, the channel must be declared inoperable immediately and the appropriate Condition(s) from Table 3.3.1-1 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 is applicable to all RTS protection functions. Condition A addresses the situation where one or more required channels for one or more functions are inoperable at the same time. The Required Action is to refer to

(continued)

(continued)

BASES (continued)

ACTIONS (continued)

Table 3.3.1-1 and to take the Required Actions for the protection functions affected. The Completion Times are those from the referenced Conditions and Required Actions.

Condition B

Condition B applies to the Manual Reactor Trip in MODE 1 or 2. This action addresses the train orientation of the [SSPS] for this function. With one train inoperable, the inoperable train must be restored to OPERABLE status within 48 hours. In this Condition, the remaining OPERABLE train is adequate to perform the safety function. [For this facility, the Completion Time of 48 hours is justified as follows: (WCAP-10271-P-A is not applicable for this function - Ref. 7).]

The plant must be placed in a MODE in which the requirement does not apply if the Manual Reactor Trip function cannot be restored to OPERABLE status in the allowed 48-hour Completion Time. This is done by placing the plant in at least MODE 3 in 6 additional hours (54 hours total time) followed by opening the reactor trip breakers in 1 additional hour (55 hours total time). The 6 additional hours to reach MODE 3 and the 1 hour to open the RTBs are reasonable, based on operating experience, to reach MODE 3 and open the RTBs from full power operation in an orderly manner and without challenging plant systems. With the RTBs open and the plant in MODE 3, this trip function is no longer required to be OPERABLE.

Condition C

Condition C applies to the following reactor trip functions in MODE 3, 4, or 5 with the RTBs closed and the CRD System capable of rod withdrawal:

- Manual Reactor Trip;
- Reactor Trip Breakers;
- Reactor Trip Breaker Undervoltage and Shunt Trip Mechanisms; and
- Automatic Trip Logic.

(continued)

(continued)

BASES (continued)

ACTIONS (continued)

This action addresses the train orientation of the [SSPS] for these functions. With one train inoperable, the inoperable train must be restored to OPERABLE status within 48 hours. In this Condition, the remaining OPERABLE train is adequate to perform the safety function. The plant must be placed in a MODE in which the requirement does not apply if the affected function(s) cannot be restored to OPERABLE status in the allowed 48-hour Completion Time. This is done by opening the RTBs within the next hour. The additional hour provides sufficient time to accomplish the action in an orderly manner. With the RTBs open, these functions are no longer required.

[For this facility, the Completion Time of 48 hours is justified as follows: (WCAP-10271-P-A is not applicable for this Function - Ref. 7).]

<u>Condition D</u>

Condition D applies to the following reactor trip function:

- Power Range Neutron Flux--High Setpoint.

The NIS power range detectors provide input to the CRD System and the SG Water Level Control System and, therefore, have a two-out-of-four trip logic. A known inoperable channel must be restored to OPERABLE status in 4 hours or placed in the tripped condition. This results in a partial trip condition requiring only one-out-of-three logic for actuation. The 6 hours allowed to place the inoperable channel in the tripped condition is justified in Reference 7.

In addition to placing the inoperable channel in the tripped condition, THERMAL POWER must be reduced to ≤ 75% of RTP within 4 hours. Reducing the power level prevents operation of the core with radial power distributions beyond the design limits. With one of the NIS power range detectors inoperable, 1/4 of the radial power distribution monitoring capability is lost.

Therefore, the power range high trip setpoint must be reduced to [85]% of RTP to ensure anomalous radial power distributions cannot cause the DNBR or linear power density SLs to be challenged in the unmonitored quadrant when the

(continued)

(continued)

BASES (continued)

ACTIONS (continued)

high power trip occurs. Four hours are allowed for a total of eight, to reduce the Power Range Neutron Flux--High Setpoint.

As an alternative to the above actions, the inoperable channel can be placed in the tripped condition within 6 hours and the QUADRANT POWER TILT RATIO (QPTR) monitored every 12 hours as per SR 3.2.4.2, QPTR verification. Calculating QPTR every 12 hours compensates for the lost monitoring capability due to the inoperable NIS power range detector and allows continued plant operation at power levels > 75% of RTP. The 6-hour Completion Time and the 12-hour Frequency are consistent with LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)."

If the Required Actions described above cannot be met within the specified Completion Times, the unit must be placed in a MODE where this function is no longer required OPERABLE. An additional 6 hours beyond the Completion Time for Required Action D.2.2 and Required Action D.3.1 are allowed to place the unit in MODE 3. Six hours is a reasonable time, based on operating experience, to reach MODE 3 from MODE 1 from full power in an orderly manner and without challenging plant systems. If Required Action D.2.3 or Required Action D.3.2 cannot be completed within their allowed Completion Times, LCO 3.0.3 must be entered.

The Required Actions have been modified by a Note, which allows placing the inoperable channel in the bypass condition for up to 4 hours while performing routine Surveillance testing of other channels and while resetting the trip setpoints of the OPERABLE channels in Required Action D.2.3. The 4-hour time limit is justified in Reference 7.

<u>Condition E</u>

Condition E applies to the following reactor trip functions:

- Power Range Neutron Flux--Low Setpoint;
- Power Range Neutron Flux--f(ΔI);
- Overtemperature ΔT;

(continued)

(continued)

BASES (continued)

ACTIONS (continued)

- Overpower ΔT;
- Power Range Neutron Flux--High Positive Rate;
- Power Range Neutron Flux--High Negative Rate;
- Pressurizer Pressure--High Setpoint;
- SG Water Level--Low Low; and
- SG Water Level Low coincident with Steam Flow/Feedwater Flow Mismatch.

A known inoperable channel must be restored to OPERABLE status or placed in the tripped condition within 6 hours. Placing the channel in the tripped condition results in a partial trip condition requiring only one-out-of-two logic for actuation of the two-out-of-three trips and one-out-of-three logic for actuation of the two-out-of-four trips. The 6 hours allowed to place the inoperable channel in the tripped condition is justified in Reference 7.

If the operable channel cannot be restored or placed in the trip condition within the specified Completion Time, the unit must be placed in a MODE where these functions are not required OPERABLE. An additional 6 hours is allowed to place the unit in MODE 3. Six hours is a reasonable time, based on operating experience, to place the unit in MODE 3 from MODE 1 from full power in an orderly manner and without challenging plant systems.

The Required Actions have been modified by a Note which allows placing the inoperable channel in the bypassed condition for up to 4 hours while performing routine Surveillance testing of the other channels. The 4-hour time limit is justified in Reference 7.

Condition F

Condition F applies to the Intermediate Range Neutron Flux trip when above the P-6 setpoint and below the P-10 setpoint. Above the P-6 setpoint and below the P-10 setpoint, the NIS intermediate range detector performs the monitoring functions. If THERMAL POWER is greater than the

(continued)

(continued)

BASES (continued)

ACTIONS (continued)

P-6 setpoint but less than the P-10 setpoint, 2 hours is allowed to either return the inoperable channels to OPERABLE status or to reduce THERMAL POWER below the P-6 setpoint or increase to THERMAL POWER above the P-10 setpoint. Required Action F.3 has been modified with a clause to restrict THERMAL POWER increases above P-10 in the event that Required Action G.3 is active. This precludes actions in two related Conditions from being in conflict with each other. The NIS Intermediate Range Neutron Flux channels must be OPERABLE when the power level is above the capability of the source range, P-6, and below the capability of the power range, P-10. If THERMAL POWER is greater than the P-10 setpoint, the NIS power range detectors perform the monitoring and protective functions and the intermediate range is not required. The Completion Times allow for a slow and controlled power adjustment above P-10 or below P-6 take into account the redundant capability afforded by the redundant OPERABLE channel and the low probability of its failure during this period. This action does not require the inoperable channel to be tripped because the function uses one-out-of-two logic. Tripping one channel would trip the reactor. Thus, the Required Actions specified in this Condition are only applicable when channel failure does not result in reactor trip.

Condition G

Condition G applies to two inoperable Intermediate Range Neutron Flux trip channels in MODE 2 above the P-6 setpoint and below the P-10 setpoint. Required Actions specified in this Condition are only applicable when channel failures do not result in reactor trip. Above the P-6 setpoint and below the P-10 setpoint, the NIS intermediate range detector performs the monitoring functions. With no intermediate range channels OPERABLE, the Required Actions are to suspend operations involving positive reactivity additions immediately. This will preclude any power level increase since there are no OPERABLE Intermediate Range Neutron Flux channels. The operator must also restore at least one channel to OPERABLE status or reduce THERMAL POWER below the P-6 setpoint within two hours. Below P-6, the Source Range Neutron Flux channels will be able to monitor the core power level. The Completion Time of 2 hours will allow a slow and controlled power reduction to less than the P-6 setpoint and takes into account the low probability of

(continued)

(continued)

BASES (continued)

ACTIONS (continued)

occurrence of an event during this period that may require the protection afforded by the NIS Intermediate Range Neutron Flux trip.

Condition H

Condition H applies to the Intermediate Range Neutron Flux trip when below the P-6 setpoint. Below the P-6 setpoint, the NIS source range performs the monitoring and protective functions. If THERMAL POWER is greater than the P-6 setpoint, the inoperable NIS intermediate range channel(s) must be returned to OPERABLE status prior to increasing power above the P-6 setpoint. The NIS intermediate range channels must be OPERABLE when the power level is above the capability of the source range, P-6, and below the capability of the power range, P-10.

Condition I

Condition I applies to one inoperable Source Range Neutron Flux trip channel when in MODE 2, below the P-6 setpoint, and performing a reactor startup. With the plant in this Condition, < P-6, the NIS source range performs the monitoring and protective functions. With one of the two channels inoperable, operations involving positive reactivity additions shall be suspended immediately.

This will preclude any power escalation. With only one source range channel OPERABLE, core protection is severely reduced and any actions that add positive reactivity to the core must be suspended immediately.

Condition J

Condition J applies to two inoperable Source Range Neutron Flux trip channels when in MODE 2, below the P-6 setpoint, and performing a reactor startup, or in MODE 3, 4, or 5 with the RTBs closed and the CRD System capable of rod withdrawal. With the plant in this Condition, < P-6, the NIS source range performs the monitoring and protective functions. With both source range channels inoperable, operations involving positive reactivity additions shall be suspended immediately. With no source range channels OPERABLE, core protection is severely reduced and any actions that add positive reactivity to the core must be

(continued)

(continued)

BASES (continued)

ACTIONS (continued)

suspended as quickly as possible. In addition to suspending positive reactivity additions, the RTBs must be opened immediately. With the RTBs open and positive reactivity additions suspended, the core is in a relatively safe and stable condition. With the RTBs open, the plant enters Condition L.

Condition K

Condition K applies to one inoperable source range channel in MODE 3, 4, or 5 with the RTBs closed and the CRD System capable of rod withdrawal. With the plant in this condition, < P-6, the NIS source range performs the monitoring and protective functions. With one of the source range channels inoperable, 48 hours is allowed to restore it to an OPERABLE status. If the channel cannot be returned to an OPERABLE status, 1 additional hour is allowed to have the RTBs open. In addition to opening the RTBs, all operations involving positive reactivity additions must be suspended within the same hour. Suspension of positive reactivity additions and opening the RTBs will preclude any power excursion. Also, all valves that could add unborated water to the RCS must be closed within the same 1 hour as specified in LCO 3.9.2, "Unborated Water Source Isolation Valves." The isolation of unborated water sources will preclude a boron dilution accident. The allowance of 48 hours to restore the channel to OPERABLE status and the additional hour to open the RTBs are justified in Reference 7. The allowance of 1 hour to stop positive reactivity additions and close the unborated water source isolation valves is based on operating experience which provides sufficient time to accomplish the Required Actions in an orderly manner.

Condition L

Condition L applies to no OPERABLE Source Range Neutron Flux trip channels when in MODE 3, 4, or 5 with the RTBs open. With the plant in this condition, the NIS source range performs the monitoring and protective functions. With no source range channels OPERABLE, operations involving positive reactivity additions shall be suspended immediately. This will preclude any power escalation. In addition to suspension of positive reactivity additions, all valves that could add unborated water to the RCS must be

(continued)

(continued)

BASES (continued)

ACTIONS (continued)

closed within 1 hour as specified in LCO 3.9.2, "Unborated Water Source Isolation Valves." The isolation of unborated water sources will preclude a boron dilution accident.

Also, the SHUTDOWN MARGIN must be verified within 1 hour and once per 12 hours as per SR 3.1.1.1, SHUTDOWN MARGIN verification. With no source range channels OPERABLE, core protection is severely reduced. Verifying the SHUTDOWN MARGIN within 1 hour allows sufficient time to perform the calculations and determine that the SHUTDOWN MARGIN requirements are met. The SHUTDOWN MARGIN must also be verified once per 12 hours thereafter to ensure that the core reactivity has not changed. Required Action L.1 precludes any positive reactivity additions; therefore, core reactivity should not be increasing, and a 12-hour Frequency is adequate. The Completion Times of within 1 hour and once per 12 hours are based on operating experience in performing the Required Actions and the knowledge that plant conditions will change slowly.

Condition M

Condition M applies to the following reactor trip functions:

- Pressurizer Pressure--Low Setpoint;
- Pressurizer Water Level--High;
- Reactor Coolant Flow--Low (Two Loops);
- RCP Breaker Open (Two Loops);
- Undervoltage RCPs; and
- Underfrequency RCPs.

With one channel inoperable, the inoperable channel must be restored to OPERABLE status or placed in the tripped condition within 6 hours. Placing the channel in the tripped condition results in a partial trip condition requiring only one additional channel to initiate a reactor trip above the P-7 setpoint and below the P-8 setpoint. This function does not have to be OPERABLE below the P-7 setpoint because there are no loss of flow trips below the P-7 setpoint. The 6 hours allowed to place the channel

(continued)

(continued)

BASES (continued)

ACTIONS (continued)

in the tripped condition is justified in Reference 7. An additional 6 hours is allowed to reduce power < P-7 if the inoperable channel cannot be restored to OPERABLE status or placed in trip within the specified Completion Time.

Allowance of this time interval takes into consideration the redundant capability provided by the remaining redundant OPERABLE channel and the low probability of occurrence of an event during this period that may require the protection afforded by the functions associated with Condition M.

The Required Actions have been modified by a Note, which allows placing the inoperable channel in the bypassed condition for up to 4 hours while performing routine Surveillance testing of the other channels. The 4-hour time limit is justified in Reference 7.

Condition N

Condition N is applicable to the Reactor Coolant Flow--Low (Single Loop) reactor trip function. With one channel inoperable, the inoperable channel must be restored to OPERABLE status, or the channel placed in trip, within 6 hours. If the channel cannot be restored to OPERABLE status or the channel placed in trip within the 6 hours, then THERMAL POWER must be reduced below the P-8 setpoint within the next 4 hours. This places the plant in a MODE where the LCO is no longer applicable. This trip function does not have to be OPERABLE below the P-8 setpoint because other RTS trip functions provide core protection below the P-8 setpoint. The 6 hours allowed to restore the channel to OPERABLE status or place in trip and the 4 additional hours allowed to reduce THERMAL POWER to below the P-8 setpoint are justified in Reference 7.

The Required Actions have been modified by a Note which allows placing the inoperable channel in the bypassed condition for up to 4 hours while performing routine Surveillance testing of the other channels. The 4-hour time limit is justified in Reference 7.

Condition O

Condition O is applicable to the RCP Breaker Open (Single Loop) reactor trip function. There is one breaker

(continued)

(continued)

BASES (continued)

ACTIONS (continued)

position device per RCP breaker. With one channel inoperable, the inoperable channel must be restored to OPERABLE status within 6 hours. If the channel cannot be restored to OPERABLE status within the 6 hours, then THERMAL POWER must be reduced below the P-8 setpoint within the next 4 hours. This places the plant in a MODE where the LCO is no longer applicable. This function does not have to be OPERABLE below the P-8 setpoint because other RTS functions provide core protection below the P-8 setpoint. The 6 hours allowed to restore the channel to OPERABLE status and the 4 additional hours allowed to reduce THERMAL POWER to below the P-8 setpoint are justified in Reference 7.

The Required Actions have been modified by a Note which allows placing the inoperable channel in the bypassed condition for up to 4 hours while performing routine Surveillance testing of the other channels. The 4-hour time limit is justified in Reference 7.

Condition P

Condition P applies to Turbine Trip on Low Fluid Oil Pressure or on Turbine Stop Valve Closure. With one channel inoperable, the inoperable channel must be restored to OPERABLE status or placed in the trip condition within 6 hours. If placed in the tripped condition, this results in a partial trip condition requiring only one additional channel to initiate a reactor trip. If the channel cannot be restored to OPERABLE status or placed in the trip condition, then power must be reduced below the P-9 setpoint within the next 4 hours. The 6 hours allowed to place the inoperable channel in the tripped condition and the 4 hours allowed for reducing power are justified in Reference 7.

The Required Actions have been modified by a Note which allows placing the inoperable channel in the bypassed condition for up to 4 hours while performing routine Surveillance testing of the other channels. The 4-hour time limit is justified in Reference 7.

Condition Q

Condition Q applies to the SI Input from ESFAS reactor trip and the RTS Automatic Trip Logic in MODES 1 and 2. These actions address the train orientation of the RTS for these

(continued)

(continued)

BASES (continued)

ACTIONS (continued)

functions. With one train inoperable, 6 hours are allowed to restore the train to OPERABLE status or the unit must be placed in MODE 3 within the next 6 hours. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging plant systems.

[For this facility, the Completion Time of 6 hours and bypassing of a train for Surveillance are justified as follows: (WCAP 10271-P-A is not applicable for this function).]

The Required Actions have been modified by a Note that allows bypassing one train up to 4 hours for Surveillance testing, provided the other train is OPERABLE.

Condition R

Condition R applies to the RTBs in MODES 1 and 2. These actions address the train orientation of the RTS for the RTBs. With one train inoperable, 1 hour is allowed to restore the train to OPERABLE status or the unit must be placed in MODE 3 within the next 6 hours. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging plant systems. The 1-hour and 6-hour Completion Times are equal to the time allowed by LCO 3.0.3 for shutdown actions in the event of a complete loss of RTS function. Placing the plant in MODE 3 removes the requirement for this particular function.

The Required Actions have been modified by a Note that allows one channel to be bypassed for up to 2 hours for Surveillance testing. One channel of reactor trip breakers and RTS logic functions may be bypassed for up to 2 hours for maintenance on undervoltage or shunt trip mechanisms provided the other channel is OPERABLE. The 2-hour time limit is justified in Reference 7.

Condition S

Condition S applies to the P-6 and P-10 interlocks. With one channel inoperable for one-out-of-two- or two-out-of-four-coincidence logic, the associated interlock must be verified to be in its required state for the

(continued)

(continued)

BASES (continued)

ACTIONS (continued)

existing plant condition within 1 hour or the unit must be placed in MODE 3 within the next 6 hours. Verifying the interlock status manually accomplishes the interlock's function. The Completion Time of 1 hour is based on operating experience and the minimum amount of time allowed for manual operator actions. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging plant systems. The 1-hour and 6-hour Completion Times are equal to the time allowed by LCO 3.0.3 for shutdown actions in the event of a complete loss of RTS function.

Condition T

Condition T applies to the P-7, P-8, P-9, and P-13 interlocks. With one channel inoperable for one-out-of-two- or two-out-of-four-coincidence logic, the associated interlock must be verified in its required state for the existing plant condition within 1 hour or the unit must be placed in MODE 2 within the next 6 hours. These actions are conservative for the case where power level is being raised. Verifying the interlock status manually accomplishes the interlock's function. The Completion Time of 1 hour is based on operating experience and the minimum amount of time allowed for manual operator actions. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 2 from full power in an orderly manner and without challenging plant systems.

Condition U

Condition U applies to the RTB Undervoltage and Shunt Trip Mechanisms, or diverse trip features, in MODES 1 and 2. With one of the diverse trip features inoperable, it must be restored to an OPERABLE status within 48 hours or the unit must be placed in a MODE where the requirement does not apply. This is accomplished by placing the plant in MODE 3 within the next 6 hours. The Completion Time of 6 hours is a reasonable time, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging plant systems. The affected breaker shall not be bypassed while one of the diverse features is inoperable except for the time required to perform maintenance to

(continued)

(continued)

BASES (continued)

ACTIONS (continued)

one of the diverse features is inoperable except for the time required to perform maintenance to restore the breaker to an OPERABLE status. The allowable time for performing maintenance of the diverse features is 2 hours for the reasons stated under Condition R.

[For this facility, the Completion Time of 48 hours is justified as follows:]

Condition V

Condition V is applicable to each one of the RTS functions presented in Table 3.3.1-1.

Required Action V.1 verifies that all required support features associated with the other redundant train(s) or channel(s) are OPERABLE within a Completion Time of 1 hour. The specified Completion Time is sufficient for plant operations personnel to make this determination. If verification determines loss of functional capability, LCO 3.0.3 must be immediately entered. However, if the support feature LCO or RTS 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 RTS function are found in the SRs column of Table 3.3.1-1 for that function.

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

Note that each channel of process protection supplies both trains of the RTS. When testing Channel I, Train A and Train B must be examined. Similarly, Train A and Train B must be examined when testing Channel II, Channel III, and Channel IV (if applicable). The CHANNEL CALIBRATION and ANALOG CHANNEL OPERATIONAL TESTS are performed in a manner that is consistent with the assumptions used in analytically calculating the required channel accuracies. For channels that include dynamic transfer functions, e.g., lag,

(continued)

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

lead/lag, rate/lag, etc., the response time test may be performed with the transfer function set to one, with the resulting measured response time compared to the appropriate FSAR response time. Alternately, the response time test can be performed with the time constants set to their nominal value provided the required response time is analytically calculated assuming the time constants are set at their nominal values. The response time may be measured by a series of overlapping tests such that the entire response time is measured.

SR 3.3.1.1

Performance of the CHANNEL CHECK once every 12 hours ensures that gross failure of instrumentation has not occurred. A CHANNEL CHECK is 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 have drifted outside its limit. If the channels are within the match criteria, it is an indication that the channels are OPERABLE. If the channels are normally off scale during times when Surveillance is required, the CHANNEL CHECK will only verify that they are off scale in the same direction. Off-scale low current-loop channels are verified to be reading at the bottom of the range and not failed downscale.

The Surveillance Frequency, about once every shift, is based on operating experience that demonstrates the rarity of channel failure. Thus, performance of the CHANNEL CHECK guarantees that undetected overt channel failure is limited to 12 hours. Since the probability of two random failures

(continued)

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

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.

In the case of RTS trips with multiple inputs, such as the Overtemperature ΔT, a CHANNEL CHECK must be performed on all inputs.

SR 3.3.1.2

SR 3.3.1.2 is the performance of a CHANNEL CALIBRATION for the NIS power range channels every 24 hours when greater than 15% of RTP. The outputs of the NIS power range channels are normalized to the results of the calorimetric.

Several Notes have been added to this SR 3.3.1.2. The first Note indicates that the CHANNEL CALIBRATION consists only of adjustments based on a comparison of the results of the calorimetric with the NIS channel output. This Surveillance is not required if < 15% of RTP. The power level must be > 15% of RTP to obtain accurate data. At lower power levels, the accuracy of calorimetric data is questionable. A second Note indicates that the NIS channel output shall be adjusted consistent with the calorimetric results if the absolute difference between the NIS channel output and the calorimetric are > 2%. A third Note indicates that the provision of SR 3.0.4 is not applicable for this SR 3.3.1.2.

The Frequency of every 24 hours is adequate, based on plant operating experience, considering instrument reliability and operating history data for instrument drift that demonstrates the change in the difference between NIS and heat balance calculated powers rarely exceeds a small fraction of 2% in any 24-hour period.

In addition, control room operators periodically monitor redundant indications and alarm to detect deviations in channel outputs.

(continued)

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.3

SR 3.3.1.3 is the performance of a CHANNEL CALIBRATION for the NIS power range channels every 31 effective full power days (EFPDs) when greater than 15% of RTP. The outputs of the NIS power range channels are normalized to the results of the incore detectors to ensure OPERABILITY of the Power Range Neutron Flux f(ΔI) function (Function 2.c). This Surveillance is not required when ≤ 15% of RTP. Several Notes have been added to this SR 3.3.1.3. The first Note indicates that the CHANNEL CALIBRATION consists only of a single point comparison of the incore to excore NIS AXIAL FLUX DIFFERENCE (AFD) when > 15% of RTP. The power level must be > 15% of RTP to obtain accurate data. At lower power levels, the accuracy of the data would be questionable. A second Note indicates that the excore NIS channel shall be recalibrated if the absolute difference between the incore and excore AFD is ≥ 3%. A third Note indicates that the provision of SR 3.0.4 is not applicable for this SR 3.3.1.3. The Frequency of every 31 EFPDs is adequate, based on plant operating experience, considering instrument reliability and operating history data for instrument drift and that the slow changes in neutron flux during the fuel cycle can be detected at this interval.

SR 3.3.1.4

SR 3.3.1.4 is the performance of a TRIP ACTUATING DEVICE OPERATIONAL TEST every 31 days on a STAGGERED TEST BASIS. This test shall verify OPERABILITY by actuation of the end devices.

Each RTB is actuated by an undervoltage coil and a shunt trip coil. The system is designed so that either de-energizing the undervoltage coil or energizing the shunt trip coil will cause the circuit breaker to open. When a RTB is opened, either during an automatic reactor trip or by using the manual push buttons in the control room, the undervoltage coil is de-energized and the shunt trip coil is energized. This makes it impossible to determine if one of the coils or associated circuitry is defective. Therefore, during undervoltage coil testing, the shunt trip coils shall remain de-energized, preventing their operation. Conversely, during shunt trip coil testing, the undervoltage coils shall remain energized, preventing their operation.

(continued)

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

This test ensures that every undervoltage coil and every shunt trip coil is capable of performing its intended function and that no single failure of any RTB component will prevent a reactor trip.

The RTB test shall include independent verification of the undervoltage and shunt trip mechanisms. The bypass breaker test shall include a local shunt trip. A Note has been added to indicate that this test must be performed on the bypass breaker prior to place it in service.

The Frequency of every 31 days on a STAGGERED TEST BASIS is justified in Reference 7.

SR 3.3.1.5

SR 3.3.1.5 is the performance of an ACTUATION LOGIC TEST. The [SSPS] is tested every 31 days on a STAGGERED TEST BASIS, using the [semiautomatic tester]. The train being tested is placed in the bypass condition, thus preventing inadvertent actuation. Through the [semiautomatic tester], all possible logic combinations, with and without applicable permissives, are tested for each protection function. The time allowed for the testing, 4 hours, and the Frequency of every 31 days on a STAGGERED TEST BASIS are justified in Reference 7.

SR 3.3.1.6

SR 3.3.1.6 is the performance of a CHANNEL CALIBRATION for the NIS power range channels every 92 EFPDs when > 50% of RTP. The outputs of the NIS power range channels are normalized to the results of the incore detectors to ensure OPERABILITY of the Power Range Neutron Flux f(ΔI) function (Function 2.c). This Surveillance is not required when THERMAL POWER ≤ 50% of RTP. A Note has been added to indicate that the CHANNEL CALIBRATION consists of a calibration of the excore NIS channels based on the incore system results (incore-excore calibration). The power level must be > 50% of RTP because lower power levels do not provide results that are as accurate. A second Note has been added to indicate that the provision SR 3.0.4 is not applicable to this SR 3.3.1.6. The Surveillance Frequency of 92 EFPDs is justified in Reference 7.

(continued)

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.7

SR 3.3.1.7 is the performance of an ANALOG CHANNEL OPERATIONAL TEST every 92 days. This test is a periodic check of the analog process control equipment while the plant is at power. When the channel is placed in the test condition, the input to the [SSPS] is changed to the tripped condition, and the input from the transmitter is removed. This allows a test signal to be introduced into the instrument loop. The input to the bistable can be measured, thus noting the accuracy of the signal conditioning of the process control modules upstream. The setpoint of the bistable can be determined by varying the input and observing the bistable test lamp. "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" values and the previous test "as left" values must be consistent with the drift allowance used in the setpoint analysis. Recalibration restores OPERABILITY of an otherwise functional component that does not meet these criteria. However, repeated failures of the same channel over a relatively small number of test intervals must be considered as potentially indicating a deterministic failure which cannot be corrected by recalibration. Individual process control modules may be tested in place using multiple sets of test jacks, or by module removal and verification in a calibration laboratory. If individual modules are checked, a verification of the loop accuracy is necessary to satisfy the statistical analyses assumptions. This test Frequency of 92 days is justified in Reference 7.

SR 3.3.1.8

SR 3.3.1.8 is the performance of an ANALOG CHANNEL OPERATIONAL TEST as described in SR 3.3.1.7, except it is modified by a Note that this test shall include verification that the P-6 and P-10 interlocks are in their required state for the existing plant condition. This test ensures that the NIS source range and intermediate range channels are OPERABLE prior to taking the reactor critical.

(continued)

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.9

SR 3.3.1.9 is the performance of a TRIP ACTUATING DEVICE OPERATIONAL TEST as described in SR 3.3.1.4, except that the test is performed every 92 days instead of every 31 days and is justified in Reference 7.

SR 3.3.1.10

CHANNEL CALIBRATION is a complete check of the instrument channel including the detector. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift between successive 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 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.

This SR 3.3.1.10 is modified by a Note that this test shall include verification that the time constants are adjusted to the prescribed values where applicable.

(continued)

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.11

SR 3.3.1.11 is the performance of a CHANNEL CALIBRATION every 18 months. The neutron detectors may be excluded from the CHANNEL CALIBRATION per the Note. This Surveillance is not required for the NIS power range detectors for entry into MODE 2 or 1, and is not required for the NIS intermediate range detectors for entry into MODE 2, because the plant must be in at least MODE 2 to perform the test for the intermediate range detectors and MODE 1 for the power range detectors. The 18-month Frequency was developed considering it was prudent that these Surveillances only be performed during a plant outage. This was due to the plant conditions needed to perform the Surveillance and the potential for unplanned plant transients if the Surveillance is performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed on the 18-month Frequency.

SR 3.3.1.12

SR 3.3.1.12 is the performance of a CHANNEL CALIBRATION, as described in SR 3.3.1.10, every 18 months. This test shall include verification of the RCS resistance temperature detector (RTD) bypass loop flowrate per the Note. This test will verify the rate-lag compensation for flow from the core to the RTDs.

CHANNEL CALIBRATION is a complete check of the instrument channel including the detector. The test verifies the channel responds to measured parameter with the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift between successive tests, to ensure that the instrument channel remains operational with the setpoint within the assumptions of the plant-specific setpoint analysis. Transmitter "as found" values 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.

(continued)

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

Recalibration restores the 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. RTD channels may be calibrated in place using cross-calibration techniques, or in a test bath after removal from piping. For cross-calibration, at least one RTD should be replaced with a newly calibrated RTD during each refueling cycle to ensure accurate RTD cross-calibration. This replacement RTD must be the same model as the remaining RTDs. Using a newly calibrated RTD as a reference assures RTD signal drift continues to remain random rather than systematic and is within the limits specified in the plant-specific setpoint analysis. The replacement interval may be extended to alternate refueling if it is demonstrated that over the extended interval, the RTDs drift is random rather than systematic and is bounded by the plant-specific setpoint analyses assumptions. This determination may use results of statistical analysis of operating 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 in the setpoint analysis.

SR 3.3.1.13

SR 3.3.1.13 is the performance of an ANALOG CHANNEL OPERATIONAL TEST, which is performed every 18 months.

[For this facility, the 18-month Frequency has been shown to be acceptable through operating experience as follows:]

(continued)

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.14

SR 3.3.1.14 is the performance of a TRIP ACTUATING DEVICE OPERATIONAL TEST, as described in SR 3.3.1.4, except that the test is performed every 18 months.

[For this facility, the Surveillance test acceptance criteria is as follows:]

[For this facility, the 18-month Frequency has been shown to be acceptable through operating experience as follows:]

SR 3.3.1.15

SR 3.3.1.15 is the performance of a TRIP ACTUATING DEVICE OPERATIONAL TEST, as described in SR 3.3.1.4, except that this test is performed prior to reactor startup. This Surveillance is not required if it has been performed within the previous 31 days per the Note. Verification of the trip setpoint does not have to be performed for this Surveillance. Performance of this test will ensure that the turbine trip-reactor trip functions are OPERABLE prior to taking the reactor critical. This test cannot be performed with the reactor at power and must therefore be performed prior to reactor startup.

SR 3.3.1.16

This SR 3.3.1.16 ensures that the train actuation response times are verified on a STAGGERED TEST BASIS. The response time values are provided in the FSAR and are the maximum values assumed in the safety analyses. Individual component response times are not modeled in the analyses. The analyses model the overall or total elapsed time, from the point at which the parameter exceeds the trip setpoint value at the sensor to the point at which the equipment reaches the required functional state (e.g., control and shutdown rods fully inserted in the reactor core). The test may be performed in one measurement or in overlapping segments, with verification that all components are measured.

Each train's response must be verified every 18 months on a STAGGERED TEST BASIS (i.e., Train A at 18 months after initial startup, Train B at 36 months, and then Train A

(continued)

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

again). The 18-month Frequency was developed because many Surveillances can only be performed during a plant outage. Response times cannot be determined during plant operation because equipment operation is required to measure response times. Experience has shown that these components usually pass this surveillance when performed on the 18-month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

The SR 3.3.1.16 is modified by exempting neutron detectors from response time testing.

REFERENCES

1. [Unit Name] FSAR, Section [ ], "Engineered Safety Features."
2. [Unit Name] FSAR, Section [ ], "Instrumentation and Controls."
3. [Unit Name] FSAR, Section [ ], "Accident Analysis."
4. Institute of Electrical and Electronic Engineers, IEEE-279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations," April 5, 1972.
5. [Plant-specific RTS/ESFAS Setpoint Methodology Study]
6. Title 10, Code of Federal Regulations, Part 50.49, "Environmental Qualifications of Electric Equipment Important to Safety for Nuclear Power Plants."
7. WCAP-10271-P-A, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System," May 1986.

B 3.3 INSTRUMENTATION

B 3.3.2 Engineered Safety Feature Actuation System (ESFAS) Instrumentation

BASES

BACKGROUND

The ESFAS initiates necessary safety systems, based upon the values of selected unit parameters, to protect against violating core design limits and the Reactor Coolant System (RCS) pressure boundary, and to mitigate accidents.

The ESFAS instrumentation is segmented into three distinct but interconnected modules as identified below:

- Field transmitters or process sensors and instrumentation: provide a measurable electronic signal based upon the physical characteristics of the parameter being measured;
- Signal processing equipment including analog protection system, Nuclear Instrumentation System, field contacts, and protection channel sets: provide signal conditioning, bistable setpoint comparison, process algorithm actuation, compatible electrical signal output to protection system devices, and control board/control room/miscellaneous indications;
- [Solid State Protection System (SSPS) including input, logic, and output bays: initiates the proper plant shutdown or engineered safety feature (ESF) actuation in accordance with the defined logic and based upon the bistable outputs from the signal process control and protection system.]

Field Transmitters and Sensors

In order to meet the design demands for redundancy and reliability, more than one, and often as many as four, field sensors or transmitters are used to measure plant parameters. In many cases field transmitters and sensors that input to the ESFAS are shared with the Reactor Protection System (RPS). In some cases, the same channels also provide control system inputs. To account for the calibration tolerances and instrument drift, which are assumed to occur between calibrations, statistical

(continued)

(continued)

BASES (continued)

BACKGROUND (continued)

allowances are provided in the setpoint ALLOWABLE VALUES. The OPERABILITY of each transmitter or sensor can be evaluated when its "as found" calibration data are compared against its documented acceptance criteria.

Signal Processing Equipment

Generally, three or four channels of process control equipment are used for the signal processing of plant parameters measured by the field instruments. The process control equipment provides signal conditioning, comparable output signals for instruments located on the main control board, and comparison of measured input signals with setpoints established by safety analyses. These setpoints are defined in References 1, 2, and 3. If the measured value of a unit parameter exceeds the predetermined setpoint, an output from a bistable is forwarded to the [SSPS] for decision evaluation. Channel separation is maintained up to and through the input bays. However, not all unit parameters require four channels of sensor measurement and signal processing. Some unit parameters provide input only to the [SSPS], while others provide input to the [SSPS], main control board, plant computer, and one or more control systems.

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

Generally, if a parameter is used for input to the [SSPS] and a control function, four channels with a two-out-of-four logic are sufficient to provide the required reliability and redundancy. The circuit must be able to withstand both an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation. Again, a single failure will neither cause nor prevent the protection function actuation.

(continued)

(continued)

BASES (continued)

BACKGROUND (continued)

These requirements are described in IEEE-279 (Ref. 4). The actual number of channels required for each unit parameter is specified in Reference 2.

Trip Setpoints and ALLOWABLE VALUES

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 + comparator setting accuracy).

The trip setpoints used in the bistables are based on the analytical limits stated in Reference 2. The selection of these trip setpoints is such that adequate protection is provided when all sensor and processing time delays are taken into account. To allow for calibration tolerances, instrumentation uncertainties, instrument drift, and severe environment errors for those ESFAS channels that must function in harsh environments as defined by 10 CFR 50.49 (Ref. 6), ALLOWABLE VALUES specified in Table 3.3.2-1 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 RPS/ESFAS Setpoint Methodology Study] (Ref. 5). The actual nominal trip setpoint entered into the bistable is normally still more conservative than that specified by the ALLOWABLE VALUE to account for changes in random measurement errors detectable by an ANALOG CHANNEL OPERATIONAL TEST. One example of such a change in measurement error is drift during the surveillance interval. If the measured setpoint does not exceed the ALLOWABLE VALUE, the bistable is considered OPERABLE.

Setpoints in accordance with the ALLOWABLE VALUE ensure that the consequences of Design Basis Accidents (DBAs) will be acceptable, providing the plant is operated from within the LCOs at the onset of the DBA and the equipment functions as designed.

Each channel can be tested on line to verify that the signal processing equipment and setpoint accuracy is within the specified allowance requirements of Reference 2. Once a designated channel is taken out of service for testing, a

(continued)

(continued)

BASES (continued)

BACKGROUND (continued)

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 3.3.2-1 are based upon the methodology described in Reference 5, 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.

Solid State Protection System

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

The SSPS performs the decision logic for actuating ESF equipment actuation; generates the electrical output signals, which initiate the required actuation; and provides the status, permissive, and annunciator output signals to the main control room of the plant.

The bistable outputs from the signal processing equipment are sensed by the SSPS equipment and combined into logic matrices that represent combinations indicative of various transients. If a required logic matrix combination is completed, the system will send actuation signals via master and slave relays to those components whose aggregate function best serves to alleviate the condition and restore the plant to a safe condition. Examples are given in the section on Applicable Safety Analyses.

(continued)

(continued)

BASES (continued)

BACKGROUND (continued)

Each train has a built-in testing device which can automatically test the decision logic matrix functions and the actuation devices while the unit is at power. When any one train is taken out of service for testing, the other train is capable of providing unit monitoring and protection until the testing has been completed. The testing device is semiautomatic to minimize testing time.

The actuation of ESF components is accomplished through master and slave relays. The SSPS energizes the master relays appropriate for the condition of the plant. Each master relay then energizes one or more slave relays, which then cause actuation of the end devices. The master and slave relays are routinely tested to ensure operation. The test of the master relays energizes the relay, which then operates the contacts and applies a low voltage to the associated slave relays. The low voltage is not sufficient to actuate the slave relays but only demonstrates signal path continuity. The SLAVE RELAY TEST actuates the devices if their operation will not interfere with continued unit operation. For the latter case, actual component operation is prevented by the SLAVE RELAY TEST circuit, and slave relay contact operation is verified by a continuity check of the circuit containing the slave relay.]

Note

No one plant ESFAS incorporates all of the functions listed in Table 3.3.2-1. In some cases (e.g., Containment Pressure--High 3, Function 2.c), the table reflects several different implementations of the same function. Typically, only one of these implementations are used at any specific plant. These different implementations of the same function are denoted by not assigning separate function numbers to the different implementations.

APPLICABLE SAFETY ANALYSES

Each of the analyzed accidents can be detected by one or more ESFAS functions. One of the ESFAS functions is the primary actuation signal for that accident. An ESFAS function may be the primary actuation signal for more than one type of accident. An ESFAS function may also be a secondary, or backup, actuation signal for one or more other accidents. For example, Pressurizer Pressure--Low is a

(continued)

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

primary actuation signal for small loss-of-coolant accidents (LOCAs) and a backup actuation signal for steam line breaks (SLBs) outside containment. Functions such as manual initiation not specifically credited in the accident safety analysis are qualitatively credited in the safety analysis and the NRC staff-approved licensing basis for the plant. These functions may provide protection for conditions which do not require dynamic transient analysis to demonstrate function performance. These functions may also serve as backups to functions that were credited in the accident analysis (Ref. 3).

The required channels of ESFAS instrumentation provide plant protection in the event of any of the analyzed accidents. ESFAS protective functions are as follows:

1. Safety Injection

Safety Injection (SI) provides two primary functions:

1. Primary-side water addition to ensure maintenance or recovery of reactor vessel water level (coverage of the active fuel for heat removal and clad integrity, peak clad temperature < 2200°F); and

2. Boration to ensure recovery and maintenance of SHUTDOWN MARGIN (k_{eff} < 1.0).

These functions are necessary to mitigate the effects of high energy line breaks (HELBs) both inside and outside of containment. The SI signal is also used to initiate other functions such as:

- Phase A Isolation;
- Containment Purge Isolation;
- Reactor Trip;
- Turbine Trip;
- Feedwater Isolation;
- Start of motor-driven auxiliary feedwater (AFW) pumps;

(continued)

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

- Control room ventilation isolation; and
- Enable automatic switchover of Emergency Core Cooling System (ECCS) suction to containment sump.

These other functions ensure:

- Isolation of nonessential systems through containment penetrations;
- Trip of the turbine and reactor to limit power generation;
- Isolation of main feedwater to limit secondary-side mass losses;
- Start of AFW to ensure secondary-side cooling capability;
- Isolation of the control room to ensure habitability; and
- Enabling ECCS suction from the refueling water storage tank (RWST) switchover on low low RWST level to ensure continued cooling via use of the containment sump.

2. Containment Spray

Containment Spray System provides 3 primary functions:

1. Lower containment pressure and temperature after an HELB in containment;

2. Reduce the amount of radioactive iodine in the containment atmosphere; and

3. Adjust the pH of the water in the containment recirculation sump after a large-break LOCA.

These functions are necessary to:

- Ensure the pressure boundary integrity of the containment structure;
- Limit the release of radioactive iodine to the environment in the event of a failure of the containment structure; and
- Minimize corrosion of the components and systems inside containment following a LOCA.

(continued)

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

Actuation of containment spray starts the containment spray pumps and aligns the discharge of the pumps to the containment spray nozzle headers in the upper levels of containment. Water is initially drawn from the RWST by the containment spray pumps and mixed with a sodium hydroxide solution from the spray additive tank. When the RWST reaches the low low level setpoint, the spray pump suctions are shifted to the containment sump if continued containment spray is required. Containment spray is actuated manually, by Containment Pressure--High 3 or Containment Pressure--High High.

3. Containment Isolation

Containment Isolation provides isolation of the containment atmosphere, and all process systems which penetrate containment, from the environment. This function is necessary to prevent or limit the release of radioactivity to the environment in the event of a large-break LOCA.

There are two separate Containment Isolation signals, phase A and phase B. Phase A isolation isolates all automatically isolable process lines except containment cooling water at a relatively low containment pressure indicative of primary or secondary system leaks. For these types of events, forced circulation cooling using the reactor coolant pumps (RCPs) and steam generators (SGs) is the preferred (but not required) mode of decay heat removal. Since containment cooling water is required to support RCP operation, not isolating containment cooling water on the low pressure phase A signal enhances plant safety by allowing operators to use forced RCS circulation to cool the plant. Isolating containment cooling water on the low pressure signal may force the use of feed-and-bleed cooling, which could prove more difficult to control.

Phase A containment isolation is actuated by safety injection, automatic actuation logic, or manually. All process lines penetrating containment with the exception of component cooling water (CCW) are

(continued)

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

isolated. CCW is not isolated at this time to permit continued operation of the RCPs with cooling water flow to the thermal barrier heat exchangers and air or oil coolers. All process lines not equipped with remote operated isolation valves are manually closed or otherwise isolated prior to reaching MODE 4.

Containment cooling water is isolated by the phase B signal, which occurs at a relatively high containment pressure that is indicative of a large-break LOCA or an SLB. For these events, the forced circulation using the RCPs is no longer desirable. Isolating the CCW at the higher pressure does not pose a challenge to the containment boundary, because the CCW System is a closed loop inside containment. Although some system components do not meet all of the American Society of Mechanical Engineers (ASME) Code requirements applied to the containment itself, the system is continuously pressurized to a pressure greater than the phase B setpoint. Thus, routine operation demonstrates the integrity of the system pressure boundary for pressures exceeding the phase B setpoint. Furthermore, because system pressure exceeds the phase B setpoint, any system leakage prior to initiation of phase B isolation would be into containment. Therefore, the combination of CCW System design and phase B isolation ensures the CCW System is not a potential path for radioactive release from containment.

Containment Isolation is accomplished by either of two switches in the control room. Either switch actuates both trains. Note that manual actuation of phase A containment isolation also actuates Containment Purge Isolation.

Phase B containment isolation is actuated by Containment Pressure--High 3 or Containment Pressure--High High, automatic actuation logic, or manually, as previously discussed. For containment pressure to reach a value high enough to actuate Containment Pressure--High 3 or Containment Pressure--High High, a large-break LOCA or SLB must have occurred and containment spray must have been actuated. RCP operation will no longer be required and CCW to the RCPs is, therefore, no longer necessary. The RCPs can

(continued)

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

be operated with seal injection flow alone and without CCW flow to the thermal barrier heat exchanger.

4. Containment Purge Isolation

Containment Purge Isolation closes the containment isolation valves in the Mini-Purge System and the Shutdown Purge System. This action isolates the containment atmosphere from the environment to minimize releases of radioactivity in the event of an accident. The Mini-Purge System may be in use during reactor operation and the Shutdown Purge System will be in use with the reactor shutdown.

5. Steam Line Isolation

Isolation of the main steam lines provides protection in the event of an SLB inside or outside containment. Rapid isolation of the steam lines will limit the steam break accident to the blowdown from one SG at most. For an SLB upstream of the isolation valves, inside or outside of containment, closure of the isolation valves limits the accident to the blowdown from only the affected SG. For an SLB downstream of the isolation valves, closure of the isolation valves terminates the accident as soon as the steam lines depressurize. For plants that do not have steam line check valves, Steam Line Isolation also mitigates the effects of a feed line break and ensures a source of steam for the turbine-driven AFW pump during a feed line break.

6. Turbine Trip and Feedwater Isolation

The primary functions of the Turbine Trip and Feedwater Isolation are to prevent damage to the turbine due to water in the steam lines and to stop the excessive flow of feedwater into the SGs. These functions are necessary to mitigate the effects of a high water level in the SGs, which could result in carryover of water into the steam lines and excessive cooldown of the primary system. The SG high water level is due to excessive feedwater flows.

(continued)

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

7. Auxiliary Feedwater

The AFW System is designed to provide a secondary-side heat sink for the reactor in the event that the Main Feedwater System is not available. The system has two motor-driven pumps and a turbine-driven pump, making it available during normal plant operation, during a loss of AC power, a loss of main feedwater, and during a Feedwater System pipe break. The normal source of water for the AFW System is the condensate storage tank (CST) (normally nonsafety-related). A low level in the CST will automatically realign the pump suctions to the Essential Service Water System (safety-related). The AFW System is aligned so that upon a pump start, flow is initiated to the respective SGs immediately.

8. Automatic Switchover to Containment Sump

At the end of the injection phase of a LOCA, the RWST will be nearly empty. Continued cooling must be provided by the ECCS to remove decay heat. The source of water for the ECCS pumps is automatically switched to the containment recirculation sump. The low head residual heat removal (RHR) pumps [and containment spray pumps] draw the water from the containment recirculation sump, the RHR pumps pump the water through the RHR heat exchanger, inject the water back into the RCS, and supply the cooled water to the other ECCS pumps. Switchover from RWST to containment sump must occur before the RWST empties to prevent damage to the RHR pumps and a loss of core cooling capability. For similar reasons, switchover must not occur before there is sufficient water in the containment sump to support ESF pump suction. Furthermore, early switchover must not occur to ensure sufficient borated water is injected from the RWST to ensure the reactor remains shutdown in the recirculation mode.

9. Control Room Emergency Ventilation

The control room must be kept habitable for the operators stationed there during accident recovery and post-accident operations. The control room is

(continued)

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

equipped with its own ventilation system to provide a habitable environment for the operators. During normal operation, control room ventilation is provided by the Auxiliary Building Ventilation System. During an accident, the control room is isolated and the Control Room Emergency Ventilation System is put into operation.

10. Engineered Safety Feature Actuation System Interlocks

To allow some flexibility in unit operations, several interlocks are included as part of the ESFAS. These interlocks permit the operator to block some signals, automatically enable other signals, prevent some actions from occurring, and cause other actions to occur. The interlock functions backup manual actions to ensure bypassable functions are in operation under the conditions assumed in the safety analyses. The LCO section discusses the basis for allowing bypass of individual functions.

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

LCO

The LCO requires all instrumentation performing an ESFAS function to be OPERABLE. Failure of any instrument renders the affected channel(s) inoperable and reduces the reliability of the affected functions.

Only the ALLOWABLE VALUES are specified for each ESFAS trip function in the LCO. Nominal trip setpoints are specified in the plant-specific setpoint calculations. The nominal setpoints are selected to ensure the setpoint measurements by ANALOG CHANNEL OPERATIONAL 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

(continued)

(continued)

BASES (continued)

LCO (continued)

instrument uncertainties appropriate to the trip function. These uncertainties are defined in the plant-specific setpoint methodology (Ref. 5).

The LCO generally requires OPERABILITY of four or three channels in each instrumentation function and two channels in each logic and manual initiation function. Four OPERABLE instrumentation channels in a two-out-of-four configuration are required when one ESFAS channel is also used as a control system input. This configuration accounts for the possibility of the shared channel failing in a manner that creates a transient which requires ESFAS action. In this case, the ESFAS will still provide protection even with random failure of one of the other three protection channels. Three OPERABLE instrumentation channels in a two-out-of-three configuration are generally required when there is no potential for control system and protection system interaction that could simultaneously create a need for ESFAS initiation and disable one ESFAS channel. The two-out-of-three and the two-out-of-four configurations allow one channel to be tripped during maintenance or testing without causing an ESFAS initiation. Two logic or manual initiation channels are required to ensure no single random failure disables the ESFAS.

The ESFAS actuation channels are considered OPERABLE when:

1. All channel components necessary to provide an ESFAS 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 is not enabled except under the conditions specified by the LCO Applicability for the function.

The following bases for each ESFAS function identify items above that are applicable to establish each function's OPERABILITY.

(continued)

(continued)

BASES (continued)

LCO (continued)

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

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

[For this facility, the supported systems impacted by the inoperability of the ESFAS instrumentation and the justification of whether or not each supported system is declared inoperable are as follows:]

The bases for the LCO on ESFAS functions are:

1. <u>Safety Injection</u>

1.a. <u>Manual Initiation</u>

The LCO requires one channel per train OPERABLE. The operator can initiate SI at any time by using either of two switches in the control room. This action will cause actuation of all components in the same manner as any of the automatic actuation signals.

The LCO on Manual Initiation ensures the proper amount of redundancy is maintained in the manual ESFAS actuation circuitry to ensure the operator has manual ESFAS initiation capability. Manual Initiation channels are OPERABLE when they satisfy OPERABILITY requirements 1 and 3.

Each channel consists of one push button and the interconnecting wiring to the actuation logic cabinet. Each push button is responsible for initiating one train of SI. However, inadvertent pressing of one button will cause one train of actuation. This configuration does not allow testing at power.

(continued)

(continued)

BASES (continued)

LCO (continued)

1.b. Automatic Actuation Logic and Actuation Relays

This LCO requires two trains OPERABLE. Actuation logic consists of all circuitry housed within the actuation subsystems, including the initiating relay contacts responsible for actuating the ESF equipment. Automatic Actuation Logic and Actuation Relays are OPERABLE when they satisfy OPERABILITY requirements 1 and 3.

1.c. Containment Pressure--High 1

This signal provides protection against the following accidents:

- SLB inside containment;
- LOCA; and
- Feed line break inside containment.

Containment Pressure--High 1 provides no input to any control functions. Thus, three OPERABLE channels are sufficient to satisfy protective requirements with a two-out-of-three logic. The transmitters (d/p cells) and electronics are located outside of containment with the sensing line (high pressure side of the transmitter) located inside containment. Containment Pressure--High 1 channels are OPERABLE when they satisfy OPERABILITY requirements 1, 2, and 3.

The containment pressure [transmitters are located just outside of containment and may be subjected to high radiation conditions during the accidents which they are intended to mitigate. The sensor portion of the transmitters are also exposed to the steam environment present in the containment following a LOCA or HELB. Therefore, the trip setpoint ALLOWABLE VALUE accounts for measurement errors induced by these environments.]

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

(continued)

(continued)

BASES (continued)

LCO (continued)

1.d. Pressurizer Pressure--Low

This signal provides protection against the following accidents:

- Inadvertent opening of an SG relief or safety valve;
- SLB;
- A Spectrum of rod cluster control assembly ejection accidents (rod ejection), and
- Inadvertent opening of a pressurizer relief or safety valve.

At some plants pressurizer pressure provides both control and protection functions: input to the Pressurizer Pressure Control System, reactor trip, and SI. Therefore, the actuation logic must be able to withstand both an input failure to the control system (which may then require the protection function actuation) and a single failure in the other channels providing the protection function actuation. Thus four OPERABLE channels are required to satisfy the requirements with a two-out-of-four logic. For units that have dedicated protection and control channels, only three protection channels are necessary to satisfy the protective requirements. Pressurizer Pressure--Low channels are OPERABLE when they satisfy OPERABILITY requirements 1, 2, 3, and 4. The transmitters are located inside containment, with the taps in the vapor space region of the pressurizer, and thus possibly experiencing adverse environmental conditions (LOCA, SLB inside containment, rod ejection). Therefore, the trip setpoint reflects the inclusion of both steady-state and adverse environmental instrument uncertainties.

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

(continued)

(continued)

BASES (continued)

LCO (continued)

1.e Steam Line Pressure

1.e.(1). Steam Line Pressure Low--Setpoint

Steam Line Pressure--Low provides protection against the following accidents:

- SLB;
- Feed line break; and
- Inadvertent opening of an SG relief or an SG safety valve.

Steam Line Pressure--Low provides no input to any control functions. Thus, three OPERABLE channels on each steam line are sufficient to satisfy the protective requirements with a two-out-of-three logic on each steam line. Steam Line Pressure--Low setpoint channels are OPERABLE when they satisfy OPERABILITY requirements 1, 2, 3, and 4. With the transmitters typically located inside the steam tunnels, it is possible for them to experience adverse environmental conditions during a secondary-side break. Therefore, the trip setpoint reflects both steady-state and adverse environmental instrument uncertainties.

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

This function is anticipatory in nature and has a typical lead/lag ratio of 50/5. [For this facility, the impact of this ratio on channel OPERABILITY is as follows:]

1.e.(2). Steam Line Pressure--High Differential Pressure Between Steam Lines

Steam Line High Differential Pressure provides protection against the following accidents:

- SLB;
- Feed line break; and

(continued)

(continued)

BASES (continued)

LCO (continued)

- Inadvertent opening of an SG relief or an SG safety valve.

Steam Line High Differential Pressure provides no input to any control functions. Thus three OPERABLE channels on each steam line are sufficient to satisfy the requirements with a two-out-of-three logic on each steam line. Steam Line High Differential Pressure channels are OPERABLE when they satisfy OPERABILITY requirements 1, 2, and 3. With the transmitters typically located inside the steam tunnels, it is possible for them to experience adverse environmental conditions during an SLB event. Therefore, the trip setpoint reflects both steady-state and adverse environmental instrument uncertainties.

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

1.f. High Steam Flow in Two Steam Lines Coincident With T_{avg}--Low Low or Coincident With Steam Line Pressure--Low

This function provides protection against the following accidents:

- SLB; and
- the inadvertent opening of an SG relief or an SG safety valve.

Two steam line flow channels per steam line are required OPERABLE for this function. These are combined in a one-out-of-two logic to indicate high steam flow in one steam line. The steam flow transmitters provide control inputs, but the control function cannot cause the events which the function must protect against. Therefore, two channels are sufficient to satisfy redundancy requirements. The one-out-of-two configuration allows on-line testing because trip of one high steam flow channel is not sufficient to cause

(continued)

(continued)

BASES (continued)

L.O (continued)

initiation. These channels are OPERABLE when they satisfy OPERABILITY requirements 1, 2, 3, and 4. [At this facility, trip does not occur unless high steam flow exists in two steam lines for the following reasons:]

One channel of T_{avg} per loop and one channel of low steam line pressure per steam line are required OPERABLE. For each parameter the channels for all loops or steam lines are combined in a logic such that two channels tripped will cause a trip for the parameter. For example, for three loop plants, the low steam line pressure channels are combined in two-out-of-three logic. Thus, the function trips on one-out-of-two high flow in any two-out-of-three steam lines if there is one-out-of-one low low T_{avg} trip in any two-out-of-three RCS loops or if there is a one-out-of-one low pressure trip in any two-out-of-three steam lines. Since the accidents this event protects against cause both low steam line pressure and low low T_{avg}, provision of one channel per loop or steam line ensures no single random failure can disable both of these functions. The steam line pressure channels provide no control inputs. The T_{avg} channels provide control inputs, but the control function cannot initiate events that the function acts to mitigate.

The high steam flow trip setpoint ALLOWABLE VALUE for high steam flow is a linear function that varies with power level. The function is a ΔP corresponding to 44% of full steam flow between 0% and 20% load to 114% of full steam flow at 100% load. [For this facility, the load reference value is measured and the number of steam flow channels affected by an inoperable load channel is as follows:]. The nominal trip setpoint is similarly calculated.

[For this facility, the basis for the T_{avg}--Low Low ALLOWABLE VALUE is as follows:]

(continued)

(continued)

BASES (continued)

LCO (continued)

With the transmitters typically located inside the containment, it is possible for them to experience adverse steady-state environmental conditions during an SLB event. Therefore, the trip setpoint reflects both steady-state andadverse environmental instrument uncertainties. The Steam Line Pressure--Low signal was discussed previously under Function 1.e.1.

2. Containment Spray

2.a. Manual Initiation

The operator can initiate containment spray at any time from the control room by simultaneously turning two containment spray actuation switches in the same train. Because an inadvertent actuation of containment spray could have such serious consequences, two switches must be turned simultaneously to initiate containment spray. There are two sets of two switches each in the control room. Simultaneously turning the two switches in either set will actuate containment spray in both trains in the same manner as the automatic actuation signal. Two Manual Initiation switches in each train are required OPERABLE to ensure no single failure disables the manual initiator function. Manual Initiation channels are OPERABLE when they satisfy OPERABILITY requirements 1 and 3. Note that Manual Initiation of containment spray also actuates phase B containment isolation.

2.b. Automatic Actuation Logic and Actuation Relays

Automatic actuation logic and actuation relays consist of same features and operate in the same manner as described for ESFAS Function 1.b.

2.c. Containment Pressure

This signal provides protection against a LOCA or an SLB inside containment. Containment Pressure channels are OPERABLE when they satisfy

(continued)

(continued)

BASES (continued)

LCO (continued)

OPERABILITY requirements 1, 2, and 3. [For this facility, the basis for ALLOWABLE VALUE for the Containment Spray function is as follows:] The transmitters (d/p cells) are located outside of containment with the sensing line (high pressure side of the transmitter) located inside containment. The containment pressure [transmitters may be subjected to high radiation conditions during the accidents which they are intended to mitigate. The sensor portion of the transmitters are also exposed to the steam environment present in the containment following a LOCA or HELB. Therefore, the trip setpoint ALLOWABLE VALUE accounts for the measurement errors induced by the environments.] This is one of the only functions that requires the bistable output to energize to perform its required action. It is not desirable to have a loss of power actuate containment spray, since the consequences of an inadvertent actuation of containment spray could be serious. Note that this function also has the inoperable channel placed in bypass rather than trip to decrease the probability of an inadvertent actuation.

Two different logic configurations are typically used. Three and four loop plants typically use four channels in a two-out-of-four logic configuration. This configuration may be called the Containment Pressure--High 3 Setpoint for three and four loop plants and Containment Pressure--High High Setpoint for other plants. Some two loop plants use three sets of two channels, each set combined in a one-out-of-two configuration, and these outputs combined so that two-out-three sets tripped initiates containment spray. This configuration is typically called Containment Pressure--High 3 Setpoint. Since containment pressure is not used for control, both of these arrangements exceed the minimum redundancy requirements. Additional redundancy is warranted because the function is energize-to-trip.

(continued)

(continued)

BASES (continued)

LCO (continued)

3. Containment Isolation

3.a. Phase A Isolation

3.a.(1). Manual Initiation

Manual Phase A Containment Isolation is accomplished by either of two switches in the control room. Either switch actuates both trains. Manual Initiation channels are OPERABLE when they satisfy OPERABILITY requirements 1 and 3. Note that manual initiation of Phase A Containment Isolation also actuates Containment Purge Isolation.

3.a.(2). Automatic Actuation Logic and Actuation Relays

Automatic Actuation Logic and Actuation Relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b.

3.a.(3). Safety Injection

Containment Phase A Isolation is also initiated by all functions that initiate SI. The Containment Phase A Isolation function requirements for these functions are the same as the requirements for their SI function. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead, Function 1, SI, is referenced for all initiating functions and requirements.

3.b. Phase B Isolation

Phase B Isolation is accomplished by the Manual Initiation, Automatic Actuation Logic and Actuation Relays, and by Containment Pressure channels. The same channels that actuate Containment Spray, Function 2. The Containment Pressure trip of Phase B Isolation is energized to trip to minimize the potential of spurious trips that may damage the RCPs. {Currently no

(continued)

(continued)

BASES (continued)

LCO (continued)

plants using three sets of two pressure channels for the High 3 function utilize Phase B Isolation.)

4. Containment Purge Isolation

4.a. Manual Initiation

The LCO requires one channel per train OPERABLE. Manual Initiation channels are OPERABLE when they satisfy OPERABILITY requirements 1 and 3. The operator can initiate Containment Purge Isolation at any time by using either of two switches in the control room.

This action will cause actuation of all components in the same manner as any of the automatic actuation signals.

The LCO on Manual Initiation ensures the proper amount of redundancy is maintained in the manual ESFAS actuation circuitry to ensure the operator has manual ESFAS initiation capability.

Each channel consists of one push button and the interconnecting wiring to the actuation logic cabinet. Each push button is responsible for initiating one train of Containment Purge Isolation. However, inadvertent pressing of one button will cause one train of actuation. This configuration does not allow testing at power.

4.b. Automatic Actuation Logic and Actuation Relays

Automatic actuation logic and actuation relays consist of same features and operate in the same manner as described for ESFAS Function 1.b.

4.c. Safety Injection

4.d. Phase A Isolation

Containment purge isolation is also initiated by all functions that initiate Safety Injection or containment phase A isolation. Table 3.3.2-1

(continued)

(continued)

BASES (continued)

LCO (continued)

references Function 1, SI, and Function 3.a, containment phase A isolation, for description of the initiating functions and the requirements on Required Channels, Surveillances, and ALLOWABLE VALUES for these functions. The Applicable MODES and conditions specified for the containment purge isolation portion of these functions are different and less restrictive than those for their phase A isolation and SI roles. If one or more of the SI or phase A isolation functions becomes inoperable in such a manner that only the Containment Purge Isolation function is affected, the Conditions applicable to their SI and phase A isolation functions need not be entered. The less restrictive Actions specified for inoperability of the Containment Purge Isolation functions specify sufficient compensatory measures for this case. Function 1, SI, is referenced for all initiating functions and requirements.

4.e. <u>Containment Radiation</u>

Containment Purge Isolation is initiated by high containment radiation. The instrument channels providing these signals are inputs to, but not formally part of, the ESFAS. LCO 3.3.7 provides the requirements on the containment radiation instrumentation that initiate containment purge isolation.

5. <u>Steam Line Isolation</u>

5.a. <u>Manual Initiation</u>

Manual initiation of Steam Line Isolation can be accomplished from the control room. There are two switches in the control room and either switch can initiate action to immediately close all main steam isolation valves (MSIVs). The LCO requires one OPERABLE channel in each train. Manual Initiation channels are OPERABLE when they satisfy OPERABILITY requirements 1 and 3.

(continued)

(continued)

BASES (continued)

LCO (continued)

5.b. Automatic Actuation Logic and Actuation Relays

Automatic actuation logic and actuation relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b.

5.c. Containment Pressure--High 2

This function actuates closure of the MSIVs in the event of a LOCA or an SLB inside containment to maintain at least one unfaulted SG as a heat sink for the reactor and to limit the mass and energy release to containment. The transmitters (d/p cells) are located outside containment with the sensing line (high pressure side of the transmitter) located inside containment. Containment Pressure--High 2 provides no input to any control functions. Thus, three OPERABLE channels are sufficient to satisfy protective requirements with two-out-of-three logic. However, for enhanced reliability, this function was designed with four channels and two-out-of-four logic. Containment Pressure--High 2 channels are OPERABLE when they satisfy OPERABILITY requirements 1, 2, and 3. [For this facility, the basis for ALLOWABLE VALUE is as follows:]

The containment pressure [transmitters are located just outside of containment and may be subjected to high radiation conditions during the accidents which they are intended to mitigate. The sensor portion of the transmitters are also exposed to the steam environment present in the containment following a LOCA or HELB. Therefore, the trip setpoint ALLOWABLE VALUE accounts for measurement errors induced by these environments.]

5.d. Steam Line Pressure

5.d.(1). Steam Line Pressure--Low Setpoint

Steam Line Pressure--Low provides closure of the MSIVs in the event of an SLB to maintain

(continued)

(continued)

BASES (continued)

LCO
(continued)

at least one unfaulted SG as a heat sink for the reactor and to limit the mass and energy release to containment. This function provides closure of the MSIVs in the event of a feed line break to ensure a supply of steam for the turbine-driven AFW pump. Steam Line Pressure--Low was discussed previously under SI function 1.e.1. [For this facility, the basis for ALLOWABLE VALUE for the Steam Line Isolation function is as follows:]

5.d.(2). Steam Line Pressure-Negative Rate--High Setpoint

Steam Line Pressure-Negative Rate--High provides closure of the MSIVs for an SLB, when less than the P-11 setpoint, to maintain at least one unfaulted SG as a heat sink for the reactor and to limit the mass and energy release to containment. When the operator manually blocks the Steam Line Pressure--Low main steam isolation signal when less than the P-11 setpoint, the Steam Line Pressure-Negative Rate--High signal is automatically enabled. Steam Line Pressure-Negative Rate--High provides no input to any control functions. Thus, three OPERABLE channels are sufficient to satisfy requirements with a two-out-of-three logic, on each steam line. Steam Line Pressure-Negative Rate--High setpoint channels are OPERABLE when they satisfy OPERABILITY requirements 1, 2, 3, and 4. [For this facility, the basis for ALLOWABLE VALUE is as follows:]

With the transmitters typically located inside the steam tunnels, it is possible for them to experience adverse environmental conditions during an SLB event. Therefore, the trip setpoint reflects both steady-state and adverse environmental instrument uncertainties.

(continued)

(continued)

BASES (continued)

LCO (continued)

5.e. High Steam Flow in Two Steam Lines Coincident with T_{avg}--Low-Low or Coincident With Steam Line Pressure--Low

This function provides closure of the MSIVs during an SLB or inadvertent opening of an SG relief or a safety valve to maintain at least one unfaulted SG as a heat sink for the reactor and to limit the mass and energy release to containment.

This function was discussed previously as Function 1.f. [For this facility, the basis for the high steam line flow, T_{avg}--Low Low, and Steam Line Pressure--Low ALLOWABLE VALUES for the Steam Line Isolation function is as follows:]

5.f. High Steam Flow Coincident With Safety Injection and Coincident With T_{avg}--Low Low

This function provides closure of the MSIVs during an SLB or inadvertent opening of an SG relief or safety valve to maintain at least one unfaulted SG as a heat sink for the reactor and to limit the mass and energy release to containment.

Two steam line flow channels per steam line are required OPERABLE for this function. These are combined in a one-out-of-two logic to indicate high steam flow in one steam line. The steam flow transmitters provide control inputs, but the control function cannot cause the events which the function must protect against. Therefore, two channels are sufficient to satisfy redundancy requirements. The one-out-of-two configuration allows on-line testing because trip of one high steam flow channel is not sufficient to cause initiation. The High Steam Flow channels are OPERABLE when they satisfy OPERABILITY requirements 1, 2, and 3.

The High Steam Flow ALLOWABLE VALUE is a ΔP corresponding to 25% of full steam flow at no-load steam pressure.

(continued)

(continued)

BASES (continued)

LCO (continued)

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

With the transmitters (d/p cells) typically located inside the steam tunnels, it is possible for them to experience adverse environmental conditions during an SLB event. Therefore, the trip setpoint reflects both steady-state and adverse environmental instrument uncertainties.

Main steam line isolates only if the high steam flow signal occurs coincident with an SI and low low RCS average temperature. The main Steam Line Isolation function requirements for the SI functions are the same as the requirements for their SI function. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead Function 1, SI, is referenced for all initiating functions and requirements.

One channel of T_{avg} per loop is required OPERABLE. The T_{avg} channels for all loops are combined in a logic such that two channels tripped cause a trip for the parameter. For example, for three loop plants, the T_{avg} channels are combined in two-out-of-three logic. Thus, the function trips on one-out-of-two high flow in any two steam lines if this is a one-out-of-one low low T_{avg} trip in any two-out-of-three RCS loops and an SI occurs. The accidents this function protects against cause reduction of T_{avg} in the entire primary system. Therefore, the provision of one OPERABLE channel per loop in a two-out-of-three or two-out-of-four configuration ensures no single random failure disables the T_{avg}--Low Low function. The T_{avg} channels provide control inputs, but the control function cannot initiate events that the function acts to mitigate. Therefore, additional channels are not required to address control-protection interaction issues.

With the transmitters typically located inside the containment, it is possible for them to experience adverse environmental conditions during an SLB event. Therefore, the trip

(continued)

(continued)

BASES (continued)

LCO (continued)

setpoint reflects both steady-state and adverse environmental instrumental uncertainties. [For this facility, the basis for the T_{avg} ALLOWABLE VALUE for the main Steam Line Isolation function is as follows:]

5.g. <u>High High Steam Flow In Two Steam Lines Coincident With Safety Injection</u>

[At this facility, the Bases for LCO requirements on Function 5.g are as follows:]

5.h. <u>High High Steam Flow in Two Steam Lines Coincident With T_{avg}--Low Low</u>

This function is the same as Function 5.g above, except that the T_{avg}--Low Low confirmatory signal is required. Operation of the T_{avg}--Low Low portion of this function is described as part of the discussion of Function 5.f. The function is slightly less susceptible to spurious steam line isolation, but otherwise performs in the same manner.

Plants generally have only one of the Functions 5.e, 5.f, 5.g, or 5.h. No plants have Function 5.f in combination with either 5.g or 5.h.

6. <u>Turbine Trip and Feedwater Isolation</u>

This function is actuated by Steam Generator Water Level--High High or by an SI signal. The Reactor Trip System also initiates a turbine trip signal whenever a reactor trip (P-4) is generated. In the event of SI, the unit is taken off line and the turbine generator must be tripped. The MFW System is also taken out of operation and the AFW System is automatically started. The SI signal was discussed previously.

(continued)

(continued)

BASES (continued)

LCO (continued)

6.a. Automatic Actuation Logic and Actuation Relays

Automatic Actuation Logic and Actuation Relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b.

6.b. Steam Generator Water Level--High High (P-14)

This signal provides protection against excessive feedwater flow. At some plants the ESFAS SG water level instruments provide input to the SG Water Level Control System. Therefore, the actuation logic must be able to withstand both an input failure to the control system (which may then require the protection function actuation) and a single failure in the other channels providing the protection function actuation. Thus four OPERABLE channels are required to satisfy the requirements with a two-out-of-four logic. These channels are OPERABLE when they satisfy OPERABILITY requirements 1, 2, and 3. For units that have dedicated protection and control channels, only three protection channels are necessary to satisfy the protective requirements. [For this facility, the basis for ALLOWABLE VALUE is as follows:] The transmitters (d/p cells) are located inside containment, however, the events which this function protect against cannot cause severe environment in containment, therefore, the trip setpoint reflects only steady-state instrument uncertainties.

6.c. Safety Injection

Turbine trip and isolation is also initiated by all functions that initiate SI. The feedwater isolation function requirements for these functions are the same as the requirements for their SI function. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead Function 1, SI, is referenced for all initiating functions and requirements.

(continued)

(continued)

BASES (continued)

LCO (continued)

7. Auxiliary Feedwater

7.a. Automatic Actuation Logic and Actuation Relays [Solid State Protection System]

Automatic actuation logic and actuation relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b.

7.b. Automatic Actuation Logic and Actuation Relays (Balance of Plant ESFAS)

Automatic actuation logic and actuation relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b.

7.c. Steam Generator Water Level--Low Low

SG Water Level--Low Low provides protection against a loss of heat sink. A feedwater system pipe break, inside or outside of containment, or a loss of main feedwater would result in a loss of SG water level. SG Water Level--Low Low provides input to the SG Level Control System. Therefore, the actuation logic must be able to withstand both an input failure to the control system which may then require the protective function actuation and a single failure in the other channels providing the protection function actuation. Thus, four OPERABLE channels are required to satisfy the requirements with two-out-of-four logic. These channels are OPERABLE when they satisfy OPERABILITY requirements 1, 2, and 3. For units that have dedicated protection and control channels, only three protection channels are necessary to satisfy the protective requirements. [For this facility, the basis for ALLOWABLE VALUE is as follows:] With the transmitters (d/p cells) located inside containment and thus possibly experiencing adverse environmental conditions (feed line break), the trip setpoint reflects the inclusion of both steady-state and adverse environmental instrument uncertainties.

(continued)

(continued)

BASES (continued)

LCO (continued)

7.d. Safety Injection

An SI signal starts the motor-driven and turbine-driven AFW pumps. The AFW initiation functions are the same as the requirements for their SI function. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead, Function 1, SI, is referenced for all initiating functions and requirements.

7.e. Loss of Offsite Power

A loss of offsite power provides indication of a loss of all AC power. A loss of offsite power to the service buses will be accompanied by a loss of reactor coolant pumping power and the subsequent need for some method of decay heat removal. The loss of offsite power is detected by a voltage drop on each service bus. Loss of power to either service bus will start the turbine-driven AFW pumps to ensure that at least one SG contains enough water to serve as the heat sink for reactor decay heat and sensible heat removal following the reactor trip. Loss of offsite power channels are OPERABLE when they satisfy OPERABILITY requirements 1, 2, and 3. [For this facility, describe relay configuration and how trip meets single failure criterion for single-phasing events. If action of other relays (e.g., phase differential current) is required to ensure trip in the event of single-phasing, discuss how OPERABILITY of these other relays affects OPERABILITY of this function and identify the LCO covering the other relays]. [For this facility, the basis for ALLOWABLE VALUE is as follows:]

7.f. Undervoltage Reactor Coolant Pump

A loss of power on the buses that provide power to the RCPs provides indication of a pending loss of RCP forced flow in the RCS. The Undervoltage Reactor Coolant Pump function senses the voltage downstream of each RCP breaker. A loss of power or open RCP breaker on two or more RCPs will

(continued)

(continued)

BASES (continued)

LCO (continued)

start the turbine-driven AFW pump to ensure that at least one SG contains enough water to serve as the heat sink for reactor decay heat and sensible heat removal following the reactor trip. Undervoltage RCP channels are OPERABLE when they satisfy OPERABILITY requirements 1, 2, and 3. [For this facility, describe the relay configuration and how trip meets single failure criterion for single-phasing events. If action of other relays (e.g., phase differential current) is required to ensure trip in the event of single-phasing, discuss how OPERABILITY of these other relays affects OPERABILITY of this function and identify the LCO covering the other relays]. [For this facility, the basis for ALLOWABLE VALUES is as follows:]

7.g. Trip Of All Main Feedwater Pumps

A Trip of all main feedwater pumps is an indication of a loss of main feedwater and the subsequent need for some method of decay heat and sensible heat removal to bring the reactor back to no-load temperature and pressure. A turbine-driven main feedwater pump is equipped with two pressure switches on the control air line for the speed control system. A low pressure signal from either of these pressure switches indicates a trip of that pump. Motor-driven main feedwater pumps are equipped with a breaker position sensing device. An open supply breaker indicates that the pump is not running. Two OPERABLE channels per pump satisfy redundancy requirements with one-out-of-two taken twice logic. These channels are OPERABLE when they satisfy OPERABILITY requirements 1, 2, and 3. A trip of all main feedwater pumps starts the motor-driven and turbine-driven AFW pumps to ensure that at least one SG is available with water to act as the heat sink for the reactor. [For this facility, the basis for the ALLOWABLE VALUE is as follows:]

(continued)

(continued)

BASES (continued)

LCO (continued)

7.h. Auxiliary Feedwater Pump Suction Transfer on Suction Pressure--Low

A low pressure on the AFW pump suction protects the AFW pumps against a loss of the normal supply of water for the pumps, the CST. Two pressure switches are located on the AFW pump suction line from the CST. A low pressure sensed by any one of the switches will cause the emergency supply of water for both pumps to be aligned or cause the AFW pumps to stop until the emergency source of water is aligned. Essential service water (safety grade) is then lined up to supply the AFW pumps to ensure an adequate supply of water for the AFW System to maintain at least one of the SGs as the heat sink for reactor decay heat and sensible heat removal. AFW Pump Suction Pressure--Low channels are OPERABLE when they satisfy OPERABILITY requirements 1, 2, and 3. [For this facility, the basis for ALLOWABLE VALUE is as follows:] Since the detectors are located in area not affected by HELBs or high radiation, they will not experience any adverse environmental conditions and the trip setpoint reflects only steady-state instrument uncertainties.

8. Automatic Switchover to Containment Sump

8.a. Automatic Actuation Logic and Actuation Relays

Automatic actuation logic and actuation relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b.

8.b. Refueling Water Storage Tank (RWST) Level--Low Low Coincident With Safety Injection

8.c. RWST Level--Low Low Coincident With Safety Injection and Coincident With Containment Sump Level--High

During the injection phase of a LOCA, the RWST is the source of water for all ECCS pumps. A low low level in the RWST coincident with an SI

(continued)

(continued)

BASES (continued)

LCO (continued)

signal provides protection against a loss of water for the ECCS pumps and indicates the end of the injection phase of the LOCA. The RWST is equipped with four level transmitters. These transmitters provide no control functions, therefore, a two-out-of-four logic is adequate to initiate the protective function actuation. Although only three channels would be sufficient, a fourth channel has been added for increased reliability. These channels are OPERABLE when they satisfy OPERABILITY requirements 1, 2, and 3. The RWST--Low Low setpoint ALLOWABLE VALUE has both upper and lower limits. The lower limit is selected to ensure switchover occurs before the RWST empties to prevent ECCS pump damage. The upper limit is selected to ensure enough borated water is injected to ensure the reactor remains shutdown. The high limit also ensures adequate water inventory in the containment sump to provide ECCS pump suction. The transmitters are located in an area not affected by HELBs or post-accident high radiation, thus, they will not experience any adverse environmental conditions and the trip setpoint reflects only steady-state instrument uncertainties.

Automatic switchover occurs only if the RWST low low level signal is coincident with SI. This prevents accidental switchover during normal operation. Accidental switchover could damage ECCS pumps if they are attempting to take suction from an empty pump. The automatic switchover function requirements for the SI functions are the same as the requirements for their SI function. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead, Function 1, SI, is referenced for all initiating functions and requirements.

In some plants, additional protection from spurious switchover is provided by requiring a Containment Sump Level--High signal as well as RWST Level--Low Low and SI.

(continued)

(continued)

BASES (continued)

LCO (continued)

During the injection phase of a LOCA, the RWST is the source of water for all ECCS pumps. As the RWST empties, the water from the RWST and the water from the RCS and accumulators will accumulate inside containment via the break. To ensure that there is sufficient water available in containment to support the recirculation phase of the accident, a Containment Sump Level--High signal must be present, in addition to the SI signal and the RWST Level--Low Low signal, to transfer the suctions of the RHR pumps to the containment sump. The containment sump is equipped with four level transmitters. These transmitters provide no control functions, therefore, a two-out-of-four logic is adequate to initiate the protective function actuation. Although only three channels would be sufficient, a fourth channel has been added for increased reliability. The containment sump level trip ALLOWABLE VALUE is selected to ensure enough borated water is injected to ensure the reactor remains shutdown. The high limit also ensures adequate water inventory in the containment sump to provide ECCS pump suction. The transmitters are located inside containment, and thus possibly experience adverse environmental conditions. Therefore, the trip setpoint reflects the inclusion of both steady-state and environmental instrument uncertainties.

Plants only have one of the Functions 8.b or 8.c.

9. Control Room Emergency Ventilation

Control room isolation consists of automatically positioning the appropriate dampers to isolate the Control Room Emergency Ventilation System from the Auxiliary Building Ventilation System.

9.a. Manual Initiation

The control room operator can manually initiate control room emergency ventilation in both trains at any time by turning either of two switches in

(continued)

(continued)

BASES (continued)

LCO (continued)

the control room. Manual Initiation channels are OPERABLE when they satisfy OPERABILITY requirements 1 and 3.

9.b. Automatic Actuation Logic and Actuation Relays

Automatic actuation logic and actuation relays consist of same features and operate in the same manner as described for ESFAS Function 1.b.

9.c. Phase A Isolation

Control room emergency ventilation initiation is also initiated by all functions that initiate containment phase A isolation. The control room emergency ventilation initiation function requirements for these functions are the same as the requirements for their containment phase A isolation function. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead Function 3.a, containment phase A isolation, is referenced for all initiating functions and requirements.

9.d. Control Room Atmosphere and Air Intake Radiation

Control room emergency ventilation is initiated by high control room air intake radiation. The instrument channels providing these signals are inputs to, but not formally part of, the ESFAS. LCO 3.3.7 provides the requirements on the Control room air intake instrumentation that activates Control Room Emergency Ventilation.

10. Engineered Safety Feature Actuation System Interlocks

To allow some flexibility in unit operations, several interlocks are included as part of the ESFAS. These interlocks permit the operator to block some signals, automatically enable other signals, prevent some actions from occurring, and cause other actions to occur.

(continued)

(continued)

BASES (continued)

LCO
(continued)

10.a. Reactor Trip, P-4

The P-4 interlock is enabled when either reactor trip breakers and its associated bypass breaker are open. Once the P-4 interlock is enabled, automatic SI initiation is blocked after a [ ] second time delay. This function allows operators to take manual control of SI systems after the initial phase of injection is complete. Once SI is blocked, automatic actuation of SI cannot occur until the reactor trip breakers have been manually closed. P-4 interlock channels are OPERABLE when they satisfy OPERABILITY requirements 1 and 3. The functions of the P-4 interlock are:

- Trip the main turbine;
- Isolate with coincident low T_{avg};
- Prevent reactuation of SI after a manual reset of SI;
- Transfer the steam dump from the load rejection controller to the plant trip controller; and
- Prevent opening of the main feedwater isolation valves if they were closed on SI or SG Water Level--High High.

[For this facility:

- Identify which of the above functions are safety related and considered a part of the ESFAS;
- Identify LCO addressing any steam dump controls considered to be ESF;
- Provide basis for required number of channels; and
- Describe whether changes in reset time delay setpoint can affect function OPERABILITY.]

The reactor trip breaker position switches that provide input to the P-4 interlock only function to energize or de-energize or open or

(continued)

(continued)

BASES (continued)

LCO (continued)

close contacts. Therefore, this function has no adjustable trip setpoint with which to associate an ALLOWABLE VALUE.

10.b. Pressurizer Pressure, P-11

The P-11 interlock permits a normal unit cooldown and depressurization without actuation of SI or main steam line isolation. P-11 Interlock channels are OPERABLE when they satisfy OPERABILITY requirements 1, 2, and 3. With two-out-of-three pressurizer pressure channels (discussed previously) less than the P-11 setpoint, the operator can manually block the pressurizer pressure--Low and Steam Line Pressure--Low SI signals and the Steam Line Pressure--Low steam line isolation signal (previously discussed). When the Steam Line Pressure--Low steam line isolation signal is manually blocked, a main steam isolation signal on Steam Line Pressure-Negative Rate--High is enabled. This provides protection for an SLB by closure of the main steam isolation valves. With two-out-of-three pressurizer pressure channels ≥ P-11 setpoint, the Pressurizer Pressure--Low and Steam Line Pressure--Low SI signals and the Steam Line Pressure--Low steam line isolation signal are automatically enabled. The operator can also enable these trips by use of the respective manual reset buttons. When the Steam Line Pressure--Low steam line isolation signal is enabled, the main steam isolation on Steam Line Pressure-Negative Rate--High is disabled. The trip setpoint reflects only steady-state instrument uncertainties. [For this facility, the basis for the ALLOWABLE VALUE is as follows:]

10.c. T_{avg}--Low Low, P-12

On increasing reactor coolant temperature, the P-12 interlock reinstates SI on High Steam Flow Coincident With Steam Line Pressure--Low or Coincident With T_{avg}--Low Low. P-12 Interlock channels are OPERABLE when they satisfy OPERABILITY requirements 1, 2, and 3.

(continued)

(continued)

BASES (continued)

LCO
(continued)

On an increasing temperature, the P-12 interlock also provides an arming signal to the Steam Dump System. On decreasing reactor coolant temperature, the P-12 interlock allows the operator to manually block SI on High Steam Flow Coincident With Steam Line Pressure--Low or Coincident with T_{avg}--Low Low. On a decreasing temperature, the P-12 interlock also removes the arming signal to the Steam Dump System to prevent an excessive cooldown of the RCS due to a malfunctioning Steam Dump System. Since T_{avg} is used as an indication of bulk RCS temperature, this function meets redundancy requirements with one OPERABLE channel in each loop. In three loop plants these channels are used in two-out-of-three logic. In four loop plants they are used in two-out-of-four logic. [For this facility, the basis for the ALLOWABLE VALUE is as follows:]

10.d. Steam Generator Water Level--High High, P-14

The P-14 interlock is actuated when level in any SG exceeds the high high setpoint and performs the following functions as part of Function 6:

- Trips the main turbine;
- Trips the main feedwater pumps;
- Initiates feedwater isolation; and
- Shuts the main feedwater regulating valves and the bypass feedwater regulating valves.

The main feedwater pumps are tripped, feedwater isolation is actuated, and the main and bypass feedwater regulating valves are closed to prevent any further addition of water to the SGs. The main turbine is tripped to prevent carryover of excessive moisture to the turbine, which would damage the turbine. This function has previously been discussed as Function 6.b.

(continued)

BASES (continued)

APPLICABILITY

1. Safety Injection

1.a. Manual Initiation

1.b. Automatic Actuation Logic and Actuation Relays

Manual and automatic initiation of SI must be OPERABLE in MODES 1, 2, and 3. In these MODES there is sufficient energy in the primary and secondary systems to warrant automatic initiation of ESF systems. Manual Initiation is also required in MODE 4 even though automatic actuation is not required. In this MODE, adequate time is available to manually actuate required components in the event of a DBA, but because of the large number of components actuated on a SI, actuation is simplified by the use of the manual actuation push buttons. Automatic actuation logic and actuation relays must be OPERABLE in MODE 4 to support system level manual initiation.

These functions are not required to be OPERABLE in MODES 5 and 6 because there is adequate time for the operator to evaluate plant conditions and respond by manually starting individual systems, pumps, and other equipment to mitigate the consequences of an abnormal condition or accident. Plant pressure and temperature are very low and many ESF components are administratively locked out or otherwise prevented from actuating to prevent inadvertent overpressurization of plant systems.

1.c. Containment Pressure--High 1

Containment Pressure--High 1 must be OPERABLE in MODES 1, 2 and 3 when there is sufficient energy in the primary and secondary systems to pressurize the containment following a pipe break. In MODES 4, 5, and 6, there is insufficient energy in the primary or secondary systems to pressurize the containment.

(continued)

(continued)

BASES (continued)

APPLICABILITY (continued)

1.d. Pressurizer Pressure--Low

This function must be OPERABLE in MODES 1, 2 and 3 (above P-11) to mitigate the consequences of a high energy line rupture inside containment. This signal may be manually blocked by the operator below the P-11 setpoint. Automatic SI actuation below this pressure is then performed by the Containment Pressure--High 1 signal.

This function is not required to be OPERABLE in MODE 3 below the P-11 setpoint. Other ESF functions are used to detect accident conditions and actuate the ESF systems in this MODE. In MODES 4, 5 and 6, this function is not needed for accident detection and mitigation.

1.e. Steam Line Pressure

1.e.(1). Steam Line Pressure Low Setpoint

Steam Line Pressure--Low must be OPERABLE in MODES 1, 2, and 3 (above P-11) when a secondary-side break or stuck open valve could result in the rapid depressurization of the steam lines. This signal may be manually blocked by the operator below the P-11 setpoint. Below P-11, feed line break is not a concern, inside containment SLB will be terminated by automatic SI actuation via Containment Pressure--High 1, and outside containment SLB will be terminated by the Steam Line Pressure-Negative Rate--High signal for steam line isolation. This function is not required to be OPERABLE in MODES 4, 5, or 6, because there is insufficient energy in the secondary side of the plant to cause an accident.

1.e.(2). High Differential Pressure between Steam Lines

Steam line high differential pressure must be OPERABLE in MODES 1, 2, and 3 when a

(continued)

(continued)

BASES (continued)

APPLICABILITY (continued)

secondary-side break or stuck open valve could result in the rapid depressurization of the steam line(s). This function is not required to be OPERABLE in MODES 4, 5, or 6, because there is not sufficient energy in the secondary side of the plant to cause an accident.

1.f. High Steam Flow in Two Steam Lines Coincident With T_{avg}--Low Low or Coincident With Steam Line Pressure--Low

This function must be OPERABLE in MODES 1, 2, and 3 (above P-12) when a secondary-side break or stuck open valve could result in the rapid depressurization of the steam line(s). This signal may be manually blocked by the operator when below the P-12 setpoint. Above P-12, this function is automatically unblocked. [At this facility, the function is not required to be OPERABLE below P-12 for the following reasons:] This function is not required to be OPERABLE in MODES 4, 5, or 6, because there is insufficient energy in the secondary side of the plant to cause an accident.

2. Containment Spray

2.a. Manual Initiation

2.b. Automatic Actuation Logic and Actuation Relays

Manual and automatic initiation of containment spray must be OPERABLE in MODES 1, 2, and 3. When there is a potential for an accident to occur and sufficient energy in the primary or secondary systems to pose a threat to containment integrity due to overpressure conditions, manual initiation is also required in MODE 4 even though automatic actuation is not required. In this MODE adequate time is available to manually actuate required components in the event of a DBA, but because of the large number of components actuated on a containment spray, actuation is simplified by the use of the manual

(continued)

(continued)

BASES (continued)

APPLICABILITY (continued)

actuation push buttons. Automatic actuation logic and actuation relays must be OPERABLE in MODE 4 to support system level manual initiation. In MODES 5 and 6, there is insufficient energy in the primary and secondary systems to result in containment overpressure. In MODES 5 and 6, there is also adequate time for the operators to evaluate unit conditions and respond to mitigate the consequences of abnormal conditions by manually starting individual components.

2.c. <u>Containment Pressure</u>

Containment Pressure--[High 3] [High High] must be OPERABLE in MODES 1, 2, and 3 when there is sufficient energy in the primary and secondary sides to pressurize the containment following a pipe break. In MODES 4, 5, and 6, there is insufficient energy in the primary and secondary sides to pressurize the containment and reach the Containment Pressure--[High 3] and [High High] setpoints.

3. <u>Containment Isolation</u>

3.a. <u>Phase A Isolation</u>

3.a.(1). <u>Manual initiation</u>

3.a.(2). <u>Automatic Actuation Logic and Actuation Relays</u>

Manual and automatic initiation of Phase A Containment Isolation must be OPERABLE in MODES 1, 2, and 3, when there is a potential for an accident to occur. Manual initiation is also required in MODE 4 even though automatic actuation is not required. In this MODE, adequate time is available to manually actuate required components in the event of a DBA, but because of the large number of components actuated on a phase A containment isolation, actuation is simplified by the use of the manual actuation push buttons. Automatic actuation logic and actuation relays

(continued)

(continued)

BASES (continued)

APPLICABILITY (continued)

must be OPERABLE in MODE 4 to support system level manual initiation. In MODES 5 and 6, there is insufficient energy in the primary or secondary systems to pressurize the containment to require phase A containment isolation. There will also be adequate time for the operator to evaluate unit conditions and manually actuate individual isolation valves in response to abnormal or accident conditions.

3.b. Phase B Isolation

3.b.(1). Manual initiation

3.b.(2). Automatic Actuation Logic and Actuation Relays

Manual and automatic initiation of Phase B containment isolation must be OPERABLE in MODES 1, 2, and 3 when there is a potential for an accident to occur. Manual initiation is also required in MODE 4 even though automatic actuation is not required. In the MODE adequate time is available to manually actuate required components in the event of a DBA, but because of the large number of components actuated on a phase B containment isolation, actuation is simplified by the use of the manual actuation push buttons. Automatic actuation logic and actuation relays must be OPERABLE in MODE 4 to support system level manual initiation. In MODES 5 and 6, there is insufficient energy in the primary or secondary systems to pressurize the containment to require phase B containment isolation. There will also be adequate time for the operator to evaluate unit conditions and manually actuate individual isolation valves in response to abnormal or accident conditions.

(continued)

(continued)

BASES (continued)

APPLICABILITY (continued)

3.b.(3). Containment Pressure

The bases for containment pressure MODE applicability is as discussed for Function 2.c above.

4. Containment Purge Isolation

Containment purge manual isolation, automatic logic, and input functions are required OPERABLE in MODES 1, 2, 3, 4 and during CORE ALTERATIONS or movement of fuel assemblies within containment with irradiated fuel in containment. Under these conditions the potential exists for an accident which could release fission-product radioactivity into containment, therefore, the containment purge and exhaust isolation must be OPERABLE in these MODES. For Function 4.c, SI, OPERABILITY in MODE 4 is required only to ensure system level manual SI also isolates the containment purge valves.

5. Steam Line Isolation

5.a. Manual Initiation

5.b. Automatic Actuation Logic and Actuation Relays

Manual and automatic initiation of steam line isolation must be OPERABLE in MODES 1, 2, and 3 when there is sufficient energy in the RCS and SGs to have an SLB or other accident resulting in the release of significant quantities of energy to cause a cooldown of the primary system. Although it is possible to close all MSIVs in MODES 2 and 3, the Steam Line Isolation function remains OPERABLE in these MODES to ensure that the MSIVs can be immediately closed if inadvertently open. In MODES 4, 5, and 6, there is insufficient energy in the RCS and SGs to experience an SLB or other accident releasing significant quantities of energy.

(continued)

(continued)

BASES (continued)

APPLICABILITY (continued)

5.c. Containment Pressure--High 2

Containment Pressure--High 2 must be OPERABLE in MODES 1, 2, and 3 when there is sufficient energy in the primary and secondary side to pressurize the containment following a pipe break. There would be a significant increase in the containment pressure, thus allowing detection and closure of the MSIVs. Although it is possible to close all MSIVs in MODES 2 and 3, the Steam Line Isolation function remains OPERABLE in these MODES to ensure that the MSIVs can be immediately closed if inadvertently open. In MODES 4, 5, and 6, there is not enough energy in the primary and secondary sides to pressurize the containment to the Containment Pressure--High 2 setpoint.

5.d. Steam Line Pressure

5.d.(1). Steam Line Pressure--Low Setpoint

Steam Line Pressure--Low function must be OPERABLE in MODES 1, 2, and 3 (above P-11) when a secondary-side break or stuck open valve could result in the rapid depressurization of the steam lines. This signal may be manually blocked by the operator below the P-11 setpoint. Below P-11, an inside containment SLB will be terminated by automatic actuation via Containment Pressure--High 2, and stuck valve transients and outside containment steam line breaks will be terminated by the Steam Line Pressure-Negative Rate--High signal for Steam Line Isolation. Although it is possible to close all MSIVs in MODES 2 and 3, the Steam Line Isolation function remains OPERABLE in these MODES to ensure that the MSIVs can be immediately closed if inadvertently open. This function is not required to be OPERABLE in MODES 4, 5, and 6, because there is insufficient energy in the secondary side of the unit to have an accident.

(continued)

(continued)

BASES (continued)

APPLICABILITY (continued)

5.d.(2). Steam Line Pressure-Negative Rate--High Setpoint

Steam Line Pressure-Negative Rate--High must be OPERABLE in MODE 3, when less than the P-11 setpoint, when a secondary-side break or stuck open valve could result in the rapid depressurization of the steam line(s). In MODES 1 and 2, and in MODE 3 when above the P-11 setpoint, this signal is automatically disabled and the Steam Line Pressure--Low signal is automatically enabled. Although it is possible to close all MSIVs in MODES 2 and 3, the Steam Line Isolation function remains OPERABLE in these MODES to ensure that the MSIVs can be immediately closed if inadvertently open. In MODES 4, 5, and 6, there is insufficient energy in the primary and secondary sides to have an SLB or other accident resulting in a release of significant quantities of energy causing a cooldown of the RCS.

5.e. High Steam Flow in Two Steam Lines Coincident With T_{avg}--Low Low or Coincident With Steam Line Pressure Low

This function must be OPERABLE in MODES 1 and 2, and in MODE 3, when above P-12 setpoints, when a secondary-side break or stuck open valve could result in the rapid depressurization of the steam lines. [At this facility, OPERABILITY below P-12 is not required for the following reasons:]. Although it is possible to close all MSIVs in MODES 2 and 3, the Steam Line Isolation function remains OPERABLE in these MODES to ensure that the MSIVs can be immediately closed if inadvertently open. This function is not required to be OPERABLE in MODES 4, 5, and 6, because there is insufficient energy in the secondary side of the unit to have an accident.

(continued)

(continued)

BASES (continued)

APPLICABILITY (continued)

5.f. High Steam Flow Coincident With Safety Injection Coincident With T_{avg}--Low Low

This function must be OPERABLE in MODES 1 and 2, and in MODE 3 when above P-12 setpoints, when a secondary-side break or stuck open valve could result in the rapid depressurization of the steam lines. [At this facility, OPERABILITY below P-12 is not required for the following reasons]. Although it is possible to close all MSIVs in MODES 2 and 3, the Steam Line Isolation function remains OPERABLE in these MODES to ensure that the MSIVs can be immediately closed if inadvertently open. This function is not required to be OPERABLE in MODES 4, 5, and 6, because there is insufficient energy in the secondary side of the unit to have an accident.

5.g. High High Steam Flow in Two Steam Lines Coincident With Safety Injection

This function must be OPERABLE in MODES 1, 2, and 3 when a secondary-side break or stuck open valve could result in the rapid depressurization of the steam lines. Although it is possible to close all MSIVs in MODES 2 and 3, the Steam Line Isolation function remains OPERABLE in these MODES to ensure that the MSIVs can be immediately closed if inadvertently open. This function is not required to be OPERABLE in MODES 4, 5, and 6, because there is insufficient energy in the secondary side of the unit to have an accident.

5.h. High Steam Flow in Two Steam Lines Coincident With T_{avg}--Low Low

This function must be OPERABLE in MODES 1, 2, and in MODE 3, when above P-12 setpoints, when a secondary-side break or stuck open valve could result in the rapid depressurization of the steam lines. [At this facility, OPERABILITY below P-12 is not required for the following reasons:]. Although it is possible to close all MSIVs in MODES 2 and 3, the Steam Line Isolation function remains OPERABLE in these MODES to ensure that

(continued)

(continued)

BASES (continued)

APPLICABILITY (continued)

the MSIVs can be immediately closed if inadvertently open. This function is not required to be OPERABLE in MODES 4, 5, and 6, because there is insufficient energy in the secondary side of the unit to have an accident.

6. Turbine Trip and Feedwater Isolation

6.a. Automatic Actuation Logic and Actuation Relays

6.b. Steam Generator Water Level--High High (P-14)

6.c. Safety Injection

These functions must be OPERABLE in MODES 1 and 2 when the Main Feedwater System is in operation and the turbine generator may be in operation. In MODES 3, 4, 5, and 6, the Main Feedwater System and the turbine generator are not in service and this function is not required to be OPERABLE.

7. Auxiliary Feedwater

7.a. Automatic Actuation Logic and Actuation Relays (Solid State Protection System)

7.b. Automatic Actuation Logic and Actuation Relays (Balance of Plant ESFAS)

7.c. Steam Generator Water Level--Low Low

7.d. Safety Injection

7.e. Loss of Offsite Power

These functions must be OPERABLE in MODES 1, 2, and 3 to ensure that the SGs remain the heat sink for the reactor. SG Water Level--Low Low in any operating SG will cause the motor-driven AFW pumps to start. The system is aligned so that upon a start of the pump, water immediately begins to flow to the SGs. SG Water Level--Low Low in any two operating SGs will cause the turbine-driven pump to start. This function does

(continued)

(continued)

BASES (continued)

APPLICABILITY (continued)

not have to be OPERABLE in MODES 5 and 6 because there is not enough heat being generated in the reactor to require the SGs as a heat sink. In MODE 4, AFW actuation does not need to be OPERABLE because either AFW or RHR will already be in operation to remove decay heat.

7.f. Undervoltage Reactor Coolant Pump

7.g. Trip of All Main Feedwater Pumps

This function must be OPERABLE in MODES 1 and 2. This ensures that at least one SG is provided with water to serve as the heat sink to remove reactor decay heat and sensible heat in the event of an accident. In MODES 3, 4, 5, the RCPs and main feedwater pumps may be normally shutdown and thus neither pump trip is indicative of a condition requiring automatic AFS initiation.

7.h. Auxiliary Feedwater Pump Suction Transfer on Suction Pressure--Low

This function must be OPERABLE in MODES 1, 2, and 3 to ensure a safety-grade supply of water for the AFS to maintain the SGs as the heat sink for the reactor. This function does not have to be OPERABLE in MODES 5 and 6, because there is not enough heat being generated in the reactor to require the SGs as a heat sink. In MODE 4, AFW actuation does not need to be OPERABLE because either AFW or RHR will already be in operation to remove decay heat.

8. Automatic Switchover to Containment Sump

8.a. Automatic Actuation Logic and Actuation Relays

8.b. RWST Level--Low Low Coincident With Safety Injection

(continued)

(continued)

BASES (continued)

APPLICABILITY (continued)

8.c. RWST Level--Low Low Coincident With Safety Injection and Coincident With Containment Sump Level--High

These functions must be OPERABLE in MODES 1, 2, 3, and 4 when there is a potential for a LOCA to occur, to ensure a continued supply of water for the ECCS pumps. This function is not required to be OPERABLE in MODES 5 and 6, because there is adequate time for the operator to evaluate unit conditions and respond by manually starting systems, pumps, and other equipment to mitigate the consequences of an abnormal condition or accident. Plant pressure and temperature are very low and many ESF components are administratively locked out or otherwise prevented from actuating to prevent inadvertent overpressurization of unit systems.

9. Control Room Emergency Ventilation

9.a. Manual Initiation

9.b. Automatic Actuation Logic and Actuation Relays

These Control Room Emergency Ventilation functions must be OPERABLE in MODES 1, 2, 3, 4, and in all MODES during CORE ALTERATIONS and movement of irradiated fuel or loads over irradiated fuel to ensure a habitable environment for the control room operators.

9.c. Phase A Isolation

Phase A isolation actuates control room emergency ventilation. Phase A isolation was discussed previously.

9.d. Control Room Atmosphere and Air Intake Radiation

The requirements for radiation monitors are located in LCO 3.3.7, "Radiation Monitors."

(continued)

(continued)

BASES (continued)

APPLICABILITY (continued)

10. Engineered Safety Feature Actuation System Interlocks

10.a. Reactor Trip, P-4

This function must be OPERABLE in MODES 1, 2, and 3 when the reactor may be critical or approaching criticality. This function does not have to be OPERABLE in MODES 4, 5, or 6, because the main turbine, the Main Feedwater System, and the Steam Dump System are not in operation.

10.b. Pressurizer Pressure, P-11

This function must be OPERABLE in MODES 1, 2, and 3 to allow an orderly cooldown and depressurization of the unit without the actuation of SI or main steam isolation. This function does not have to be OPERABLE in MODE 4, 5, or 6, because plant pressure must already be below the P-11 setpoint for the requirements of the heatup and cooldown curves to be met.

10.c. T_{avg}--Low Low, P-12

This function must be OPERABLE in MODES 1, 2, and 3 when a secondary-side break or stuck open valve could result in the rapid depressurization of the steam lines. This function does not have to be OPERABLE in MODES 4, 5, or 6, because there is insufficient energy in the secondary side of the unit to have an accident.

10.d. Steam Generator Water Level--High High, P-14

This function must be OPERABLE in MODES 1 and 2 when the turbine generator and the Main Feedwater System may be in operation. This function does not have to be OPERABLE in MODE 3, 4, 5, or 6, because the turbine generator and the Main Feedwater System are not in service.

A Note has been added in the Applicability to provide clarification that for this LCO, each function specified in Table 3.3.2-1 shall be treated as an

(continued)

(continued)

BASES (continued)

APPLICABILITY (continued)	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 channel is inoperable when it does not satisfy the OPERABILITY criteria for the function's channels. These criteria are outlined, for each function, in the LCO section of the Bases. The most frequent occurrence to render a protection function inoperable is the determination that a bistable or process module has drifted sufficiently to exceed the ALLOWABLE VALUE. Typically, the drift is found to be small and results in a delay of actuation rather than a total loss of function. This determination is generally made during the performance of an ANALOG CHANNEL OPERATIONAL TEST, when the process instrumentation is set up for adjustment to bring it within specification. If the trip setpoint is less conservative than the ALLOWABLE VALUE, the channel must be declared inoperable immediately and the appropriate Condition from Table 3.3.2-1 should be entered. In the event a channel's trip setpoint is found nonconservative with respect to the ALLOWABLE VALUE, or the transmitter, a rack module, or an [SSPS] module is found inoperable, then the function which that channel provides must be declared inoperable and the LCO Condition entered for the particular protection function affected. When the Required Channels are specified only on a per steam line, per loop, per SG, etc., basis, then the Condition may be entered separately for each steam line, loop, SG, etc., as appropriate. When the number of inoperable channels in a trip function exceed those specified in one or other related Conditions associated with a trip function, then the plant is outside the safety analysis. Therefore, LCO 3.0.3 should be immediately entered if applicable in the current MODE of operation.

(continued)

(continued)

BASES (continued)

ACTIONS (continued)

Condition A

Condition A is applicable to all ESFAS protection functions. Condition A 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 3.3.2-1 and to take the Required Actions for the protection functions affected. The Completion Times are those from the referenced Conditions and Required Actions.

Condition B

Condition B is applicable to manual initiation of:

- Safety Injection;
- Containment Spray;
- Phase A Isolation; and
- Phase B Isolation.

This action addresses the train orientation of the [SSPS] for the functions listed above. If a train is inoperable, 48 hours is allowed to return it to an OPERABLE status. Note that for containment spray and phase B isolation, failure of one or both channels in one train render the train inoperable. Condition B, therefore, encompasses both situations. If the train cannot be restored to OPERABLE status, the plant must be placed in a MODE in which the LCO does not apply. This is done by placing the plant in at least MODE 3 within an additional 6 hours (54 hours total time) and in MODE 5 within an additional 36 hours (84 hours total time). The Completion Times are reasonable, based on operating experience, to reach the required MODES from full power in an orderly manner and without challenging plant systems.

[For this facility, the Completion Time of 48 hours is justified as follows: Note: (WCAP-10271-P-A is not applicable for this function-Ref. 7)].

Condition C

Condition C is applicable to the automatic actuation logic and actuation relays for the following functions:

- Safety Injection;

(continued)

(continued)

BASES (continued)

ACTIONS (continued)

- Containment Spray;
- Phase A Isolation;
- Phase B Isolation; and
- Automatic Switchover to Containment Sump.

This action addresses the train orientation of the [SSPS] and the master and slave relays. If one train is inoperable, 6 hours are allowed to restore the train to OPERABLE status. The specified Completion Time is reasonable considering that there is another train OPERABLE and the low probability of an event occurring during this interval. If the train cannot be restored to OPERABLE status, the plant must be placed in a MODE in which the LCO does not apply. This is done by placing the plant in at least MODE 3 within an additional 6 hours (12 hours total time) and in MODE 5 within an additional 36 hours (42 hours total time). The Completion Times are reasonable, based on operating experience, to reach the required MODE from full power in an orderly manner and without challenging plant systems.

The Required Actions are modified by a Note which allows one train to be bypassed for up to 4 hours for surveillance testing, provided the other train is OPERABLE. This allowance is based upon the reliability analysis assumption of Reference 7 that 4 hours is the average time required to perform channel surveillance.

Condition D

Condition D is applicable to:

- Containment Pressure--High 1;
- Pressurizer Pressure--Low (four loop plants);
- Pressurizer Pressure--Low (two, three, and four loop plants);
- Steam Line Pressure--Low Setpoints;
- Steam Line Differential Pressure--High;
- High Steam Flow in Two Steam Lines Coincident With T_{avg}--Low Low or Coincident With Steam Line Pressure--Low;
- Containment Pressure--High 2;
- Steam Line Pressure-Negative Rate--High Setpoints;
- High Steam Flow Coincident With Safety Injection Coincident With T_{avg}--Low Low;

(continued)

(continued)

BASES (continued)

ACTIONS (continued)

- High High Steam Flow Coincident With Safety Injection;
- High Steam Flow in Two Steam Lines Coincident With T_{avg}--Low Low, OR Coincident With Steam Line Pressure Low;
- SG Water Level--Low Low (four loop plants); and
- SG Water level--Low Low (two, three and four loop plants).

If one channel is inoperable, 6 hours are allowed to restore channel to OPERABLE status or to place it in the tripped position. Generally this Condition applies to functions that operate on two-out-of-three logic. Therefore, failure of one channel places the function in a two-out-of-two configuration. One channel must be tripped to place the function in a one-out-of-three configuration that satisfies redundancy requirements.

At some plants, both pressurizer pressure and SG water level provide inputs to control and protection functions. Pressurizer pressure inputs to pressurizer pressure control. SG water level inputs to SG level control. It is therefore necessary to be able to sustain two simultaneous channel failures, one for the initiating failure, which necessitates protection system actuation, and one for the protection system in order to satisfy the redundancy and control-protection independence requirements. When one channel fails it must be placed in trip in order to create an effective one-out-of-three logic necessary to satisfy this requirement.

The Containment Pressure--High 2 signals provide no input to control systems and therefore, two-out-of-three logic would normally be acceptable. However, these channels feed energize-to-trip bistables that actuate containment spray. Therefore, to provide enhanced reliability and avoid the need to trip a containment spray channel in the event of a single failure, the Containment Pressure--High 2 function has been designed with two-out-of-four logic. If placed in the tripped condition where one-out-of-two or one-out-of-three logic will result in actuation.

Failure to restore the inoperable channel to OPERABLE status or place it in the tripped condition within 6 hours requires the unit be placed in MODE 3 within the following 6 hours and MODE 4 within the next 6 hours.

(continued)

(continued)

BASES (continued)

ACTIONS (continued)

The time allowed to reach MODES 3 and 4 from MODE 1 is reasonable, based on operating experience, to reach the required MODE from full power in an orderly manner and without challenging plant systems. In MODE 4, these functions are no longer required OPERABLE. The Required Actions are modified by a Note which allows the inoperable channel to be bypassed for up to 4 hours for surveillance testing of other channels. The 6 hours allowed to restore channel to OPERABLE status or to place the inoperable channel in the tripped condition and the 4 hours allowed for testing is justified in Reference 7.

Condition E

Condition E is applicable to:

- Containment Spray Containment Pressure--High 3 (two, three, and four loop plants);
- Containment Pressure Containment Pressure--High High setpoint;
- Containment Phase B Isolation Containment Pressure--High 3; and
- Containment Phase B Isolation Containment Pressure--High High.

None of these signals has input to a control function, thus two-out-of-three logic is acceptablc protective requirements. However, a two-out-of-three design would required tripping a failed channel. This is undesirable, because a single failure would then cause spurious containment spray initiation. Spurious spray actuation is undesirable because of the cleanup problems presented. Therefore, these channels are designed with two-out-of-four logic so that a failed channel may be bypassed rather than tripped. If one channel is inoperable, the Required Action allows 6 hours to restore the channel to OPERABLE condition. Note that one channel may be bypassed and still satisfy single failure criterion. Furthermore, with one channel bypassed, a single instrumentation channel failure will not spuriously initiate containment spray.

In order to avoid the inadvertent actuation of containment spray and phase B containment isolation, the inoperable channel should not be placed in the tripped position. Instead it is bypassed. Restoring the channel to OPERABLE

(continued)

(continued)

BASES (continued)

ACTIONS (continued)

status or placing the inoperable channel in the bypass condition within 6 hours is sufficient to assure that the function remains OPERABLE and minimizes the time that the function may be in a partial trip condition (assuming the inoperable channel has failed high), and is further justified based on the low probability of an event occurring during this interval. Failure to restore the inoperable channel to OPERABLE status or place it in the bypassed condition within 6 hours requires the unit be placed in MODE 3 within the following 6 hours and MODE 4 within the next 6 hours. The time allowed is reasonable, based on operating experience, to reach the required MODE from full power in an orderly manner and without challenging plant systems. In MODE 4, these functions are no longer required OPERABLE.

The Required Actions are modified by a Note which allows one additional channel to be bypassed for up to 4 hours for surveillance testing. Placing a second channel in the bypass condition for up to 4 hours for testing purposes is acceptable based on the results of Reference 7.

<u>Conditions F, G, H, and I</u>

Conditions F, G, H, and I are applicable to the containment purge isolation function and address the train orientation of the [SSPS] and the master and slave relays for this function. Conditions F and G are applicable in MODES 1, 2, 3, and 4. Condition H and I are applicable in any MODE when any containment purge or exhaust penetration is open during movement of fuel assemblies in containment, with irradiated fuel in containment, or during CORE ALTERATIONS.

If a train is inoperable, operation may continue as long as the Containment Purge System supply and exhaust valves are placed and maintained in the closed position within 4 hours. The specified Completion Time is reasonable considering that there is another train OPERABLE and the low probability of an event occurring during this interval. If two trains are inoperable, the valves must be placed and maintained in the closed position within 1 hour. This action accomplishes the purpose of the actuation function. Once the valves are closed, this condition may continue for an indefinite period of time. If for some reason the valves cannot be closed within the allowed time, the unit must be placed in MODE 3

(continued)

(continued)

BASES (continued)

ACTIONS (continued)

within the next 6 hours and MODE 5 within the following 36 hours. The time allowed is reasonable, based on operating experience, to reach the required MODE from full power in an orderly manner and without challenging plant systems. During movement of fuel assemblies within containment, with irradiated fuel in containment and any containment purge or exhaust penetration open, or during CORE ALTERATIONS, with any purge or exhaust penetration open, these activities must be suspended if the purge and exhaust penetrations cannot be isolated. Suspending these activities minimizes the risk of an accident that could require mitigation by the Purge Isolation System.

Condition J

Condition J is applicable to:

- Manual Initiation of Steam Line Isolation;
- Loss of Offsite Power;
- Auxiliary Feedwater Pump Suction Transfer on Suction Pressure--Low; and
- P-4 Interlock.

For the Manual Initiation and the P-4 Interlock functions this action addresses the train orientation of the [SSPS] for these functions. For the Loss of Offsite Power function this action recognizes the lack of manual trip provision for a failed channel. For the AFW System pump suction transfer channels, this action recognizes that placing a failed channel in trip during operation is not necessarily a conservative action. Spurious trip of this function could align the AFW System to a source that is not immediately capable of supporting pump suction. If a train or channel is inoperable, 48 hours is allowed to return it to an OPERABLE status. If the function cannot be returned to an OPERABLE status, the unit must be placed in MODE 3 within the next 6 hours and MODE 4 within the following 6 hours. The time allowed is reasonable, based on operating experience, to reach the required MODE from full power in an orderly manner and without challenging plant systems. In MODE 4, the plant does not have any analyzed transients or conditions which require the explicit use of the protection functions notes above.

(continued)

(continued)

BASES (continued)

ACTIONS (continued)

[For this facility, the Completion Time of 48 hours is justified as follows: (WCAP-10271-P-A is not applicable for this function-Ref. 7)].

Condition K

Condition K is applicable to the automatic actuation logic and actuation relays for the Steam Line Isolation and AFW actuation functions.

This action addresses the train orientation of the [SSPS] and the master and slave relays for these functions. If one train is inoperable, 6 hours are allowed to restore the train to OPERABLE status. The Completion Time for restoring a train to OPERABLE status is reasonable considering that there is another train OPERABLE and the low probability of an event occurring during this interval. If the train cannot be returned to an OPERABLE status, the unit must be placed in MODE 3 within the next 6 hours and MODE 4 within the following 6 hours. The time allowed is reasonable, based on operating experience, to reach the required MODE from full power in an orderly manner and without challenging plant systems. Placing the unit in MODE 4 removes all requirements for OPERABILITY of the protection channels and actuation functions. In this MODE, the unit does not have analyzed transients or conditions which require the explicit use of the protection functions noted above. The Required Actions are modified by a Note which allows one train to be bypassed for up to 4 hours for surveillance testing provided the other train is OPERABLE. This allowance is based upon the reliability analysis (Ref. 7) assumption that 4 hours is the average time required to perform channel surveillance.

Condition L

Condition L is applicable to the automatic actuation logic and actuation relays for the Turbine Trip and Feedwater Isolation function.

This action addresses the train orientation of the [SSPS] and the master and slave relays for this function. If one train is inoperable, 6 hours are allowed to restore the train to OPERABLE status or the unit must be placed in MODE 3 within the next 6 hours. The Completion Time for restoring a train to OPERABLE status is reasonable

(continued)

(continued)

ACTIONS (continued)

considering that there is another train OPERABLE and the low probability of an event occurring during this interval. The Completion Time of 6 hours is reasonable, based on operating experience and normal cooldown rates, to reach MODE 3 from MODE 1 in an orderly manner and without challenging plant systems. These functions are no longer required in MODE 3. Placing the unit in MODE 3 removes all requirements for OPERABILITY of the protection channels and actuation functions. In this MODE, the unit does not have analyzed transients or conditions which require the explicit use of the protection functions noted above. The Required Actions are modified by a Note which allows one train to be bypassed for up to 4 hours for surveillance testing provided the other train is OPERABLE. This allowance is based upon the reliability analysis (Ref. 7) assumption that 4 hours is the average time required to perform channel surveillance.

Condition M

Condition M is applicable to:

- SG Water Level--High High (P-14) (two, three, and four loop plants), and
- Undervoltage Reactor Coolant Pump.

If one channel is inoperable, 6 hours are allowed to restore one channel to OPERABLE status or to place it in the tripped condition. If placed in the tripped condition, the function is then in a partial trip condition where one-out-of-two or one-out-of-three logic will result in actuation. Failure to restore the inoperable channel to OPERABLE status or place it in the tripped condition within 6 hours requires the unit be placed in MODE 3 within the following 6 hours. The time allowed is reasonable, based on operating experience and normal cooldown rates, to reach MODE 3 from MODE 1 in an orderly manner and without challenging plant systems. In MODE 3, these functions are no longer required OPERABLE.

The Required Actions are modified by a Note that allows the inoperable channel to be bypassed for up to 4 hours for surveillance testing of other channels. The 6 hours allowed to place the inoperable channel in the tripped condition and the 4 hours allowed for a second channel to be in the bypassed condition for testing are justified in Reference 7.

(continued)

(continued)

BASES (continued)

ACTIONS (continued)

Condition N

Condition N is applicable to the AFW pump start on trip of all Main Feedwater pumps. This action addresses the train orientation of the [SSPS] for the auto-start function of the AFW System on loss of all main feedwater pumps. The OPERABILITY of the AFW System must be assured by allowing automatic start of the AFW System pumps. If a train or channel is inoperable, 48 hours are allowed to return it to an OPERABLE status. If the function cannot be returned to an OPERABLE status, 6 hours are allowed to place the unit in MODE 3. The Completion Time of 6 hours is reasonable, based on operating experience and normal cooldown rates, to reach MODE 3 from MODE 1 in an orderly manner and without challenging plant systems. In MODE 3, the unit does not have any analyzed transients or conditions which require the explicit use of the protection function noted above. The allowance of 48 hours to return the train to an OPERABLE status and 6 hours to reach MODE 3 is justified in Reference 7.

Condition O

Condition O is applicable to:

- RWST Level--Low Low Coincident with Safety Injection; and
- RWST Level--Low Low Coincident with Safety Injection and Coincident with Containment Sump Level--High.

RWST Level--Low Low Coincident With Safety Injection and Coincident With Containment Sump Level--High provides actuation of switchover to the containment sump. Note that this function requires the bistables to energize to perform their required action. The failure of up to two channels will not prevent the operation of this function. However, placing a failed channel in the tripped condition could result in the premature switchover to the sump, prior to the injection of the minimum volume from the RWST. Placing the inoperable channel in bypass results in a two-out-of-three logic configuration, which satisfies the requirement to allow another failure without disabling actuation of the switchover when required. Restoring the channel to OPERABLE status or placing the inoperable channel in the bypass condition within 6 hours is sufficient to assure that the

(continued)

(continued)

BASES (continued)

ACTIONS (continued)

function remains OPERABLE and minimizes the time that the function may be in a partial trip condition (assuming the inoperable channel has failed high). If the channel cannot be returned to OPERABLE status or placed in the bypass condition within 6 hours, the plant must be placed in MODE 3 within the following 6 hours and MODE 5 within the next 30 hours. The Completion Times of 6 and 30 hours are reasonable, based on operating experience, to reach the required MODE from full power in an orderly manner and without challenging plant systems. In MODE 5, the unit does not have any analyzed transients or conditions which require the explicit use of the protection functions noted above.

The Required Actions are modified by a Note that allows placing a second channel in the bypass condition for up to 4 hours for surveillance testing. The total of 12 hours to reach MODE 3 and 4 hours for a second channel to by bypassed is acceptable based on the results of Reference 7.

Conditions P, Q, and R

Conditions P, Q, and R are applicable to manual and automatic actuation of control room emergency ventilation.

This action addresses the train orientation of the [SSPS] for this function. Condition P applies to the failure of a single train in MODE 1, 2, 3, or 4. If one train is inoperable, 7 days are permitted to restore it to OPERABLE status. The 7-day Completion Time is the same as is allowed if one train of the mechanical portion of the system is inoperable. The basis for this time is the same as provided in the Control Room Emergency Ventilation System LCO. If the train cannot be restored to OPERABLE status, the plant must be placed in MODE 3 within the following 6 hours and MODE 5 within the next 30 hours. The Completion Times are reasonable, based on operating experience, to reach the required MODE from full power in an orderly manner and without challenging plant systems.

Condition Q applies to the failure of one train during CORE ALTERATIONS and when moving irradiated fuel. If one train is inoperable, 7 days are permitted to restore it to OPERABLE status or place one OPERABLE Control Room Emergency Filtration System (CREFS) train in emergency filtration mode. If neither the ESFAS instrumentation train nor CREFS

(continued)

(continued)

BASES (continued)

ACTIONS (continued)

train can be restored to OPERABLE status or placed in operation respectively, CORE ALTERATIONS, positive reactivity additions, and movement of irradiated fuel must be suspended immediately following the 7 days allowed Completion Time. The justification of the 7-day Completion Time is the same as discussed above for Condition P.

Condition R applies to the failure of two trains during CORE ALTERATIONS and when moving irradiated fuel. If two trains are inoperable, Required Actions are immediately taken to restore one ESFAS Instrumentation train to OPERABLE status, or place one OPERABLE CREFS train in emergency filtration mode, or CORE ALTERATIONS, positive reactivity additions, and movement of irradiated fuel are immediately suspended. The Required Actions for Conditions Q and R have been modified by a Note specifying that CREFS be placed manually in the emergency filtration mode, if the auto-swapover emergency filtration is inoperable.

Condition S

Condition S is applicable to the P-11 and P-12 interlocks.

With pressurizer pressure above the P-11 setpoint, the Pressurizer Pressure--Low and Steam Line Pressure--Low (only in MODE 3) SI and Steam Line Pressure--Low steam line isolation signals are automatically enabled with pressurizer pressure below P-11 setpoint, and the operator can manually block the Pressurizer Pressure--Low and Steam Line Pressure--Low SISs and the Steam Line Pressure Low steam line isolation signal. Additionally, below P-11 setpoint, the Steam Line Pressure-Negative Rate--High provides closure of the MSIVs for an SLB, to maintain at least one unfaulted SG as a heat sink for the reactor and to limit the mass and energy release to containment. With T_{avg} above the P-12 setpoint, SI on High Steam Flow Coincident With Either Steam Line Pressure--Low or T_{avg}--Low Low is also reinstated. With one channel inoperable the operator must verify that the interlock is in the required state for the existing plant conditions. This action manually accomplishes the function of the interlock. Determination must be made within 1 hour. [For this facility, the repetition of action is accomplished after MODE changes as follows:] The 1-hour Completion Time is equal to the time allowed by LCO 3.0.3 to initiate shutdown actions in the event of a complete loss of

(continued)

(continued)

BASES (continued)

ACTIONS (continued)

ESFAS function. If the determination that the interlock is not in the required state for the existing conditions, the unit must be placed in MODE 3 within the next 6 hours and MODE 4 within the following 6 hours. The time allowed is reasonable, based on operating experience, to reach the required MODE from full power in an orderly manner and without challenging plant systems. Placing the plant in MODE 4 removes all requirements for OPERABILITY of these interlocks.

Condition T

Condition T is applicable to the P-14 interlock.

The actions for Condition T are identical to those for Condition S except that the P-14 interlock is not required to be OPERABLE in MODE. Therefore, shutdown to MODE 3 within 7 hours is required if interlock status cannot be verified within 1 hour. The Completion Times are reasonable, based on operating experience, to reach the required MODE from full power in an orderly manner and without challenging plant systems.

Condition U

Condition U is applicable to each one of the ESFAS functions presented in Table 3.3.2-1.

Required Action U.1 verifies that the Required Actions have been initiated for those supported systems declared inoperable because of the inoperability of the support channel(s) or train(s) within a Completion Time of 1 hour. The specified Completion Time is sufficient for plant operations personnel to make this determination.

Required Action U.1 ensures that those identified Required Actions associated with supported systems impacted by the inoperability of channel(s) or train(s) associated with each ESFAS function have been initiated. This can be accomplished by entering the supported systems' LCOs independently or as a group of Required Actions that need to be initiated every time Condition U is entered. [For this facility, the identified supported systems' Required Actions associated with each ESFAS function are as follows:]

(continued)

(continued)

BASES (continued)

ACTIONS (continued)

Required Action U.2 verifies that all required support or supported features associated with the other redundant train(s) or 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 should 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 ESFAS function are found in the SRs column of Table 3.3.2-1.

Note that each channel of process protection supplies both trains of the ESFAS. When testing channel I, train A and train B must be examined. Similarly, train A and train B must be examined when testing channel II, channel III, and channel IV (if applicable). The CHANNEL CALIBRATION and ANALOG CHANNEL OPERATIONAL TESTS are performed in a manner that is consistent with the assumptions used in analytically calculating the required channel accuracies. For channels that include dynamic transfer functions (e.g., lag, lead/lag, rate/lag, etc.), the response time test may be performed with the transfer functions set to one with the resulting measured response time compared to the appropriate FSAR response time. Alternately, the response time test can be performed with the time constants set to their nominal value provided the required response time is analytically calculated assuming the time constants are set at their nominal values. The response time may be measured by a series of overlapping tests such that the entire response time is measured.

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 SER that establishes the acceptability of each topical report for that facility.

(continued)

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.2.1

Performance of the CHANNEL CHECK once ever 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.

Agreement criteria are determined by the plant staff, based on a combination of the channel instrument uncertainties, including isolation, indication, and reliability. If a channel is outside the match criteria, it may be an indication that the transmitter or the signal processing equipment has drifted outside its limit. If the channels are within the match criteria, it is an indication that the channels are OPERABLE. If the channels are normally off scale during times when surveillance is required, the CHANNEL CHECK will only verify that they are off scale in the same direction. Off-scale low current-loop channels are verified to be reading at the bottom of the range and not failed downscale.

The Surveillance Frequency, about once every shift, is based on operating experience that demonstrates the rarity of channel failure. Thus, performance of the CHANNEL CHECK guarantees that undetected overt channel failure is limited to 12 hours. Since the probability of two random failures in redundant channels in any 12-hour period is extremely low, the CHANNEL CHECK minimizes the chance of loss of protective function due to failure of redundant channels. The CHANNEL CHECK supplements less formal, but more frequent, checks of channel OPERABILITY during normal operational use of the displays associated with the LCO required channels.

In the case of functions that trip on a combination of several measurements, high steam line flow for example, the CHANNEL CHECK must be performed on each input.

(continued)

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.2.2

SR 3.3.2.2 is the performance of an ACTUATION LOGIC TEST. The [SSPS] is tested every 31 days on a STAGGERED TEST BASIS, using the [semiautomatic tester]. The continuity check does not have to be performed for this Surveillance as indicated by Note in the SR. [For this facility, the basis for excluding continuity test and the SR accomplishing this are as follows:] The train being tested is placed in the bypass condition, thus preventing inadvertent actuation. Through the [semiautomatic tester], all possible logic combinations, with and without applicable permissive, are tested for each protection function. In addition, the master relay coil is pulse tested for continuity. This verifies that the logic modules are OPERABLE and there is an intact voltage signal path to the master relay coils. The time allowed for the testing (4 hours) and the Surveillance Frequency are justified in Reference 7.

SR 3.3.2.3

SR 3.3.2.3 is the performance of an ACTUATION LOGIC TEST as described in SR 3.3.2.2, except that the continuity check MUST be performed for SR 3.3.2.3. [For this facility, the continuity check constitutes of the following:] This test is also performed every 31 days on a STAGGERED TEST BASIS. The surveillance interval and the time allowed for the testing (4 hours) are justified in Reference 7.

SR 3.3.2.4

SR 3.3.2.4 is the performance of a MASTER RELAY TEST. The MASTER RELAY TEST is the energizing of the master relay, verifying contact operation and a low voltage continuity check of the slave relay coil. Upon master relay contact operation, a low voltage is injected to the slave relay coil. This voltage is insufficient to pick up the slave relay, but large enough to demonstrate signal path continuity. This test is performed every 31 days on a STAGGERED TEST BASIS. The time allowed for the testing (4 hours) and the surveillance interval are justified in Reference 7.

(continued)

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.2.5

SR 3.3.2.5 is the performance of an ANALOG CHANNEL OPERATIONAL TEST. This test is a periodic check of the analog process control equipment while the unit is at power. When the channel is placed in the test condition, the input to the [SSPS] is changed to the tripped condition and the input from the transmitter is removed. This allows a test signal to be introduced into the instrument loop. The input to the bistable can be measured, thus noting the accuracy of the signal conditioning of the process control modules upstream. The trip setpoint of the bistable can be determined by varying the input and observing the bistable test lamp. "As found" and "as left" values for bistable trip setpoint 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 test "as left" values must be consistent with the drift allowance used in the setpoint analysis. Recalibration restores OPERABILITY of a 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. Individual process control modules may be tested in place using multiple sets of test jacks, or by module removal and verification in a calibration laboratory. If individual modules are checked, a verification of the loop accuracy is necessary to satisfy the statistical analyses assumptions. This test is performed every 92 days and is justified in Reference 7.

SR 3.3.2.6

SR 3.3.2.6 is the performance of a SLAVE RELAY TEST. The SLAVE RELAY TEST is the energizing of the slave relays. Contact operation is verified in one of two ways. Actuation equipment which may be operated in the design mitigation MODE is either allowed to function or is placed in a condition where the relay contact operation can be verified without operation of the equipment. Actuation equipment which may not be operated in the design mitigation MODE is prevented from operation by the slave relay test circuit. For this latter case, contact operation is verified by a continuity check of the circuit containing the slave relay.

(continued)

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

This test is performed every 92 days. The time allowed for the testing (4 hours) and the Surveillance Frequency are justified in Reference 7.

SR 3.3.2.7

SR 3.3.2.7 is the performance of a TRIP ACTUATING DEVICE OPERATIONAL TEST. This test is a check of the Loss of Offsite Power and Undervoltage Reactor Coolant Pump functions. Each function is tested up to, and including, the master transfer relay coils.

The test also includes trip devices that provide actuation signals directly to the [SSPS]. For these tests, the relay trip setpoints are verified and adjusted as necessary. The Frequency is justified in Reference 7.

SR 3.3.2.8

SR 3.3.2.8 is the performance of a TRIP ACTUATING DEVICE OPERATIONAL TEST. This test is a check of the Manual Actuation functions and is performed every [18] months. Each Manual Actuation function is tested up to, and including, the master relay coils. In some instances, the test includes actuation of the end device (i.e., pump starts, valve cycles, etc).

The test also includes trip devices that provide actuation signals directly to the [SSPS], bypassing the analog process control equipment. For these tests, the relay trip setpoints are verified and adjusted as necessary. The Frequency is justified in Reference 7.

SR 3.3.2.9

SR 3.3.2.9 is the performance of a CHANNEL CALIBRATION. CHANNEL CALIBRATION is a complete check of the instrument channel including the detector. The test verifies the channel responds to measured parameter with the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift between successive tests to ensure that the instrument channel remains operational 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

(continued)

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

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. Resistance temperature detector (RTD) channels may be calibrated in place using cross-calibration techniques, or in a test bath after removal from piping. For cross-calibration, at least one RTD should be replaced with a newly calibrated RTD during each refueling cycle to ensure accurate RTD cross-calibration. This replacement RTD must be the same model as the remaining RTDs. Using a newly calibrated RTD as a reference assures RTD signal drift continues to remain random rather than systematic, and is within the limits specified in the plant setpoint analysis. The replacement interval may be extended to alternate refueling if it is demonstrated that over the extended interval, the RTD's drift is random rather than systematic and is bounded by the plant-specific setpoint analyses assumptions. This determination may use results of statistical analysis of operating data and calibrating 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 in the setpoint analysis.

This SR is modified by a Note that this test should include verification that the time constants are adjusted to the prescribed values where applicable.

(continued)

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.2.10

This SR ensures that the ESF SYSTEM RESPONSE TIMES are verified on a STAGGERED TEST BASIS. The response time values are provided in the FSAR and are the maximum values assumed in the safety analyses. Individual component response times are not modeled in the analyses.

The analyses model the overall or total elapsed time, from the point at which the parameter exceeds the trip setpoint value at the sensor, to the point at which the equipment in both trains reaches the required functional state (e.g., pumps at rated discharge pressure, valves in full open or closed position). [Response time testing acceptance criteria for this facility are contained in the following document:]

The test may be performed in one measurement or overlapping segments, with verification that all components are measured.

The 18-month Frequency was developed because many surveillances can only be performed during a plant outage. Response time tests are conducted on an 18-month STAGGERED TEST BASIS. This results in the interval between successive tests of a given channel of n times 18 months, where n is the number of channels in the function. Testing of the final actuation devices, which make up the bulk of the response time, is included in the testing of each channel. Therefore, staggered testing results in response time verification of these devices every 18 months. The 18-month Frequency is based upon plant operating experience, which shows that random failures of instrumentation components causing serious response time degradation, but not channel failure, are infrequent occurrences. Response times cannot be determined at power since equipment operation is required.

This SR is modified with a Note indicating that the provisions of SR 3.0.4 are not applicable for the turbine-driven AFW pump. [At this facility, SR 3.0.4 is not applicable to this function for the following reasons:]

(continued)

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.2.11

SR 3.3.2.11 is the performance of a TRIP ACTUATING DEVICE OPERATIONAL TEST as described in SR 3.3.2.8, except that it is performed for the P-4 Reactor Trip Interlock, and the Frequency is once per reactor trip breaker cycle. This Frequency is based upon operating experience that undetected failure of the P-4 interlock sometimes occurs when the reactor trip breaker is cycled.

REFERENCES

1. [Unit Name] Updated FSAR, Section [ ], "[Engineered Safety Features]."
2. [Unit Name] Updated FSAR, Section [ ], "[Instrumentation and Controls]."
3. [Unit Name] Updated FSAR, Section [ ], "[Accident Analysis]."
4. Institute of Electrical and Electronic Engineers, IEEE-279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations," April 5, 1972.
5. [Unit Name], "[Plant-Specific RPS/ESFAS Setpoint Methodology Study.]"
6. Code of Federal Regulations, Title 10, Part 50.49, "Environmental Qualification of Electrical Equipment Important to Safety for Nuclear Power Plants."
7. WCAP-10271-P-A, Supplement 2, Rev. 1, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System," dated June 1990.
8. Title 10, Code of Federal Regulations, Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants."

B 3.3 INSTRUMENTATION

B 3.3.3 Post-Accident Monitoring (PAM) Instrumentation

BASES

BACKGROUND

Indications of plant variables are required by the control room operating personnel during accident situations to:

- Provide information required to permit the operator to take preplanned manual actions to accomplish safe plant shutdown;
- Determine whether the Reactor Trip System, engineered safety feature systems, manually initiated safety systems, and other systems important to safety are performing their intended functions (i.e., reactivity control, core cooling, maintaining Reactor Coolant System (RCS) integrity and maintaining containment OPERABILITY);
- Provide information to the operators that will enable them to determine the likelihood of a gross breach of the barriers to radioactivity release (i.e., fuel cladding, reactor coolant pressure boundary, and containment) and to determine if a gross breach of a barrier has occurred; and
- Provide information regarding the release of radioactive materials to allow for early indication of the need to initiate action necessary to protect the public and to estimate 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.

(continued)

(continued)

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 they may not have the necessary range to track the course of the accident; consequently, multiple instruments with overlapping ranges may be necessary. It is essential that degraded conditions and their magnitude be identified so the operators can take actions that are available to mitigate the consequences. It is not intended that operators be encouraged to prematurely circumvent systems important to safety, but that they be adequately informed in order that unplanned actions can be taken when necessary.

Examples of serious events that could threaten safety if conditions degrade are loss-of-coolant accidents (LOCAs), overpressure transients, anticipated operational occurrences that become accidents such as anticipated transients without scram, and reactivity excursions that result in releases of radioactive materials. Such events require that the operators understand, within a short time period, the ability of the barriers to limit radioactivity release (i.e., that they understand the potential for breach of a barrier or whether an actual breach of a barrier has occurred because of an accident in progress).

It is essential that the required instrumentation be capable of surviving the accident environment in which it is located for the length of time its function is required. It is, therefore, either designed to withstand the accident environment or to be protected by a local protected environment.

Variables for accident monitoring are selected to provide the essential information needed by the operator to determine if the plant safety functions are being performed. The availability of such instrumentation is important so that responses to corrective actions can be observed and the need for, and magnitude of, further actions can be determined.

These essential instruments are identified by plant-specific documents (Ref. 1) addressing the recommendations of

(continued)

(continued)

BASES (continued)

BACKGROUND (continued)

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 plant-specific implementation of Regulatory Guide 1.97 as Type A variables and Category 1 variables.

Type A variables are included in this LCO because they provide the primary information required to permit the control room operator to take specific manually controlled actions for which no automatic control is provided and that are required for safety systems to accomplish their safety functions for Design Basis Accidents (DBAs). Primary information is information that is essential for the direct accomplishment of the specified safety functions; it does not include those variables that are associated with contingency actions that may also be identified in written procedures. Because the list of Type A variables widely differs between plants, Table 3.3.3-1 contains no examples of Type A variables, except for those that may also be Category 1.

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

- Determine whether other systems important to safety are performing their intended functions;
- Provide information to the operators that will enable them to determine the likelihood of a gross breach of the barriers to radioactivity release; and
- Provide information regarding the release of radioactive materials to allow for early indication of the need to initiate action necessary to protect the public and to estimate the magnitude of any impending threat.

These key variables are identified by plant-specific Regulatory Guide 1.97 analyses. These analyses identified the plant-specific Type A variables and provided justification for deviating from the NRC-proposed list of Category 1 variables.

(continued)

(continued)

BASES (continued)

BACKGROUND (continued)

Table 3.3.3-1 provides a list of variables typical of those identified by plant-specific Regulatory Guide 1.97 analysis. [Table 3.3.3-1 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 are discussions of the specified instrument functions listed in Table 3.3.3-1. These discussions are intended as examples of what should be provided for each function when the plant-specific list is prepared.

1, 2. Power Range and Source Range Neutron Flux

Power Range and Source Range Neutron Flux indication is provided to verify reactor shutdown. The two ranges are necessary to cover the full range of flux that may occur post accident. [For this facility, the Power and Source Range Neutron Flux channels consist of the following:]

3, 4. Reactor Coolant System Hot and Cold Leg Temperature

Reactor Coolant System Hot and Cold Leg Temperatures are Category 1 variables provided for verification of core cooling and long-term surveillance.

Reactor outlet temperature inputs to the RPS are provided by two fast-response resistance elements and associated transmitters in each loop. The channels provide indication over a range of 32°F to 700°F.

5. Reactor Coolant System Pressure (Wide Range)

RCS wide-range pressure is a Category 1 variable provided for verification of core cooling and RCS integrity long-term surveillance.

(continued)

(continued)

BASES (continued)

BACKGROUND (continued)

Wide-range RCS loop pressure is measured by pressure transmitters with a span of 0-3000 psig. The pressure transmitters are located inside of containment. Redundant monitoring capability is provided by two trains of instrumentation. Control room indications are provided through the inadequate core cooling (ICC) plasma display. The ICC plasma display is the primary indication used by the operator during an accident. Therefore, the accident monitoring specification deals specifically with th portion of the instrument channel.

In some plants, RCS pressure is a Type A variable because the operator uses this indication to monitor the cooldown of the RCS following a steam generator (SG) tube rupture or small-break LOCA. Operator actions to maintain a controlled cooldown, such as adjusting SG pressure or level, would use this indication. Furthermore, RCS pressure is one factor that may be used in decisions to terminate reactor coolant pump operation.

6. Reactor Vessel Water Level

Reactor Vessel Water Level is provided for verification and long-term surveillance of core cooling.

The Reactor Vessel Level Monitoring System provides a direct measurement of the collapsed liquid level above the fuel alignment plate. The collapsed level represents the amount of liquid mass which is in the reactor vessel above the core. Measurement of the collapsed water level is selected because it is a direct indication of the water inventory.

The collapsed level is obtained over the same temperature and pressure range as the saturation measurements, thereby encompassing all operating and accident conditions where it must function. Also, it functions during the recovery interval. Therefore, it is designed to survive the high steam temperature, which may occur during the preceding core recovery interval.

(continued)

(continued)

BASES (continued)

BACKGROUND (continued)

The level range extends from the top of the vessel down to the top of the fuel alignment plate. The response time is short enough to track the level during small-break LOCA events. The resolution is sufficient to show the initial level drop, the key locations near the hot-leg elevation, and the lowest levels just above the alignment plate. This provides the operator with adequate indication to track the progression of the accident and to detect the consequences of its mitigating actions or the functionality of automatic equipment.

A Heated Junction Thermocouple System measures reactor coolant liquid inventory with discrete heated junction thermocouple sensors located at different levels with a separator tube ranging from the top of the core to the reactor vessel head. The basic principle of system operation is the detection of a temperature difference between adjacent heated and unheated thermocouples.

The heated junction thermocouple sensor consists of a chromel-alumel thermocouple surrounded by a resistance wire heater (or heated junction) and another chromel-alumel thermocouple (or unheated junction) positioned 4-1/2 inches above the heater. In a fluid with relatively good heat transfer properties, the temperature difference between the adjacent thermocouples is very small. In a fluid with relatively poor heat transfer properties, the temperature difference between the thermocouples is large.

Two design features ensure proper operation under all thermal-hydraulic conditions. First, each heated junction thermocouple is shielded to avoid overcooling due to direct water contact during two-phase fluid conditions. The heated junction thermocouple with the splash shield is referred to as the heated junction thermocouple sensor. Second, each string of heated junction thermocouple sensors is enclosed in a separator tube that separates them from the turbulent liquid and vapor phases that surround the heated junction thermocouple during a reactor coolant inventory transient.

(continued)

(continued)

BASES (continued)

BACKGROUND (continued)

The separator tube creates a collapsed liquid level that the heated junction thermocouple sensors measure. This collapsed liquid level is directly related to the average liquid fraction of the fluid in the reactor head volume above the fuel alignment plate. This mode of direct in-vessel sensing reduces spurious effects due to pressure, fluid properties, and nonhomogeneities of the fluid medium. The string of heated junction thermocouple sensors and the separator tube is referred to as the heated junction thermocouple instrument. The Heated Junction Thermocouple System is composed of two channels of heated junction thermocouple instruments. Each heated junction thermocouple instrument is manufactured into a probe assembly. The probe assembly includes eight heated junction thermocouple sensors, a seal plug, and electrical connectors. The eight heated junction thermocouple sensors are electrically independent and are located at eight levels from the reactor vessel head to the fuel alignment plate.

7. Containment Sump Water Level (Wide Range)

Containment Sump Water Level is provided for verification and long-term Surveillance of RCS integrity. [For this facility, Containment Sump Water Level instrumentation consists of the following:]

8. Containment Pressure (Wide Range)

Containment Pressure (Wide Range) is provided for verification of RCS and containment OPERABILITY. [For this facility, Containment Pressure instrumentation consists of the following:]

9. Containment Isolation Valve Position

Containment Isolation Valve Position is provided for verification of Containment OPERABILITY. [For this facility, Containment Isolation Valve Position consists of the following:]

(continued)

(continued)

BASES (continued)

BACKGROUND (continued)

10. Containment Area Radiation (High Range)

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, Containment Area Radiation instrumentation consists of the following:]

11. Containment Hydrogen Concentration

Containment Hydrogen Concentration is provided to detect high hydrogen concentration conditions which represent a potential for containment breach. This variable is also important in verifying the adequacy of mitigating actions.

[For this facility, containment hydrogen instrumentation consists of the following:]

12. Pressurizer Level

Pressurizer water level is used to determine whether to terminate safety injection, if still in progress, or to re-initiate safety injection if it has been stopped. Knowledge of pressurizer water level is also used to verify the plant conditions necessary to establish natural circulation in the RCS and to verify that the plant is maintained in a safe shutdown condition. [For this facility, Pressurizer Level instrumentation consists of the following:]

13. Steam Generator Water Level

Steam Generator Water Level is provided to monitor operation of decay heat removal via the SGs. The Category 1 indication of SG level is the extended startup range level instrumentation. The extended startup range level covers a span of 6 to 394 inches above the lower tubesheet. The measured differential pressure is displayed in inches of water at 68°F.

(continued)

(continued)

BASES (continued)

BACKGROUND (continued)

Temperature compensation of this indication is performed manually by the operator. Redundant monitoring capability is provided by two trains of performed manually by the operator. Redundant monitoring capability is provided by two trains of instrumentation. The uncompensated level signal is input to the plant computer, a control room indicator, and the Emergency Feedwater Control System.

At some plants, operator action is based on the control room indication of SG level. The RCS response during a design basis small break LOCA is dependent on the break size. For a certain range of break sizes, the boiler-condenser mode of heat transfer is necessary to remove decay heat. Extended startup range level is a Type A variable because the operator must manually raise and control SG level to establish boiler-condenser heat transfer. Operator action is initiated on a loss of subcooled margin. Feedwater flow is increased until the indicated extended startup range level reaches the boiler-condenser setpoint.

14. Condensate Storage Tank Level

Condensate Storage Tank (CST) Level is provided to ensure water supply for auxiliary feedwater (AFW). The CST provides the assured safety grade water supply for the AFW System. The CST consists of two identical tanks connected by a common outlet header. Inventory is monitored by a 0 to 144 inch level indication for each tank. Condensate Storage Tank Level is displayed on a control room indicator, strip chart recorder, and plant computer. In addition, a control room annunciator alarms on low level.

At some plants, Condensate Storage Tank Level is considered a Type A variable because the control room meter and annunciator are considered the primary indication used by the operator.

The DBAs which require AFW are the loss of electric power, steam line break (SLB), and small-break LOCA. The CST is the initial source of water for the AFW System. However, as the CST is depleted, manual

(continued)

(continued)

BASES (continued)

BACKGROUND (continued)

operator action is necessary to replenish the CST or align suction to the AFW pumps from the hotwell.

15, 16, 17, 18. Core Exit Temperature

Core Exit Temperature is provided for verification and long-term surveillance of core cooling.

An evaluation was made of the minimum number of valid core exit thermocouples necessary for ICC detection. The evaluation determined the reduced complement of core exit thermocouples necessary to detect initial core recovery and trend the ensuing core heatup. The evaluations account for core nonuniformities including incore effects of the radial decay power distribution and excore effects of condensate runback in the hot legs and nonuniform inlet temperatures. Based on these evaluations, adequate or ICC detection is assured with two valid core exit thermocouples per quadrant.

The design of the Incore Instrumentation System includes a Type K (chromel-alumel) thermocouple within each of the 56 incore instruments detector assemblies. The junction of each thermocouple is located a few inches above the fuel assembly inside a structure which supports and shields the incore instrument detector assembly string from flow forces in the outlet plenum region. These core exit thermocouples monitor the temperature of the reactor coolant as it exits the fuel assemblies.

The core exit thermocouples have a usable temperature range from 32°F to up to 2300°F, although accuracy is reduced at temperatures above 1800°F.

19. Auxiliary Feedwater Flow

Auxiliary Feedwater Flow is provided to monitor operation of decay heat removal via the SGs.

The Auxiliary Flow Water Flow to each SG is determined from a differential pressure measurement calibrated for a range of 0-1200 gpm. Redundant monitoring

(continued)

(continued)

BASES (continued)

BACKGROUND (continued)

capability is provided by two independent trains of instrumentation for each SG. Each differential pressure transmitter provides an input to a control room indicator and the plant computer. Since the primary indication used by the operator during an accident is the control room indicator, the accident monitoring specification deals specifically with this portion of the instrument channel.

At some plants, Auxiliary Feedwater Flow is a Type A variable because operator action is required to throttle flow during an SLB accident in order to prevent the AFW pumps from operating in runout conditions. AFW flow is also used by the operator to verify that the AFW System is delivering the correct flow to each SG. However, the primary indication used by the operator to ensure an adequate inventory is SG level.

APPLICABLE SAFETY ANALYSES

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

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

(continued)

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

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

PAM instrumentation that meets the definition of Type A in Regulatory Guide 1.97 satisfies the requirements of Criterion 3 of the NRC Interim Policy Statement. Category 1 PAM instrumentation must be retained in Technical Specifications (TS) because they are intended to assist operators in minimizing the consequences of accidents. Therefore, Category 1 variables are important in reducing public risk.

LCO

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

- Permit the operator to take pre-planned manual actions to accomplish safe plant shutdown;
- Determine whether other systems important to safety are performing their intended functions;
- Provide information to the operators that will enable them to determine the likelihood of a gross breach of the barriers to radioactivity release and to determine if a gross breach of a barrier has occurred; and
- Provide information regarding the release of radioactive materials to allow for early indication of the need to initiate action necessary to protect the public to estimate the magnitude of any impending threat.

Two channels are required to be OPERABLE for most functions. Two OPERABLE channels ensure no single failure prevents the operators from being presented the information necessary for them to determine the safety status of the plant and to bring the plant to and maintain it in a safe condition following that accident. This includes failures within

(continued)

(continued)

BASES (continued)

LCO (continued)

either the PAM instrumentation, its auxiliary supporting features, or its power sources concurrent with the failures that are a condition of or result from a specific accident.

Furthermore, provision of two channels allows a CHANNEL CHECK during the post-accident phase to confirm the validity of displayed information. More than two channels may be required at some plants if the Regulatory Guide 1.97 analysis determined that failure of one accident monitoring channel results in information ambiguity (that is, the redundant displays disagree) that could lead operators to defeat or fail to accomplish a required safety function.

This might also be accomplished by providing an independent channel to monitor a different variable that bears a known relationship to the multiple channels (addition of a diverse channel).

In Table 3.3.3-1 the exceptions to the two-channel requirement are Core Exit Temperatures, loop- and SG-related variables, and Containment Isolation Valve Position.

Two OPERABLE channels of Core Exit Temperature are required for each channel in each quadrant to provide indication of radial distribution of the coolant temperature rise across representative regions of the core. Power distribution symmetry was considered in determining the specific number and locations provided for diagnosis of local core problems. Therefore, two randomly selected thermocouples are not sufficient to meet the two thermocouples per channel requirement in any quadrant. The two thermocouples in each channel must meet the additional requirement that one is located near the center of the core and the other near the core perimeter, such that the pair of Core Exit Temperatures indicate the radial temperature gradient across their core quadrant. Plant-specific evaluations in response to Item II.F.2 of NUREG-0737 should have identified the thermocouple pairings that satisfy these requirements. Two sets of two thermocouples ensures a single failure will not disable the ability to determine the radial temperature gradient.

(continued)

(continued)

BASES (continued)

LCO (continued)

For loop- and SG-related variables, the required information is individual loop temperature and individual SG level. In these cases, two channels are required OPERABLE for each loop of the SG to redundantly provide the necessary information.

In the case of Containment Isolation Valve Position, the important information is the status of the containment penetrations. The LCO requires one position indicator for each active containment isolation valve. This is sufficient to redundantly verify the isolation status of each isolable penetration either via indicated status of the active valve and prior knowledge of passive valve or system boundary status. If a normally active containment isolation valve 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.

A PAM channel is OPERABLE when:

- All channel components necessary to provide the required indication are functional;
- Channel measurement uncertainties are known (via test, analysis, or design information) to be sufficiently small such that measurement and indication errors will not mislead operators into actions that would challenge plant safety limits; and
- Required Surveillance testing is current and has demonstrated performance within each Surveillance test's acceptance criteria.

LCO Table 3.3.3-1 is for illustration purposes only. Plant-specific TS tables 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:]

(continued)

(continued)

BASES (continued)

LCO (continued)

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

APPLICABILITY

The PAM instrumentation LCO is applicable in MODES 1, 2, and 3. These variables are related to the diagnosis and preplanned actions required to mitigate DBAs. The applicable DBAs are assumed to occur in MODES 1, 2, and 3. In MODES 4, 5, and 6, plant conditions are such that the likelihood of an event that would require PAM instrumentation is low; therefore, the PAM instrumentation is not required to be OPERABLE in these MODES.

A Note has been added to indicate that the provisions of LCO 3.0.4 are not applicable to the functions contained in the LCO. A second Note provides clarification that, for this LCO, each function specified in Table 3.3.3-1 is treated as an independent entity with an independent Completion Time.

ACTIONS

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 a one or two of the channels leaves at least two channels OPERABLE to meet the LCO requirements. Therefore, for this example, Condition A need not be entered unless three channels fail. The 30-day Completion Time is based on operating experience and takes into account the remaining OPERABLE channel and the low probability of an event requiring PAM instrumentation during this interval.

(continued)

(continued)

BASES (continued)

ACTIONS (continued)

Condition B

With two required channels inoperable in one or more functions, at least one channel in each function should be restored to OPERABLE status within 7 days. The Completion Time of 7 days is based on the relatively low probability of an event requiring PAM instrument operation and the availability of alternate means to obtain the required information. Continuous operation with two required channels inoperable is not acceptable because the alternate indications may not fully meet all performance qualification requirements applied to the PAM instrumentation. Therefore, requiring restoration of at least one inoperable channel limits the risk that the PAM function will be in a degraded condition should an accident occur.

Condition C

Required Action C.1 directs the operator to follow the directions given in Table 3.3.3-1 for entering either Condition D or E immediately.

Condition D

For the majority of functions in Table 3.3.3-1, 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 6 hours and MODE 4 within 12 hours. The Completion Times are reasonable, based on operating experience, to reach the required MODES from full power in an orderly manner and without challenging plant systems.

Condition E

At this facility, alternate means of monitoring [Reactor Vessel Water Level and 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 down the plant but rather to follow the directions of Specification 5.9.2.c., "Special Reports," in the Administrative Controls section of the TS. The report

(continued)

(continued)

BASES (continued)

ACTIONS (continued)	provided to the NRC should discuss the alternate means invoked, describe the degree to which the alternate means are equivalent to the installed PAM channels, justify the areas in which they are not equivalent, and provide a schedule for restoring the normal PAM channels. [At this facility, the alternate monitoring provisions consist of the following:]
SURVEILLANCE REQUIREMENTS	The following SRs apply to each PAM instrumentation function in Table 3.3.3-1: <u>SR 3.3.3.1</u> Performance of the CHANNEL CHECK once every 31 days ensures that a gross instrumentation failure has not occurred. A CHANNEL CHECK is a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION. The high radiation instrumentation should be compared to similar plant instruments located throughout the plant. If the radiation monitor employs keep-alive sources or check sources operable from the control room, the CHANNEL CHECK should also note the detector's response to these sources. Agreement criteria are determined by the plant staff, based on a combination of the channel instrument uncertainties, including isolation, indication, and readability. If a channel is outside the match criteria, it may be an indication that the transmitter or the signal-processing equipment has drifted outside its limit. If the channels are within the match criteria, it is an indication that the channels are OPERABLE. If the channels are normally off scale during times when Surveillance is required, the

(continued)

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

CHANNEL CHECK will verify only that they are off scale in the same direction. Off-scale low current-loop channels are verified to be reading at the bottom of the range and not failed downscale.

The Surveillance Frequency of 31 days is based on plant operating experience with regard to channel OPERABILITY and drift, which demonstrates that failure of more than one channel of a given function in any 31-day interval is a rare event. The CHANNEL CHECK supplements less formal, but more frequent, checks of channel OPERABILITY during normal operational use of those displays associated with the required channels of this LCO.

SR 3.3.3.2

A CHANNEL CALIBRATION is performed every 18 months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument channel including the detector. The test verifies that the channel responds to measured parameter 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 ensures 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 an alternate refueling 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 calibrating data from similar plants using the same model of RTD in the same environmental conditions.

(continued)

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)	The Surveillance Frequency is based on the assumption of an 18-month calibrating interval in the determination of the magnitude of equipment drift.
REFERENCES	1. [Plant-specific document (e.g., FSAR, NRC Regulatory Guide 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. 3. NUREG-0737, Supplement 1, "Clarification of TMI Action Plan Requirements," U.S. Nuclear Regulatory Commission. 4. Title 10, Code of Federal Regulations, Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants."

B 3.3 INSTRUMENTATION

B 3.3.4 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 that the control room becomes inaccessible or a fire destroying all equipment in one facility fire area disables critical control room instruments or controls. A safe shutdown condition is defined as MODE 3. With the facility in MODE 3, the Auxiliary Feedwater (AFW) System and the steam generator (SG) safety valves or the SG atmospheric dump valves (ADVs) can be used to remove core decay heat and meet all safety requirements. The long-term supply of water for the AFW System and the ability to borate the Reactor Coolant System (RCS) from outside the control room allows extended operation in MODE 3.

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 control and instrumentation functions ensures there is sufficient information available on selected plant parameters to place and maintain the plant in MODE 3 should the control room become inaccessible or critical control room displays or controls become unavailable.

APPLICABLE SAFETY ANALYSES

The Remote Shutdown System is required to provide equipment at appropriate locations outside the control room with a

(continued)

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

capability to promptly shut down and maintain the plant in a safe condition in MODE 3.

Furthermore, in the event of a fire in any one plant fire area, the Remote Shutdown System is designed to ensure one train of systems necessary to achieve and maintain MODE 3 conditions from either the control room or emergency control station(s) is OPERABLE. The criteria governing the design of the Remote Shutdown System are 10 CFR 50, Appendix A, GDC 19 (Ref. 1) and 10 CFR 50, Appendix R (Ref. 2).

Specific system requirements are presented in Reference 1 and the NRC staff-approved plant-specific fire protection topical report.

The Remote Shutdown System is considered an important contributor to the reduction of plant risk to 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 control, typically required are listed in Table B 3.3.4-1. 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 licensing basis as described in the NRC plant-specific Safety Evaluation Report (SER). Generally, two divisions are required OPERABLE. However, only one channel per a 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 required for:

- Core reactivity control (initial and long term);
- RCS pressure control;

(continued)

(continued)

BASES (continued)

LCO (continued)

- Decay heat removal via the AFW System and the SG safety valves or SG ADVs;
- RCS inventory control via charging flow; and
- Safety support systems for the above functions, including service water, component cooling water, and onsite power, including the diesel generators.

A division of a Remote Shutdown System is OPERABLE if all instrument and control channels needed to support the function are OPERABLE in that division. In some cases, Table B 3.3.4-1 may indicate that the required information or control capability is available from several alternate sources. In these cases, the function is OPERABLE as long as one channel of any of the alternate information or control sources is OPERABLE.

Remote shutdown instrumentation channels are OPERABLE when:

- All channel components necessary to provide the required indication are functional,
- Channel measurement uncertainties are known (via test, analysis, or design information) to be sufficiently small such that measurement and indication errors will not mislead remote shutdown 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 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.

(continued)

(continued)

BASES (continued)

LCO (continued)

The remote shutdown equipment covered by this LCO does not need to be in operation to be considered OPERABLE. This LCO is intended to ensure the equipment will be OPERABLE if 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 3.3.4-1, are OPERABLE.

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

[For this facility, those required support systems which, upon their failure, do not 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, 5, or 6. In these MODES, the facility is already subcritical and in a condition of reduced RCS energy. Under these conditions, considerable time is available to restore necessary instrument control functions if control room instruments or controls become unavailable.

A Note has been added to indicate that LCO 3.0.4 does not apply to the Remote Shutdown System. 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 are equally applicable to startup conditions as to continued operation in MODE 1 or 2. Furthermore, Remote Shutdown System equipment can generally be repaired during operation without significant risk of spurious trip.

(continued)

BASES (continued)

ACTIONS

Condition A

Condition A addresses the situation where one or more divisions of any required function(s) or channel(s) of a function of the Remote Shutdown System is inoperable. This includes any function listed in Table B 3.3.4-1, as well as the control and transfer switches.

When a division includes a function that only requires a channel to be OPERABLE, the failure of the single channel constitutes the failure of the function, and, as a consequence, the division becomes inoperable.

The Required Action is to restore the divisions to OPERABLE status within 30 days. The Completion time is based on operating experience and takes into account the remaining OPERABLE division and the low probability of an event that would require evacuation of the control room.

A Note has been added to provide clarification 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 in 30 days, the prudent action is to place the plant in a MODE in which the LCO requirements are not applicable. This is done by placing the plant in at least MODE 3 within 6 hours and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required MODE from full power in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.3.4.1

Performance of the CHANNEL CHECK once every 31 days ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value.

(continued)

(continued)

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 of 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 Surveillance is required, the CHANNEL CHECK will verify only 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 operating experience related to channel OPERABILITY and drift, which demonstrates that failure of more than one channel of a given function in any 31-day interval is a rare event. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels, which occur during normal operational use of the displays associated with this LCO's required channels.

SR 3.3.4.2

SR 3.3.4.2 verifies that each required Remote Shutdown System transfer switch and control circuit perform their intended functions. This verification is performed from the remote shutdown panel and locally, as appropriate. This will ensure that if the control room becomes inaccessible, the unit can be placed and maintained in MODE 3 from the remote shutdown panel and the local control stations. The 18-month Frequency was developed considering it is prudent that these surveillances be performed only 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

(continued)

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

at power. Operating experience demonstrates that remote shutdown control channels usually pass the surveillance test when performed on the 18-month Frequency.

SR 3.3.4.3

CHANNEL CALIBRATION is a complete check of the instrument channel, including the detector. The test verifies that the channel responds to measured parameters with the necessary range and accuracy.

Transmitter "as found" and "as left" values are recorded and used to verify drift assumptions. Field transmitters may be calibrated in place, removed and calibrated in a laboratory, or replaced with an equivalent laboratory-calibrated unit.

Resistance temperature detector (RTD) channels may be calibrated in place using cross-calibration techniques, or in a test bath after removal from piping. For cross-calibration, at least one RTD should be replaced with a newly calibrated RTD during each refueling cycle to ensure accurate RTD cross-calibration. This replacement RTD must be the same model as the remaining RTDs. Using a newly calibrated RTD as a reference assures RTD signal drift continues to remain random rather than systematic and is within the limits specified in the plant setpoint analysis. The replacement interval may be extended to an alternate refueling 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 under 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.

SR 3.3.4.4

SR 3.3.4.4 is the performance of a TRIP ACTUATING DEVICE OPERATIONAL TEST every 18 months. This test should verify the OPERABILITY of the reactor trip breakers (RTBs) open and closed indication on the remote shutdown panel, by actuating the RTBs. The Surveillance Frequency of 18 months was

(continued)

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)	chosen because the RTBs cannot be exercised while the facility is at power. Operating experience has shown that these components usually pass the surveillance test when performed at an 18-month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.
REFERENCES	1. Title 10, Code of Federal Regulations, Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants." 2. Title 10, Code of Federal Regulations, Part 50, Appendix R, "Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979."

Table B 3.3.4-1 (page 1 of 2)
Remote Shutdown System Instrumentation and Controls

FUNCTION/INSTRUMENT OR CONTROL PARAMETER	LOCATION	REQUIRED NUMBER OF DIVISIONS

---NOTE---
This table is for illustration purposes only. It does not attempt to encompass every function used at every plant, but does contain the types of functions commonly found.

---

1. Reactivity Control		
a. Source Range Neutron Flux		[1]
b. Reactor Trip Breaker Position		[1 per trip breaker]
c. Manual Reactor Trip		[4]
2. Reactor Coolant System (RCS) Pressure Control		
a. Pressurizer Pressure <u>OR</u> RCS Wide Range Pressure		[1]
b. Pressurizer Power-Operated Relief Valve (PORV) Control and Block Valve Control		[1, controls must be for PORV & block valves on same line]
3. Decay Heat Removal via Steam Generators		
a. Reactor Coolant Hot Leg Temperature		[1 per loop]
b. Reactor Coolant Cold Leg Temperature		[1 per loop]
c. Auxiliary Feedwater Controls Condensate Storage Tank Level		[1]

(continued)

(continued)

Table B 3.3.4-1 (page 2 of 2)
Remote Shutdown System Instrumentation and Controls

FUNCTION/INSTRUMENT OR CONTROL PARAMETER	LOCATION	REQUIRED NUMBER OF DIVISIONS
d. Steam Generator (SG) Pressure		[1 per SG]
e. Steam Generator Level <u>OR</u> Auxiliary Feedwater Flow		[1 per SG]
4. Reactor Coolant System Inventory Control		
a. Pressurizer Level		[1]
b. Charging Pump Controls		[1]

B 3.3 INSTRUMENTATION

B 3.3.5 Boron Dilution Protection System (BDPS)

BASES

BACKGROUND

The primary purpose of the BDPS is to mitigate the consequences of the inadvertent addition of unborated primary-grade water into the Reactor Coolant System (RCS) when the reactor is in a shutdown condition, i.e., MODES [2], 3, 4, and 5.

The BDPS utilizes the two channels of source-range instrumentation. Each source-range channel provides a signal to both trains of BDPS. A plant computer is used to continuously record the counts per minute provided by these signals. At the end of each minute, an algorithm is used to compare the counts-per-minute value (flux rate) at any given time with the counts-per-minute value for each of the prior nine 1-minute flux rates. If the flux rate at any given minute is greater than or equal to twice the flux rate at any of the prior nine 1-minute intervals, the BDPS provides a signal to initiate mitigating actions.

Upon detection of a flux doubling by either source-range instrumentation train, an alarm is sounded to alert the operator and valve movement is automatically initiated to terminate the dilution and start boration. Valves which isolate the refueling water storage tank (RWST) are opened to supply 2000 ppm borated water to the suction of the charging pumps, and valves which isolate the Chemical and Volume Control System (CVCS) are closed to terminate the dilution.

The trip setpoints used in the bistables are based on the analytical limits discussed in Reference 1. The selection of these actuation 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 BDPS components which must function in harsh environments as defined in 10 CFR 50.49 (Ref. 4), ALLOWABLE VALUES are conservatively adjusted with respect to the analytical limits. [For this facility the methodology used to calculate the actuation setpoints, including their explicit uncertainties, is as

(continued)

(continued)

BASES (continued)

BACKGROUND (continued)

follows:] The actual nominal actuation setpoint entered into the bistable is normally still more conservative than the ALLOWABLE VALUE, to account for changes in random measurement errors detectable by a CHANNEL FUNCTIONAL TEST. One example of such a change in measurement error is drift during the Surveillance Frequency. If the measured setpoint does not exceed the ALLOWABLE VALUE, the bistable is considered OPERABLE. Setpoints in accordance with the ALLOWABLE VALUE will assure that Safety Limits (SLs) are not violated during inadvertent boron dilution events, and the consequences of the events will be acceptable, providing the plant is operated from within the LCOs at the onset of the event, and the equipment functions as designed, allowing for a single random active-component failure.

APPLICABLE SAFETY ANALYSES

The BDPS senses abnormal increases in source-range counts-per-minute (flux rate) and actuates CVCS and RWST valves to mitigate the consequences of an inadvertent boron dilution event as described in Reference 1. The criteria governing the design and operation of the BDPS is presented in 10 CFR 50, Appendix A, Section III (Ref. 3). Accident analyses rely on automatic BDPS actuation t mitigate the consequences of inadvertent boron dilution events. [For this facility, the capability for manual actuation of BDPS is as follows:]

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

LCO

The LCO requires all instrumentation performing a BDPS function to be OPERABLE. Failure of any instrument renders the affected channel(s) inoperable and reduces the reliability of the affected system functions. The specific criteria for determining channel OPERABILITY are discussed below:

1. A BDPS function will be initiated when necessary; and

(continued)

(continued)

BASES (continued)

LCO (continued)

2. Sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance.

The ALLOWABLE VALUES specified ensure that violation of any SLs does not occur and that the consequences of inadvertent boron dilution events will be acceptable. The ALLOWABLE VALUES are contained in SR 3.3.5.2.

[For this facility, the provisions of the design for bypasses and interlocks are as follows:]

Only the ALLOWABLE VALUES are specified for the BDPS actuation function in the LCO. Nominal trip setpoints are specified in the plant-specific setpoint calculations. The nominal setpoints are selected to ensure the setpoint measured by ANALOG CHANNEL OPERATIONAL 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 is consistent with the assumptions of the plant-specific setpoint calculations. Each ALLOWABLE VALUE specified is more conservative than the analytical limit assumed in the safety analysis in order to account for instrumentation uncertainties appropriate to the trip function. The uncertainties are defined in the plant-specific setpoint methodology.

The limit on the number of BDPS channels that must be OPERABLE exists to ensure that a single failure in one channel will not result in loss of the ability to automatically actuate the required system components.

The following BDPS actuation channels are considered OPERABLE when:

1. All channel components necessary to provide a BDPS 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

(continued)

(continued)

BASES (continued)

LCO (continued)

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

[Plant-specific Technical Specifications will include a discussion of operational bypasses and blocks if applicable and add the following statement: The associated operational bypass or block is not enabled except under the conditions specified by the LCO Applicability for the function.]

The BDPS LCO provides the requirements for OPERABILITY of the instrumentation and controls which operate to mitigate the consequences of a boron dilution event. Two redundant trains are required to provide protection against single failure.

Because the BDPS utilizes the source-range instrumentation as its detection system, the OPERABILITY of the detection system is also part of the OPERABILITY of the Reactor Trip System. The flux doubling algorithm, the alarms, and signals to the various valves all must be OPERABLE for each train in the system to be considered OPERABLE.

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

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

[For this facility, the supported systems impacted by the inoperability of the BDPS and the justification of whether or not each supported system is declared inoperable are as follows:]

APPLICABILITY

The BDPS must be OPERABLE in MODES [2], 3, 4, and 5 as the safety analysis identifies this system as the primary means to mitigate an inadvertent boron dilution of the RCS.

The BDPS OPERABILITY requirements are not applicable in MODES 1 [and 2], because an inadvertent boron dilution would be terminated by a source-range trip, a trip on the Power

(continued)

(continued)

BASES (continued)

APPLICABILITY (continued)

Range Neutron Flux--High (low setpoint nominally 25% RTP), or Overtemperature ΔT.

In MODE 6, a dilution event is precluded by locked valves that isolate the RCS from the potential source of unborated water (LCO 3.9.2, "Unborated Water Source Isolation Valves)."

The Applicability is modified by a Note that allows the flux-doubling signal to be blocked during startup in MODES 2 and 3. Blocking the flux-doubling signal is acceptable during startup while in MODE 3, provided the reactor trip breakers are closed with the intent to withdraw rods for startup.

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, the channel must be declared inoperable immediately, and the appropriate Conditions from [ ] 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 BDPS bistable is found inoperable, then the function which that channel provides must be declared inoperable and the LCO Condition entered for the particular protection function affected.

(continued)

(continued)

ACTIONS (continued)

Condition A

With one train of BDPS OPERABLE, Required Action A.1 requires that the inoperable train must be restored to OPERABLE status within 72 hours. In this condition, the remaining BDPS train is adequate to provide protection. The 72-hour Completion Time is based on the BDPS function and is consistent with Engineered Safety Feature System Completion Times for loss of one redundant train. Also, the remaining OPERABLE train provides continuous indication of core power status to the operator, has an alarm function, and sends a signal to both trains of BDPS to assure system actuation.

Condition B

With two trains inoperable, or the Required Action and associated Completion Time of Condition A not met, the initial action (Required Action B.1) is to suspend all actions involving positive reactivity additions immediately. This includes withdrawal of control or shutdown rods and intentional boron dilution. A Completion Time of 1 hour is provided to restore one train to OPERABLE status. The justification for the 1-hour Completion Time is presented under Required Action B.3.1.

As an alternate to restoring one train to OPERABLE status (Required Action B.2), Required Action B.3.1 requires valves (Required Action A.2 of LCO 3.9.2) listed in LCO 3.9.2 to be secured to prevent the flow of unborated water into the RCS. Once it is recognized that two trains of the BDPS are inoperable, the operators will be aware of the possibility of a boron dilution, and the 1-hour Completion Time is adequate to complete the requirements of LCO 3.9.2.

Required Action B.3.2 accompanies Required Action B.3.1 to verify the SHUTDOWN MARGIN (SDM) within 1 hour and once per 12 hours thereafter. This backup action is intended to confirm that no unintended boron dilution has occurred while the BDPS was inoperable, and that the required SDM has been maintained. The specified Completion Time takes into consideration sufficient time for the initial determination of SDM, and other information available in the control room related to SDM.

(continued)

(continued)

BASES (continued)

ACTIONS (continued)

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 train within a Completion Time of 1 hour. The specified Completion Time is sufficient for plant operations personnel to make this determination.

Required Action C.1 ensures that those identified Required Actions associated with supported systems impacted by the inoperability of a train associated with each BDPS function have been initiated. This can be accomplished by entering the supported systems' LCOs or independently as a group of Required Actions needed to be initiated every time Condition C is entered. [For this facility, the identified supported systems' Required Actions associated with each BDPS function are as follows:]

Required Action C.2 verifies that all required support or supported features associated with the other redundant trains 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 BDPS functions are subject to an ANALOG CHANNEL OPERATIONAL TEST and a CHANNEL CALIBRATION.

SR 3.3.5.1

SR 3.3.5.1 requires the performance of an ANALOG CHANNEL OPERATIONAL TEST every 92 days, to ensure that each train of the BDPS and associated trip setpoints are fully operational. This test shall include verification that the boron dilution alarm setpoint is equal to or less than an increase of twice the count rate within a 10-minute period. The Surveillance Frequency of 92 days is consistent with the requirements for source-range channels in Reference 2.

(continued)

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

"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" values must be consistent with the drift allowance used in the setpoint analysis. Recalibration restores OPERABILITY of an otherwise functional component that does not meet these criteria. However, repeated failures of the same channel over a relatively small number of test intervals must be considered as potentially indicating a deterministic failure which cannot be corrected by recalibration.

SR 3.3.5.2

SR 3.3.5.2 is the performance of a CHANNEL CALIBRATION every 18 months. CHANNEL CALIBRATION is a complete check of the instrument channel including the detector. The test verifies the channel responds to 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. For BDPS, the CHANNEL CALIBRATION shall include verification that on a simulated or actual boron dilution flux-doubling signal the centrifugal charging pump suction valves from the RWST open, and the normal CVCS volume control tank discharge valves close in the required closure time of ≤ 20 seconds.

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.

(continued)

(continued)

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.

The Surveillance Frequency is based on the assumption of an 18-month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

REFERENCES

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

2. WCAP-10271-P-A, Supplement 2, Revision 1, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System," June 1990.

3. Title 10, Code of Federal Regulations, Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants."

4. Title 10, Code of Federal Regulations, Part 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants."

B 3.3 INSTRUMENTATION

B 3.3.6 Miscellaneous Safeguards Actuations

BASES

BACKGROUND

The actuation instrumentation included in this LCO initiates safety systems that perform support functions for engineered safety features (ESFs) or are required to minimize radioactive release during an accident, but are not otherwise included in LCO 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation." Specifically included are those non-Nuclear Steam Supply System (non-NSSS) functions which because of differences in purpose, design, or operating requirements are not included in LCO 3.3.2. Details of this LCO are for illustration only. Individual plants shall include those functions and LCO requirements applicable to them. The actuation instrumentation of this LCO may include signals from the [Solid State Protection System (SSPS)] or from plant instrumentation unrelated to the [SSPS].

1. Emergency Diesel Generator Loss of Power Start

The emergency diesel generators (EDGs) provide a source of emergency power when offsite power is either unavailable or is insufficiently stable to allow safe plant operation. Undervoltage protection will generate a loss of power start (LOPS) in the event a loss of voltage or degraded voltage condition occurs in the switchyard. There are two LOPS, one for each 4.16 kV vital bus.

Four undervoltage relays with [inverse time] characteristics are provided on each 4160 Class 1E instrument bus for the purpose of detecting a sustained undervoltage condition or a loss-of-bus voltage. The relays are combined in a [two-out-of-four] logic to generate an LOPS if the voltage is below 75% for a short time or below 90% for a long time. The LOPS-initiated actions are described in Reference 1.

(continued)

(continued)

BASES (continued)

BACKGROUND (continued)

Trip Setpoints and ALLOWABLE VALUES

The trip setpoints used in the bistables are based on the analytical limits presented in Reference 2. The selection of these trip setpoints is such that adequate protection is provided when all sensor and processing time delays are taken into account. To allow for calibration tolerances, instrumentation uncertainties, and instrument drift, ALLOWABLE VALUES specified in Table 3.3.6-1 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 1.

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 (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.

The undervoltage protection scheme has been designed to protect the plant from spurious trips caused by the offsite power source. This is made possible by the [inverse voltage-time] characteristics of the relays used. A complete loss of offsite power will result in approximately a [1]-second delay in LOPS actuation. The EDG starts and is available to accept loads within a [10]-second time interval on Engineered Safety Feature Actuation System (ESFAS) or LOPS. Emergency power is established within the maximum time delay assumed for each event analyzed in the accident analysis (Ref. 2).

Since there are four protective channels in a two-out-of-four trip logic for each division of the 4160 V power supply, no single failure will cause or prevent protective system actuation. This arrangement meets IEEE-279-1971 criteria (Ref. 4).

(continued)

(continued)

BASES (continued)

BACKGROUND (continued)

2. Fuel Building Air Cleanup Actuation System

The Fuel Building Air Cleanup System (FBACS) ensures that radioactive materials in the fuel building atmosphere following a fuel handling accident or a loss-of-coolant accident (LOCA) are filtered and adsorbed prior to exhausting to the environment. The system is described in the Bases for LCO 3.7.15, "Fuel Building Air Cleanup System (FBACS)." The system initiates filtered ventilation of the fuel building automatically following receipt of a high radiation signal (gaseous or particulate) or a safety injection (SI) signal. Initiation may also be performed manually as needed from the main control room. LCO 3.3.6 contains requirements only for the logic that actuates fuel building air cleanup. LCO 3.3.2, "ESFAS Instrumentation," and LCO 3.3.7, "Radiation Monitoring Instrumentation," provide LCO requirements on the measurement channels that input to this logic.

[At this facility, the Fuel Building Air Cleanup System Actuation logic and trip activation devices functions as follows:]

APPLICABLE SAFETY ANALYSES

1. Emergency Diesel Generator Loss of Power Start

The EDG LOPS is required for the ESFAS to function in any accident with a loss of offsite power. Accident analyses credit the loading of the EDG based on the loss of offsite power during a LOCA. The actual EDG start has historically been associated with the ESFAS actuation. The diesel loading has been included in the delay time associated with each safety system component requiring EDG-supplied power following a loss of offsite power. The analysis assumes a non-mechanistic EDG loading, which does not explicitly account for each individual component of loss-of-power detection and subsequent actions. [At this facility, the total actuation time for the limiting systems is as follows:] This delay time includes contributions from the EDG start, EDG loading, and SI System component actuation. The response of the EDG to a loss of power must be demonstrated to fall within this

(continued)

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

analysis response time when including the contributions of all portions of the delay.

The required channels of LOPS, in conjunction with the ESF Systems powered from the EDGs, provide plant protection in the event of any of the analyzed accidents discussed in Reference 2, in which a loss of offsite power is assumed. EDG LOPS channels are required to meet the redundancy and testability requirements of 10 CFR 50, Appendix A, GDC 21 (Ref. 3).

The delay times assumed in the safety analysis for the ESF equipment include the 10-second EDG start delay, and the appropriate sequencing delay, if applicable. The response of the EDG to a loss of power must be demonstrated to fall within this analysis response time. The response times for ESFAS-actuated equipment in LCO 3.3.2 include the appropriate EDG loading and sequencing delay.

2. Fuel Building Air Cleanup Actuation System

[At this facility, high radiation provides protection for the following accidents:]

The OPERABILITY of the Fuel Building Air Cleanup Actuation System is necessary to meet the assumptions of the safety analyses and provide a filtered exhaust path from the fuel building within limits of 10 CFR 100.

The miscellaneous safeguards actuations satisfy Criterion 3 of the NRC Interim Policy Statement.

LCO

The EDG LOPS and Fuel Building Air Cleanup Actuation Systems are OPERABLE when the following operational criteria are met:

- All channel automatic and manual initiation logic and components necessary to provide functional capability are in service;

(continued)

(continued)

BASES (continued)

LCO (continued)

- Channel measurement uncertainties are known via test, analysis, or design information to be within the assumptions of the setpoint calculations; and
- Required surveillance testing is current and has demonstrated performance within each surveillance test's criteria.

1. Emergency Diesel Generator Loss of Power Start

The LCO for the LOPS requires that four channels per bus of each LOPS instrumentation function be OPERABLE in MODES 1, 2, 3, and 4. The LOPS instrumentation supports safety systems associated with the ESFAS. In MODES 5 and 6, the channels must be OPERABLE whenever the associated EDG is required to be OPERABLE to ensure that the automatic start of the EDG is available when needed.

Loss of LOPS function could result in the delay of safety systems initiation when required. This could lead to violation of the SLs during certain AOOs, or unacceptable consequences during accidents. During the loss of offsite power, which is an AOO, the EDG powers the motor-driven auxiliary feedwater pumps. Failure of these pumps to start would leave only one turbine-driven pump, as well as an increased potential for a loss of decay-heat removal through the secondary system.

Only ALLOWABLE VALUES are specified for each function in the LCO. Nominal trip setpoints are specified in the plant-specific setpoint calculations. The nominal setpoints are selected to ensure that the setpoint measured by the ANALOG CHANNEL OPERATIONAL TESTS does not exceed the ALLOWABLE VALUE if the bistable is performing as required. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within the ALLOWABLE VALUE, is acceptable provided that operation and testing is consistent with the assumptions of the plant-specific setpoint calculation. Each ALLOWABLE VALUE specified is more conservative than the analytical limit assumed in the transient and accident analysis in order to account for instrument uncertainties appropriate to

(continued)

(continued)

BASES (continued)

LCO (continued)

the trip function. These uncertainties are defined in the plant-specific setpoint methodology (Ref. 6).

[At this facility, relay configuration is as follows:]
[At this facility, the trip meets single-failure criterion for single-phasing events as follows:]

[At this facility, the time-delay setpoint is controlled as follows:]

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

2. Fuel Building Air Cleanup Actuation System

One channel per train (two trains) of the Fuel Building Air Cleanup Actuation System must be OPERABLE in MODES 1, 2, 3, and 4 and during movement of irradiated fuel within the fuel building. This ensures no single failure will disable both trains.

No ALLOWABLE VALUE is associated with the automatic actuation logic or manual initiation functions. Automatic initiation on high radiation is covered by LCO 3.3.7.

[For this facility, the following support systems are required to be OPERABLE to ensure EDG LOPS and Fuel Building Air Cleanup Actuation System OPERABILITY:]

[For this facility, those required support systems which, upon their failure, do not require declaring the EDG LOPS and Fuel Building Air Cleanup Actuation System inoperable and their justification are as follows:]

[For this facility, the supported systems impacted by the inoperability of the EDG LOPS and Fuel Building Air Cleanup Actuation System and the justification of whether or not each supported system is declared inoperable are as follows:]

(continued)

BASES (continued)

APPLICABILITY

The EDG LOPS actuation function is required in MODES 1, 2, 3, and 4 because ESF functions are designed to provide protection in these MODES. Actuation in MODES 5 or 6 is required whenever the required EDG must be OPERABLE, so that it can perform its function on a loss of power or degraded power to the vital bus.

The manual and automatic Fuel Building Air Cleanup Actuation System trains MUST be OPERABLE in MODES 1, 2, 3, and 4 and when moving irradiated fuel in the fuel building to ensure the Fuel Building Air Cleanup Actuation System operates to remove fission products associated with leakage after a LOCA or a fuel-handling accident.

A Note has been added to provide clarification that for this LCO, each function in Table 3.3.6-1 is treated as an independent entity 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 in the LCO section of the Bases. The most frequent occurrence to render a protection function inoperable is the determination that a bistable or process module has drifted sufficiently to exceed the ALLOWABLE VALUE. Typically, the drift is small and results in a delay of actuation rather than a total loss of function. This determination is generally made during the performance of an ANALOG CHANNEL OPERATIONAL TEST, when the process instrumentation is set up for adjustment to bring it within specification. If the trip setpoint is less conservative than the ALLOWABLE VALUE, the channel must be declared inoperable immediately and the appropriate Condition from Table 3.3.6-1 must be entered.

In the event a channel's trip setpoint is found non-conservative with respect to the ALLOWABLE VALUE, or the transmitter, a rack module, or an [SSPS] module is found inoperable, then the function that channel provides must be declared inoperable and the LCO Condition entered for the particular protection function affected.

(continued)

(continued)

BASES (continued)

ACTIONS (continued)

When the required channels are specified on a per bus or per train basis, then the Condition may be entered separately for each bus or train as appropriate.

Condition A

Condition A is applicable to all protection functions. Condition A 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 3.3.6-1 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 EDG LOPS function with one loss-of-voltage or degraded voltage channel inoperable.

A Note is added to allow bypassing a channel for up to 4 hours for surveillance testing of other channels. This allowance is made where bypassing the channel does not cause an actuation and where at least two other channels are monitoring that parameter. [For this facility, the 4-hour bypass is justified as follows:]

B.1, B.2.1, and B.2.2

Restoring channel OPERABILITY (Required Action B.1) is the preferred action because it restores full functional capability of the LOPS. The 1-hour Completion Time is reasonable to evaluate and take action to correct a degraded condition in an orderly manner and takes into account the low probability of an event requiring LOPS occurring during this interval.

If the channel cannot be restored to OPERABLE status, in compliance with Required Action B.1, Required Action B.2.1 requires that the channel shall be tripped within 1 hour. Otherwise, the affected EDG must be declared inoperable in accordance with Required Action D.1. The justification for Completion Time is the same as that for Required Action B.1. With a channel in trip, the LOPS logic becomes one-out-of-three.

(continued)

(continued)

BASES (continued)

ACTIONS (continued)

Required Action B.1 requires the single inoperable loss-of-voltage or degraded voltage trip channel in a single bus to be restored to OPERABLE status or Required Action B.2.1 requires the inoperable channel to be placed in trip within 1 hour. With a channel in trip the LOPS channels are configured to provide a one-out-of-three logic to initiate a trip of the incoming offsite power. In trip, one additional valid actuation will cause an LOPS signal on the bus. The 1-hour Completion Time is justified on the same basis as for Required Action B.1.

Required Action B.2.2 provides a Completion Time of prior to the next TRIP ACTUATING DEVICE OPERATIONAL TEST a time period which could be as long as 31 days. Restoring the channel before the next TRIP ACTUATING DEVICE OPERATIONAL TEST should allow ample time to repair most failures with the LOPS which is still capable of performing its design function given an additional failure.

Condition C

Condition C applies when more than one undervoltage or more than one degraded voltage channel in a single bus is inoperable.

Restoring one channel to OPERABLE status (Required Action C.1) is the preferred action. The 1-hour Completion Time should allow ample time to repair most failures and takes into account the low probability of an event requiring LOPS occurring during this interval.

Condition D

Condition D applies to each of the EDG LOPS functions when the Required Actions and associated Completion Times of Conditions B or C are not met.

The affected diesel generator is required to be declared inoperable and the actions specified in LCO 3.8.1, "AC Sources—Operating" or 3.8.2, "AC Sources—Shutdown" are required immediately. Also, other supported systems affected by LOPS channel inoperability are declared inoperable and the corresponding LCOs entered. [For this facility, the supported systems impacted by LOPS channel inoperability are as follows:]

(continued)

(continued)

BASES (continued)

ACTIONS (continued)

Condition E

Condition E is applicable to the manual initiation and automatic actuation logic and actuation relays for the Fuel Building Air Cleanup Actuation System (FBACS). These Required Actions address the train orientation of the actuation system and include master and slave relays as applicable. If one train is inoperable, 7 days are allowed to restore the train to OPERABLE status. The Completion Time of 7 days is consistent with the Completion time specified in LCO 3.7.15, "Fuel Building Air Cleanup System (FBACS)." For one filter train inoperable. If the channel cannot be restored to OPERABLE status, the plant must be placed in MODE 3 within the following 6 hours and MODE 5 within the next 30 hours. Six hours and 30 hours are reasonable, based on operating experience and normal cooldown rates, to reach MODES 3 and 5 from MODE 1 in an orderly manner and without challenging plant systems. The justification for this Completion Time is the same as provided in Bases B 3.7.15, "Fuel Building Air Cleanup System (FBACS)." If the train cannot be restored to OPERABLE status, the plant should be placed in MODE 3 within the following 6 hours and in MODE 5 within the next 30 hours. Six hours and 30 hours are reasonable times, based on operating experience and normal cooldown rates, to reach MODES 3 and 5 from full power in an orderly manner and without challenging plant systems.

Condition F

Condition F represents a loss of the FBACS automatic initiation function in MODE 5 or 6 when moving irradiated fuel or loads over irradiated fuel in the fuel building.

Restoring at least one train to OPERABLE status (Required Action F.1) is the preferred action. If one channel is inoperable, 7 days are permitted to restore it to OPERABLE status or place one OPERABLE FBACS train in operation. If the channel cannot be restored to OPERABLE status and an FBACS train cannot be placed in operation movement of irradiated fuel must be suspended immediately following the 7-day allowed Completion Time. The justification of the 7-day Completion Time is the same as discussed above for Condition F.

(continued)

(continued)

BASES (continued)

ACTIONS (continued)

If two trains become inoperable 1 hour is allowed to restore the train to OPERABLE status or place one FBACS train in service. Required Action F.1 restores system function, but not single-failure tolerance. Completion of Required Action F.1 returns the system to the configuration described by Condition E. Required Action F.2 places the FBACS in the configuration that would be achieved by logic actuation. Therefore, after completion of Required Action F.2, logic OPERABILITY is unnecessary. The Completion Time of 1 hour is based upon the relative improbability of an event requiring actuation during that time. During this interval, the FBACS may still be actuated by manual operation of individual components. One hour is consistent with the time allowed by LCO 3.0.3 for loss of function.

Condition G

Condition G applies to the failure of two channels during CORE ALTERATIONS an when moving irradiated fuel.

If two channels are inoperable, Required Actions are immediately taken to restore one channel to OPERABLE status, or place one OPERABLE FBACS train in operation, or movement of irradiated fuel and movement of loads over irradiated fuel in the fuel building are suspended.

Condition H

Condition H is applicable to each one of the miscellaneous safeguards actuations functions presented in Table 3.3.6-1.

Required Action H.1 verifies that the Required Actions have been initiated for those supported systems declared inoperable because of the inoperability of the support channel(s) or train(s) within a Completion Time of 1 hour. The specified Completion Time is sufficient for plant operations personnel to make this determination.

Required Action H.1 ensures that those identified Required Actions associated with supported systems impacted by the inoperability of channel(s) or trains(s) associated with each miscellaneous safeguards actuations function have been initiated. This can be accomplished by entering the supported systems LCOs or independently as a group of Required Actions needed to be initiated every time

(continued)

(continued)

BASES (continued)

ACTIONS (continued)

Condition H is entered. [For this facility, the identified supported systems Required Actions associated with each miscellaneous safeguard actuations function are as follows:]

Required Action H.2 verifies that all required support or supported features associated with the other redundant train(s) or 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 actuation function are found in the SR column of Table 3.3.6-1 for that function.

SR 3.3.6.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is a comparison of the indicated parameter on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or something even more serious. CHANNEL CHECK will detect gross channel failure, thus it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including isolation, indication, and readability. If a channel is outside the match criteria, it may be an indication that the transmitter or the signal processing equipment has drifted outside its limit. If the channels are within the match criteria, it is an indication that the channels are OPERABLE. If the channels are normally off scale during times when Surveillance is required, the

(continued)

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

CHANNEL CHECK will only verify that they are off scale in the same direction. Off-scale low current-loop channels are verified to be reading at the bottom of the range and not failed downscale.

The Surveillance Frequency, about once every shift, is based on operating experience that demonstrates the rarity of channel failure. Thus, performance of the CHANNEL CHECK guarantees that undetected overt channel failure is limited to 12 hours. Since the probability of two random failures in redundant channels in any 12-hour period is extremely low, the CHANNEL CHECK minimizes the chance of loss of protective function due to failure of redundant channels. The CHANNEL CHECK supplements less formal, but more frequent checks of channel operability during normal operational use of the displays associated with this LCO required channels.

SR 3.3.6.2

SR 3.3.6.2 is the performance of a TRIP ACTUATING DEVICE OPERATIONAL TEST. This test is performed every 31 days. These check trip devices that provide actuation signals directly, bypassing the analog process control equipment. For these tests, the relay trip setpoints are verified and adjusted as necessary. [For this facility, the 31-day Frequency is justified as follows:]

SR 3.3.6.3

SR 3.3.6.3 is the performance of an ACTUATION LOGIC TEST. The actuation logic is tested every 31 days on a STAGGERED TEST BASIS, using the semiautomatic tester if appropriate. The train being tested is placed in the bypass condition, thus preventing inadvertent actuation. All possible logic combinations, with and without applicable permissive, are tested for each protection function. In addition, the master relay coil is pulse tested for continuity. This verifies that the logic modules are OPERABLE and there is a complete voltage signal path to the master relay coils. [For this facility, the Surveillance Frequency is justified as follows:]

(continued)

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.6.4

SR 3.3.6.4 is the performance of a MASTER RELAY TEST. The MASTER RELAY TEST is the energizing of the master relay, verifying contact operation and a low-voltage continuity check of the slave relay coil. Upon master relay contact operation, a low voltage is injected to the slave relay coil. This voltage is insufficient to pick up the slave relay, but large enough to demonstrate signal path continuity. This test is performed every 31 days on a STAGGERED TEST BASIS. [For this facility, the Surveillance Frequency is justified as follows:]

SR 3.3.6.5

SR 3.3.6.5 is the performance of a SLAVE RELAY TEST. The SLAVE RELAY TEST is the energizing of the slave relays. Contact operation is verified in one of two ways. Actuation equipment that may be operated in the design mitigation mode is either allowed to function or is placed in a condition where the relay contact operation can be verified without operation of equipment. Actuation equipment that may not be operated in the design mitigation mode is prevented from operation by the SLAVE RELAY TEST circuit. For this latter case, contact operation is verified by a continuity check of the circuit containing the slave relay. This test is performed every 92 days. [For this facility, the Surveillance Frequency is justified as follows:]

SR 3.3.6.6

CHANNEL CALIBRATION is a complete check of the instrument channel including the detector. The test verifies the channel responds to measured parameter with the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift between successive tests, to ensure that the instrument channel remains operational with the setpoint within the assumptions of the plant-specific setpoint analysis. Detector "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

(continued)

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

determination and readjustment must be performed consistent with the assumptions of the plant-specific setpoint analysis.

Recalibration restores OPERABILITY of an otherwise functional component found to have errors larger than assumed by the setpoint analysis. However, repeated failures of the same channel over a relatively small number of test intervals must be considered as potentially indicating a deterministic failure which cannot be corrected by recalibration.

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.

The setpoints to a loss-of-voltage and a degraded-voltage condition shall be tested. The degraded voltage test shall include a single-point verification that the trip occurs within the required delay time, as shown in Figure 3.3.3-1. [For this facility, the Surveillance Frequency is justified as follows:]

SR 3.3.6.7

SR 3.3.6.7 is the performance of a TRIP ACTUATING DEVICE OPERATIONAL TEST. This test is a check of the manual actuation functions and is performed every 18 months. Each manual actuation function is tested up to, and including, the master relay coils. In some instances, the test includes actuation of the end device (e.g., pump starts, valve cycles). [For this facility, the Surveillance Frequency is justified as follows:]

REFERENCES

1. [Unit Name] FSAR, Section [8.3], "[Title]."

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

3. Title 10, Code of Federal Regulations, Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants."

(continued)

(continued)

BASES (continued)

REFERENCES (continued)

4. Institute of Electrical and Electronic Engineers, IEEE-279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations," April 5, 1972

5. [Unit Name] FSAR, Section [7], "[Instrumentation and Controls]."

6. "[Plant-Specific RPS/ESFAS Setpoint Methodology Study.]"

B 3.3 INSTRUMENTATION

B 3.3.7 Radiation Monitoring Instrumentation

BASES

BACKGROUND

The radiation monitoring instrumentation serves to detect high radiation conditions and, via LCO 3.3.2, "Engine ed Safety Feature Actuation System (ESFAS)" and LCO 3.3.6, "Miscellaneous Safeguards Actuations" logics initiates action to mitigate high radiation conditions. Radiation monitoring instrumentation covers three radiation monitoring functional units: containment purge isolation, control room emergency ventilation, and fuel building air cleanup.

1. Containment Purge Isolation--Radiation High

Four radiation monitoring channels are provided as input to the ESFAS—Containment Purge Isolation function (LCO 3.3.2, "Engineered Safety Feature Actuation System (ESFAS)," Function 4). The four channels mea ure containment radiation at two locations. One channel is a containment area gamma monitor, and the other three measure radiation in a sample of the containment purge exhaust. The three purge exhaust radiation detectors are of three different types: gaseous, particulate, and iodine monitors. Since the purge exhaust monitors constitute a sampling system various components such as sample line valves, sample line heaters, sample pumps, and filter motors are required to support monitor OPERABILITY. Because the four monitors provide different radiation measurements, they are not considered redundant to each other.

A high-radiation signal from any one of the four channels initiates containment purge isolation, which closes containment isolation valves in the Mini-Purge System and the Shutdown Purge System. The Mini-Purge System may be in use during reactor operation and the Shutdown Purge System will be in use with the reactor shutdown. Each of the purge systems has inner and outer containment isolation valves in its supply and exhaust ducts. These systems are described in the Bases for LCO 3.6.6, "Containment Isolation Valves." The Containment Purge Isolation Radiation monitoring

(continued)

(continued)

BASES (continued)

BACKGROUND (continued)

instrumentation isolates the containment atmosphere from the environment to minimize releases of radioactivity in the event of an accident. Containment Purge Isolation also initiates on an automatic Safety Injection (SI) function through the Containment Phase A Isolation function or by manual actuation of Phase A Isolation. The Bases for LCO 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," discuss these other modes of initiation.

2. Control Room Emergency Ventilation--Radiation High

During normal operation, the Auxiliary Building Ventilation System provides control room ventilation. The radiation monitoring instrumentation required by LCO 3.3.7 redundantly monitors levels of radiation in the control room air and the air entering the control room from the unfiltered outside air intake. A high-radiation signal from either detector actuates the ESFAS control room emergency ventilation logic (LCO 3.3.2, "Engineered Safety Feature Actuation System Instrumentation," Function 9) to initiate isolation of the control room and start both trains of the Control Room Emergency Filtration System. This system is described in the Bases for LCO 3.7.12, "Control Room Emergency Filtration System (CREFS)."

Control room emergency ventilation may also be actuated manually and by containment Phase A isolation. LCO 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," identifies the requirements for these other initiation signals.

3. Fuel Building Air Cleanup System--Radiation High

High gaseous and particulate radiation monitored by either of two monitors provides Fuel Building Air Cleanup System (FBACS) initiation. Each FBACS train is initiated by high radiation detected by a channel dedicated to that train. There are a total of two channels, one for each train. Each channel contains a gaseous and particulate monitor. The monitors activate the FBACS logic required by LCO 3.3.6, "Miscellaneous Safeguards Actuation." High radiation

(continued)

(continued)

BASES (continued)

BACKGROUND (continued)

detected by any monitor [or an SI signal from the ESFAS] initiates fuel building isolation and starts the FBACS. These actions function to prevent exfiltration of contaminated air by initiating filtered ventilation, which imposes a negative pressure on the fuel building. The Bases for LCO 3.7.15, "Fuel Building Air Cleanup System (FBACS)," describe the FBACS. Since the radiation monitors constitute a sampling system, various components such as sample line valves, sample line heaters, sample pumps, and filter motors are required to support monitor OPERABILITY.

The trip setpoints used in the bistables are based 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, and instrument drift, ALLOWABLE VALUES specified in Table 3.3.1 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 Protection System Selection of Trip Setpoint Values (Ref. 4). The nominal trip setpoint entered into the bistable is normally still more conservative than that specified by the ALLOWABLE VALUE, to account for changes in random measurement errors detectable by a CHANNEL FUNCTIONAL TEST. One example of such a change in measurement error is drift during the Surveillance Frequency. If the measured setpoint does not exceed the ALLOWABLE VALUE, the bistable is considered OPERABLE.

Setpoints in accordance with the ALLOWABLE VALUE will assure that the consequences of Design Basis Accidents (DBA) will be acceptable, providing the plant is operated from within the LCOs at the onset of the anticipated operational occurrence (AOO) or DBA, and the equipment functions as designed.

Note that in LCO 3.3.7 the ALLOWABLE VALUES of Table 3.3.7-1 are the LSSS. These ALLOWABLE VALUES are established to prevent violation of the Safety Limits during normal plant operation and AOOs.

(continued)

(continued)

BASES (continued)

BACKGROUND (continued)

The ALLOWABLE VALUES listed in Table 3.3.7-1 are based on the methodology described in Reference 2, which incorporates all of the known uncertainties applicable for each channel. The magnitudes of these uncertainties are factored into the determination of each trip setpoint. All field sensors and signal processing equipment for these channels are assumed to operate within the allowances of these uncertainty magnitudes.

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

APPLICABLE SAFETY ANALYSES

1. Containment Purge Isolation--Radiation High

The safety analyses assume that the containment remains intact with penetrations unnecessary for core cooling isolated early in the event, approximately 60 seconds. The isolation of the purge valves has not been analyzed mechanistically in the dose calculations, although its rapid isolation is assumed. The containment purge isolation radiation monitors act as backup to the SI signal to ensure closing of the purge and exhaust valves. They are also the primary means for automatically isolating containment in the event of a fuel-handling accident during shutdown. Containment isolation in turn ensures meeting the containment leakage-rate assumptions of the safety analyses, and that the calculated accidental offsite radiological doses are below 10 CFR 100 (Ref. 1) limits. OPERABILITY of the containment purge isolation radiation monitors and the purge isolation logic is necessary to comply with 10 CFR 50, Appendix A, GDC 54 (Ref. 2).

2. Control Room Emergency Ventilation--Radiation High

The Control Room Emergency Ventilation--Radiation High monitor acts to terminate the supply of unfiltered

(continued)

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

outside air to the control room and to actuate the Emergency Filtration System. These actions are necessary to ensure the control room is kept habitable for the operators stationed there during accident recovery and post-accident operations. Operation of these monitors is necessary to minimize radiation exposure of control room personnel and to ensure any exposure throughout the duration of any of the postulated accidents does not exceeds the limits set by 10 CFR 50, Appendix A, GDC 19 (Ref. 2).

3. Fuel Building Air Cleanup System--Radiation High

The FBACS ensures that radioactive materials in the fuel building atmosphere following a fuel-handling accident or a loss-of-coolant accident (LOCA) are filtered and adsorbed prior to being exhausted to the environment. This action reduces the radioactive content in the fuel building exhaust following a LOCA or fuel-handling accident so that offsite doses remain within the limits specified in 10 CFR 100 (Ref. 1).

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

LCO

The LCO requirements ensure that the radiation monitoring instrumentation necessary to initiate Containment Purge Isolation, Control Room Emergency Ventilation, and Fuel Building Air Cleanup remain OPERABLE.

The radiation monitoring instrumentation is OPERABLE when:

- All channel components necessary to provide an initiation signal on a high-radiation signal are functional and in service. For sampling systems, OPERABILITY requires correct valve lineups, sample pump operation, and filter motor operation as well as detector OPERABILITY if these supporting features are necessary for trip to occur under the conditions assumed by the safety analyses or setpoint analysis;

(continued)

(continued)

BASES (continued)

LCO (continued)

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

- Required surveillance testing is current and has demonstrated performance within the surveillance test acceptance criteria.

Only the ALLOWABLE VALUES are specified for each trip function in the LCO. Nominal trip setpoints are specified in the plant-specific setpoint calculations. The nominal setpoints are selected to ensure the setpoint measured by the ANALOG CHANNEL OPERATIONAL 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 are 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 and accounted for in the plant-specific setpoint analysis and the offsite dose calculation manual.

1. Containment Purge Isolation--Radiation High

   [At this facility, one operable channel for each function is sufficient for the following reasons:]

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

2. Control Room Emergency Ventilation--Radiation High

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

3. Fuel Building Air Cleanup System--Radiation High

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

(continued)

(continued)

BASES (continued)

LCO (continued)

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

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

[For this facility, the supported systems impacted by the inoperability of the radiation monitoring instrumentation and the justification of whether or not each supported system is declared inoperable are as follows:]

APPLICABILITY

1. Containment Purge Isolation--Radiation High

The purge isolation on high radiation is required to be OPERABLE whenever the reactor building purge valves are open in MODES 1, 2, 3, and 4. Outside of these MODES, the purge isolation must be OPERABLE whenever CORE ALTERATIONS or movement of irradiated fuel is taking place. These latter conditions are those under which the potential for fuel damage, and thus radiation release, is the greatest. While in MODES 5 and 6 without fuel handling in progress, the Purge Valve Isolation System need not be OPERABLE since the potential for radioactive releases is minimized and operator action is sufficient to ensure post-accident offsite doses are maintained within the limits of 10 CFR 100. In MODES 5 and 6 the purge valves are needed in preparation for entry. This capability is required to minimize doses for personnel entering the building.

2. Control Room Emergency Ventilation--Radiation High

The Control Room Emergency Ventilation--Radiation High is required to be OPERABLE in MODES 1, 2, 3, and 4 to protect operators from potential fission-product release resulting from a LOCA or steam generator tube rupture. The high-radiation function must also be OPERABLE in any MODE during CORE ALTERATIONS, movement

(continued)

(continued)

BASES (continued)

APPLICABILITY (continued)

of loads over fuel, or movement of fuel to ensure automatic initiation of the CREFS is OPERABLE when the potential for a fuel-handling accident exists. These are the conditions under which the potential for fuel damage, and thus radiation release, is the greatest and for which there may be a need to isolate the control room to ensure a habitable environment for operators. While in MODES 5 and 6 without fuel handling in progress, the high radiation instrumentation need not be OPERABLE since the potential for radioactive releases is minimized and manual operator action is sufficient to prevent doses from exceeding the limits of 10 CFR 50, Appendix A, GDC 19.

3. Fuel Building Air Cleanup System--Radiation High

High-radiation initiation of the FBACS is required in MODES 1, 2, 3, and 4 to remove fission products caused by post-LOCA Emergency Core Cooling System leakage in the fuel building. The high-radiation function must also be OPERABLE in any MODE during movement of loads over fuel or during movement of fuel in the fuel building to ensure automatic initiation of the FBACS when the potential for a fuel-handling accident exists. While in MODES 5 and 6 without fuel handling in progress, the high-radiation instrumentation need not be OPERABLE since the potential for radioactive releases is minimized and manual operator action is sufficient to prevent offsite dose limits from being exceeded.

A Note has been added to provide clarification that for this LCO, each function specified in Table 3.3.7-1 is treated as an independent entity 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

(continued)

(continued)

BASES (continued)

ACTIONS (continued)

drift is found to be small and results in a delay of actuation rather than a total loss of function. This determination is generally made during the performance of an ANALOG CHANNEL OPERATIONAL TEST, when the process instrumentation is set up for adjustment to bring it within specification. If the trip setpoint is less conservative than the ALLOWABLE VALUE, the channel must be declared inoperable immediately and the appropriate Condition from Table 3.3.7-1 must be entered. In the event that a channel trip setpoint is found non-conservative with respect to the ALLOWABLE VALUE, or the required sampling equipment is found inoperable, then all affected functions provided by that channel must be declared inoperable and the LCO Condition entered for the particular protective function affected.

When the number of inoperable channels in a trip function exceeds those specified in one or other related Conditions associated with the same trip function, then the plant is outside of the safety analysis. Therefore, LCO 3.0.3 is immediately entered, if applicable.

Condition A

Condition A is applicable to all radiation monitoring instrumentation protection functions. Condition A 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 3.3.7-1 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 is applicable to the containment purge isolation radiation monitor channels in MODES 1, 2, 3, and 4. Since the four containment radiation monitors measure different parameters, failure of a single channel may result in loss of the radiation monitoring function for certain events.

B.1, B.2, B.3.1, and B.3.2

Required Action B.1, restoring channel to OPERABLE status, is the preferred action as it completely restores the Containment Purge Isolation--Radiation High function.

(continued)

(continued)

BASES (continued)

ACTIONS (continued)

If the inoperable channel cannot be restored, operation may continue as long as the Containment Purge System supply and exhaust valves are maintained in the closed position as required by Required Action B.2. This action accomplishes the safety function of the Containment Purge Isolation--Radiation High function. The 1-hour Completion Time is reasonable considering the time required to isolate the penetration and recognizes the fact that Condition B addresses, in the limit, a simultaneous loss of all four radiation monitoring channels. In MODES 1, 2, and 3 automatic purge isolation is also provided by a safety injection signal required by LCO 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation."

Required Action B.3.1 and Required Action B.3.2 provide an alternative in the event that neither Required Action B.1 nor B.2 can be met within the associated Completion Times. In this case, the plant must be placed in a MODE in which the LCO does not apply. This is done by placing the plant in at least MODE 3 within the following 6 hours and in MODE 5 within the following 36 hours. The times allowed to reach MODES 3 and 5 from MODE 1 are reasonable, based on operating experience, to reach the required MODES from full power in an orderly manner and without challenging plant systems.

Condition C

Condition C is applicable to the containment purge isolation radiation monitor channels during movement of irradiated fuel within containment or during CORE ALTERATIONS. Since the four containment radiation monitors measure different parameters, failure of a single channel may result in loss of the radiation monitoring function for certain events.

C.1, C.2, C.3.1, and C.3.2

Required Action C.1, restoring channel to OPERABLE status, is the preferred action as it completely restores the Containment Purge Isolation--Radiation High function. If the inoperable channel cannot be restored, operation maycontinue as long as the Containment Purge System supply and exhaust valves are maintained in the closed position as required by Required Action C.2. This action accomplishes

(continued)

(continued)

BASES (continued)

ACTIONS (continued)

the safety function of the Containment Purge Isolation--Radiation High function.

Required Action C.3.1 and Required Action C.3.2 provide an alternative in the event that neither Required Action C.1 nor Required Action C.2 can be met within the associated Completion Times. In this case, CORE ALTERATIONS and movement of fuel assemblies within Containment are suspended.

The requirement to complete these actions immediately recognizes the fact that the high-radiation signal is the only function that automatically isolates containment in response to radiation release in the event of a fuel-handling accident. Furthermore, Condition C represents a range of Conditions from failure of the monitor for a single parameter to failure of all four radiation monitoring channels.

Condition D, Condition E, and Condition F

Condition D, Condition E, and Condition F are applicable to the CREVS radiation monitor channels.

Condition D applies to the failure of a single channel in MODE 1, 2, 3, or 4. If one channel is inoperable, 7 days is allowed to restore it to OPERABLE status. The 7-day Completion Time is the same that allowed if one channel of the mechanical portion of the system is inoperable. The basis for this time is the same as provided in LCO 3.7.12, "Control Room Emergency Filtration System (CREFS)." If the channel cannot be restored to OPERABLE status, the plant must be placed in MODE 3 within the following 6 hours and in MODE 5 within the next 30 hours. The times allowed to reach MODES 3 and 5 from MODE 1 are reasonable, based on operating experience, to reach the required MODES from full power in an orderly manner and without challenging plant systems.

Condition E applies to the failure of one channel during CORE ALTERATIONS and when moving irradiated fuel. If one channel is inoperable, 7 days is allowed to restore it to OPERABLE status or place one OPERABLE Control Room Emergency Filtration System (CREFS) train in emergency filtration mode. If neither the radiation monitoring instrumentation channel nor the CREFS train can be restored to OPERABLE

(continued)

(continued)

BASES (continued)

ACTIONS (continued)

status or placed in operation respectively, CORE ALTERATIONS, positive reactivity additions, and movement of irradiated fuel must be suspended immediately following the 7-day allowed Completion Time. The justification of the 7-day Completion Time is the same as discussed above for Condition D.

Note that in certain circumstances, such as fuel handling in the fuel building during power operation, both Condition D and Condition E may be applicable in the event of a single channel failure.

Condition F applies to the failure of two channels during CORE ALTERATIONS and when moving irradiated fuel. If two channels are inoperable, Required Actions are immediately taken to restore one radiation monitoring instrumentation channel to OPERABLE status, or place one OPERABLE CREFS channel in the emergency filtration mode, or immediately suspend CORE ALTERATIONS, positive reactivity additions, and movement of irradiated fuel.

Condition E and Condition F have been modified by a Note, which specifies that the CREFS be placed manually in the emergency filtration mode, if the auto-swapover emergency filtration is inoperable.

Condition G, Condition H, and Condition I

Condition G, Condition H, and Condition I are applicable to the radiation monitoring instrumentation for the FBACS.

Condition G applies to the failure of a single channel in MODE 1, 2, 3, or 4. If one channel is inoperable, 7 days are permitted to restore it to OPERABLE status. The 7-day Completion Time is the same as that allowed if one channel of the mechanical portion of the system is inoperable. The justification for this Completion Time is the same as provided in Bases B 3.7.15, "Fuel Building Air Cleanup System (FBACS)." If the channel cannot be restored to OPERABLE status, the plant must be placed in MODE 3 within the following 6 hours and in MODE 5 within the next 30 hours.

(continued)

(continued)

BASES (continued)

ACTIONS (continued)

The times allowed are reasonable, based on operating experience, to reach MODES 3 and 5, from full power in an orderly manner and without challenging plant systems

Condition H applies to the failure of one channel during movement of irradiated fuel or loads over irradiated fuel in the fuel building. If one channel is inoperable, 7 days are permitted to restore it to OPERABLE status, or place one OPERABLE FBACS train in operation, or suspend movement of irradiated fuel or loads over irradiated fuel in the fuel building. The justification of the 7-day Completion Time is the same as that provided above for Condition G.

Note that in certain circumstances, such as fuel handling in the fuel building during power operation, both Condition G and Condition H may be applicable in the event of a single channel failure.

Condition I applies to the failure of two channels during movement of irradiated fuel or loads over irradiated fuel in the fuel building. The Required Actions are to restore immediately one channel to OPERABLE status, or suspend immediately movement of irradiated fuel or loads over irradiated fuel in the fuel building.

Condition J

Condition J is applicable to each one of the radiation monitoring functions presented in Table 3.3.7-1.

J.1 and J.2

Required Action J.1 verifies that the Required Actions have been initiated for those supported systems declared inoperable because of the inoperability of the support channel(s) within a Completion Time of 1 hour. The specified Completion Time is sufficient for plant operations personnel to make this determination.

Required Action J.1 ensures that those identified Required Actions associated with supported systems impacted by the inoperability of channel(s) associated with each radiation monitoring instrumentation function have been initiated. This can be accomplished by entering the supported systems'

(continued)

(continued)

BASES (continued)

ACTIONS (continued)

LCOs independently or as a group of Required Actions that need to be initiated every time Condition J is entered.

[For this facility, the identified supported systems Required Actions associated with each radiation monitoring function are as follows:]

Required Action J.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 radiation monitoring function are found in the SR column of Table 3.3.7-1 for that function.

SR 3.3.7.1

Performance of the CHANNEL CHECK for the radiation monitoring instrumentation once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. Performance of the CHANNEL CHECK helps to ensure that the instrumentation continues to operate properly between each CHANNEL CALIBRATION. The high-radiation instrumentation should be compared to similar plant instruments located throughout the plant. If the radiation monitor 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.

(continued)

(continued)

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. The Surveillance Frequency, about once every shift, is based on operating experience that demonstrates the rarity of channel failure. Thus, SR 3.3.7.1 ensures that loss of function will be identified within 12 hours. [At this facility, administrative controls and design features (e.g., down-scale alarms) that immediately alert operators to loss of the Containment Purge Isolation--Radiation High function are as follows:] [For this facility the CHANNEL CHECK verification of sample system alignment and operation for gaseous, particulate, and iodine monitors constitutes the following:]

SR 3.3.7.2

An ANALOG CHANNEL OPERATIONAL TEST helps ensure that the channels can perform their intended functions and is required to be performed once every 31 days. This test verifies the capability of the instrumentation to provide the reactor building isolation. [For this facility, the ANALOG CHANNEL OPERATIONAL TEST consists of the following:] [For this facility the 31-day Frequency is justified as follows:]

SR 3.3.7.3

Performance of a CHANNEL CALIBRATION every 18 months ensures that the instrument channel remains operational with the correct setpoint. The calibration is a complete check of the instrumentation and detector.

This test is a complete check of the process control instrument loop and the transmitter. Transmitter "as found" and "as left" values are recorded and used to verify drift assumptions. The radiation monitor may be calibrated in place or on a bench using test equipment, or it may be replaced by an equivalent, laboratory-calibrated unit. For OPERABLE channels, CHANNEL CALIBRATION shall find that measurement errors and bistable setpoint errors are within the assumptions of the plant-specific setpoint analysis.

(continued)

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

Recalibration restores operability of an otherwise functional component that does not meet these criteria. However, repeated failures of the same channel over a relatively small number of test intervals must be considered as potentially indicating a deterministic failure that cannot be corrected by recalibration.

Completion of this test results in the channel being properly adjusted and expected to remain within the "as found" tolerance assumed by the setpoint analysis until the next scheduled surveillance. Measurement and setpoint error determination and readjustment must be performed consistent with the assumptions of the plant-specific setpoint analysis.

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.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 100.11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance," 1973.
2. Title 10, Code of Federal Regulations, Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants."
3. [Unit Name] FSAR, Section [], "[Accident Analysis]."
4. [Unit Name] "Plant Protection System Selection of Trip Setpoint Values."

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)

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)

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	diocytl 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

(continued)

APPENDIX A (continued)

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 power tilt
QPTR	QUADRANT POWER TILT RATIO
QS	quench spray
RACS	Rod Action Control System
RAOC	relaxed axial offset control
RAS	recirculation actuation signal
RB	reactor building
RBM	rod block monitor
RCCA	rod cluster control assembly
RCIC	reactor core isolation cooling
RCIS	Rod Control and Information System
RCP	reactor coolant pump
RCPB	reactor coolant pressure boundary
RCS	Reactor Coolant System
REA	rod ejection accident
RHR	residual heat removal
RHRSW	residual heat removal service water
RMCS	Reactor Manual Control System
RPB	reactor pressure boundaries
RPC	rod pattern controller
RPCB	reactor power cutback

(continued)

RPIS	Rod Position Information System
RPS	Reactor Protection System
RPV	reactor pressure vessel
RS	recirculation spray
RT	reference temperature
RT_{NDT}	nil-ductility reference temperature
RTCB	reactor trip circuit breaker
RTD	resistance temperature detector
RTM	reactor trip module
RTP	RATED THERMAL POWER
RTS	Reactor Trip System
RWCU	reactor water cleanup
RWE	rod withdrawal error
RWL	rod withdrawal limiter
RWM	rod worth minimizer
RWP	Radiation Work Permit
RWST	refueling water storage tank
RWT	refueling water tank
SAFDL	specified acceptable fuel design limits
SBCS	Steam Bypass Control System
SBO	station blackout
SBVS	Shield Building Ventilation System
SCAT	spray chemical addition tank
SCI	secondary containment isolation
SCR	silicon controlled rectifier
SDV	scram discharge volume
SDM	SHUTDOWN MARGIN
SER	Safety Evaluation Report
SFRCS	Steam and Feedwater Rupture Control System
SG	steam generator
SGTR	steam generator tube rupture
SGTS	Standby Gas Treatment System
SI	safety injection
SIAS	safety injection actuation signal
SIS	safety injection signal
SIT	safety injection tank
SJAE	steam jet air ejector
SL	Safety Limit
SLB	steam line break
SLC	standby liquid control
SLCS	Standby Liquid Control System
SPMS	Suppression Pool Makeup System
SRM	source range monitor

(continued)

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

NRC FORM 335 (2-89) NRCM 1102, 3201, 3202

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

(See instructions on the reverse)

1. REPORT NUMBER (Assigned by NRC. Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG-1431
Vol. 2

2. TITLE AND SUBTITLE

Standard Technical Specifications
Westinghouse Plants
Bases (Sections 2.0-3.3)

Draft Report for Comment

3. DATE REPORT PUBLISHED

MONTH	YEAR
January	1991

4. FIN OR GRANT NUMBER

5. AUTHOR(S)

6. TYPE OF REPORT

DRAFT

7. PERIOD COVERED (Inclusive Dates)

8. PERFORMING ORGANIZATION – NAME AND ADDRESS (If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)

Division of Operational Events Assessment
Office of Nuclear Reactor Regulation
U. S. Nuclear Regualtory Commission
Washington, D.C. 20555

9. SPONSORING ORGANIZATION – NAME AND ADDRESS (If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)

Same as above

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 Westinghouse Owners Group. The new STS were developed based on the criteria in the interim Commission Policy Statement on Technical Specification Improvements for Nuclear Power Reactors, dated February 6 1987. The new STS will be used as bases for individual nuclear power plant c..ners to develop improved plant-specific technical specifications. The NRC staff is issuing this draft new STS f.. a 30 working-day comment period. Following the comment period, the NRC s..f will analyze comments received, finalize the new STS, and issue them f.. plant-specific implementation. This report contains three volumes. Volume 1 contains the Specifications for all sections of the new STS. Volume 2 contains the Bases for Sections 2.0 - 3.3 of the new STS and Volume 3 contains the Bases for Sections 3.4 - 3.9 of the new STS.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)

Technical Specifications
Westinghouse
PWR

13. AVAILABILITY STATEMENT

Unlimited

14. SECURITY CLASSIFICATION

(This Page)

Unclassified

(This Report)

Unclassified

15. NUMBER OF PAGES

16. PRICE

NRC FORM 335 (2-89)