



*Pacific Gas and  
Electric Company*

**Lawrence F. Womack**  
Vice President  
Nuclear Services

Diablo Canyon Power Plant  
P.O. Box 56  
Avila Beach, CA 93424

805.545.4600  
Fax: 805.545.4234

November 28, 2001

PG&E Letter DCL-01-121

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

Docket No. 50-275, OL-DPR-80  
Docket No. 50-323, OL-DPR-82  
Diablo Canyon Units 1 and 2  
Changes to Technical Specification Bases

Dear Commissioners and Staff:

Diablo Canyon Power Plant (DCPP) Unit 1 and Unit 2 Technical Specification (TS) 5.5.14, "Technical Specifications (TS) Bases Control Program," provides a means for processing changes to the Bases of the TS. TS 5.5.14b states that licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following: a change in the TS incorporated in the license; or a change to the updated Final Safety Analysis Report or Bases that requires NRC approval pursuant to 10 CFR 50.59. Further, TS section 5.5.14 requires that Bases changes made without prior NRC approval be provided to the NRC on a frequency consistent with 10 CFR 50.71(e). In accordance with this requirement, PG&E is submitting Revision 1 of the DCPP TS Bases. Revision 1 includes changes made to the TS Bases since implementation of the improved TS on June 27, 2000, through October 31, 2001, and replaces Revision 0 in its entirety.

If you have additional questions regarding this revision to the DCPP TS Bases, please contact Mr. Patrick Nugent at (805) 545-4720.

Sincerely,

Lawrence F. Womack

A001



Document Control Desk  
November 28, 2001  
Page 2

PG&E Letter DCL-01-121

cc: Edgar Bailey, DHS w/o enclosure  
Ellis W. Merschoff w/o enclosure  
David L. Proulx w/o enclosure  
Girija S. Shukla w/o enclosure  
Diablo Distribution w/o enclosure

Enclosure

Diablo Canyon Power Plant  
Technical Specification Bases  
Revision 1

(This revision replaces Revision 0 in its entirety)

## B 2.0 SAFETY LIMITS (SLs)

### B 2.1.1 Reactor Core SLs

#### BASES

---

##### BACKGROUND

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

The restrictions of this SL prevent overheating of the fuel 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 fuel centerline melting occurs. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime, where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Fuel centerline melting occurs when the local 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.

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

---

(continued)

BASES (continued)

---

APPLICABLE  
SAFETY  
ANALYSES

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

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

The Reactor Trip System setpoints (Ref. 2), in combination with 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.

Protection for these reactor core SLs is provided by the steam generator safety valves and the following automatic reactor trip functions:

- a. High pressurizer pressure trip;
- b. Low pressurizer pressure trip;
- c. Overtemperature  $\Delta T$  trip;
- d. Overpower  $\Delta T$  trip;
- e. Power Range Neutron Flux trip

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  $\Delta T$  measured by instrumentation, used in the 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 Departure from Nucleate Boiling (DNB) Limits," or the assumed initial conditions of the safety analyses (as indicated in the FSAR, Ref. 5) provide more restrictive limits to ensure that the SLs are not exceeded.

---

(continued)

BASES (continued)

---

**SAFETY LIMITS**      The safety limit is based upon the loci of points of THERMAL POWER, RCS pressure, and average temperature below which the calculated DNBR is not less than the design DNBR value, that fuel centerline temperature remains below melting, that the average enthalpy in the hot leg is less than or equal to the enthalpy of saturated liquid, or that the exit quality is within the limits defined by the DNBR correlation.

The curves are based on enthalpy rise hot channel factor limits provided in the COLR.

The SL is higher than the limit calculated when the AFD is within the limits of the  $F_1(\Delta I)$  function of the overtemperature  $\Delta T$  reactor trip. When the AFD is not within the tolerance, the AFD effect on the overtemperature  $\Delta T$  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 steam generator safety valves or automatic protection actions serve to prevent RCS heatup to the reactor core SL conditions or to initiate a reactor trip function, which forces the unit into MODE 3. Setpoints for the reactor trip functions are specified in LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation." In MODES 3, 4, 5, and 6, Applicability is not required since the reactor is not generating significant THERMAL POWER.

---

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

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

---

- REFERENCES**
1. 10 CFR 50, Appendix A, GDC 10 (associated with 1967 GDC 6 per FSAR Appendix 3.1A).
  2. FSAR, Chapter 7.2.
  3. WCAP-8746-A, March 1977.
  4. WCAP-9273-NP-A, July 1985.
  5. FSAR, Chapter 15.
-

## B 2.0 SAFETY LIMITS (SLs)

### B 2.1.2 Reactor Coolant System (RCS) Pressure 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 to be exceeded during normal operation and anticipated operational occurrences (AOOs). Also, in accordance with GDC 28, "Reactivity Limits" (Ref. 1), reactivity accidents, including rod ejection, do not result in damage to the RCPB greater than limited local yielding.

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

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

---

(continued)

BASES (continued)

---

APPLICABLE  
SAFETY  
ANALYSES

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

The RCS pressurizer safety valves are sized to prevent system pressure from exceeding the design pressure by more than 10%, as specified in Section III of the ASME Code for Nuclear Power Plant Components (Ref. 2). The transient that establishes the required relief capacity, and hence 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 side are assumed to open when the steam pressure reaches the safety valve settings. Main feedwater supply is lost at the time of turbine trip.

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

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

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

---

(continued)

BASES (continued)

---

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

---

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

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

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

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

---

- REFERENCES
1. 10 CFR 50, Appendix A, GDC 14 (associated with 1967 GDC 9 per FSAR Appendix 3.1A), GDC 15 (no direct correlation to 1967 GDC; however, intent of 1971 GDC is per met per FSAR Appendix 3.1A), and GDC 28 (associated with 1967 GDC 30 per FSAR Appendix 3.1A).
  2. ASME, Boiler and Pressure Vessel Code, Section III, Summer 1969.
  3. ASME, Boiler and Pressure Vessel Code, Section XI, 1977 Edition and Summer 1978 Addenda.
  4. 10 CFR 100.
  5. FSAR, Section 7.2.
  6. DCM S-7, 3.4.1.
-

B 3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

BASES

LCOs	LCO 3.0.1 through LCO 3.0.6 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.
LCO 3.0.1	LCO 3.0.1 establishes the Applicability statement within each individual Specification as the requirement for when the LCO is required to be met (i.e., when the unit is in the MODES or other specified conditions of the Applicability statement of each Specification).
LCO 3.0.2	<p>LCO 3.0.2 establishes that upon discovery of a failure to meet an LCO, the associated ACTIONS shall be met. The Completion Time of each Required Action for an ACTIONS Condition is applicable from the point in time that an ACTIONS Condition is entered. The Required Actions establish those remedial measures that must be taken within specified Completion Times when the requirements of an LCO are not met. This Specification establishes that:</p> <ol style="list-style-type: none"> <li>a. Completion of the Required Actions within the specified Completion Times constitutes compliance with a Specification; and</li> <li>b. Completion of the Required Actions is not required when an LCO is met within the specified Completion Time, unless otherwise specified.</li> </ol> <p>There are two basic types of Required Actions. The first type of Required Action specifies a time limit in which the LCO must be met. This time limit is the Completion Time to restore an inoperable system or component to OPERABLE status or to restore variables to within specified limits. If this type of Required Action is not completed within the specified Completion Time, a shutdown may be required to place the unit in a MODE or condition in which the Specification is not applicable. (Whether stated as a Required Action or not, correction of the entered Condition is an action that may always be considered upon entering ACTIONS.) The second type of Required Action specifies the remedial measures that permit continued operation of the unit that is not restricted by the Completion Time. In this case, compliance with the Required Actions provides an acceptable level of safety for continued operation.</p> <p>Completing the Required Actions is not required when an LCO is met or is no longer applicable, unless otherwise stated in the individual Specifications.</p> <p>The nature of some Required Actions of some Conditions necessitates that, once the Condition is entered, the Required Actions must be completed even though the associated Conditions no longer exist. The</p>

(continued)

BASES

---

LCO 3.0.2  
(continued)

individual LCO's ACTIONS specify the Required Actions where this is the case. An example of this is in LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits."

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

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

---

LCO 3.0.3

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

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

This Specification delineates the time limits for placing the unit in a safe MODE or other specified condition when operation cannot be

(continued)

---

BASES

---

LCO 3.0.3  
(continued)

maintained within the limits for safe operation as defined by the LCO and its ACTIONS. It is not intended to be used as an operational convenience that permits routine voluntary removal of redundant systems or components from service in lieu of other alternatives that would not result in redundant systems or components being inoperable.

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

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

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

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

(continued)

BASES

---

LCO 3.0.3  
(continued)

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

Exceptions to LCO 3.0.3 are provided in instances where requiring a unit shutdown, in accordance with LCO 3.0.3, would not provide appropriate remedial measures for the associated condition of the unit. An example of this is in LCO 3.7.15, "Fuel Storage Pool Water Level." LCO 3.7.15 has an Applicability of "During movement of irradiated fuel assemblies in the fuel storage pool." Therefore, this LCO can be applicable in any or all MODES. If the LCO and the Required Actions of LCO 3.7.15 are not met while in MODE 1, 2, 3, or 4, there is no safety benefit to be gained by placing the unit in a shutdown condition. The Required Action of LCO 3.7.15 of "Suspend movement of irradiated fuel assemblies in the fuel storage pool" is the appropriate Required Action to complete in lieu of the ACTIONS of LCO 3.0.3. These exceptions are addressed in the individual Specifications.

---

LCO 3.0.4

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 unit in a MODE or other specified condition stated in that Applicability (e.g., Applicability desired to be entered) when the following exist:

- a. Unit conditions are such that the requirements of the LCO would not be met in the Applicability desired to be entered; and
- b. Continued noncompliance with the LCO requirements, if the Applicability were entered, would result in the unit being required to exit the Applicability desired to be entered to comply with the Required Actions.

Compliance with Required Actions that permit continued operation of the unit for an unlimited period of time in a MODE or other specified condition provides an acceptable level of safety for continued operation. This is without regard to the status of the unit before or after the MODE change. Therefore, in such cases, entry into a MODE or other specified condition in the Applicability may be made in accordance with the provisions of the Required Actions. The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to OPERABLE status before entering an associated MODE or other specified condition in the Applicability.

(continued)

BASES

---

LCO 3.0.4  
(continued)

The provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any unit shutdown.

Exceptions to LCO 3.0.4 are stated in the individual Specifications. These exceptions allow entry into MODES or other specified conditions in the Applicability when the associated ACTIONS to be entered do not provide for continued operation for an unlimited period of time. Exceptions may apply to all the ACTIONS or to a specific Required Action of a Specification.

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

MODES 5 and 6, or in other specified conditions of the Applicability (unless in MODES 1, 2, 3, or 4) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken. In some cases (e.g., where a review has concluded that specific restriction on MODE changes should be included) these ACTIONS provide a Note that states "While this LCO is not met, entry into a MODE or other specified condition in the Applicability is not permitted, unless required to comply with ACTIONS." This Note is a requirement explicitly precluding entry into a MODE or other specified condition of the Applicability.

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, changing MODES or other specified conditions while in an ACTIONS Condition, in compliance with LCO 3.0.4 or where an exception to LCO 3.0.4 is stated, is not a violation of SR 3.0.1 or SR 3.0.4 for those Surveillances that do not have to be performed due to the associated inoperable equipment. However, SRs must be met to ensure OPERABILITY prior to declaring the associated equipment OPERABLE (or variable within limits) and restoring compliance with the affected LCO.

---

(continued)

BASES (continued)

---

LCO 3.0.5

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

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

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

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

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

---

LCO 3.0.6

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

When a support system is inoperable and there is an LCO specified for it in the TS, the supported system(s) are required to be declared inoperable if determined to be inoperable as a result of the support system inoperability. However, it is not necessary to enter into the supported systems' Conditions and Required Actions unless directed to

(continued)

BASES

LCO 3.0.6  
(continued)

do so by the support system's Required Actions. The potential confusion and inconsistency of requirements related to the entry into multiple support and supported systems' LCOs' Conditions and Required Actions are eliminated by providing all the actions that are necessary to ensure the unit is maintained in a safe condition in the support system's Required Actions.

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

Specification 5.5.15, "Safety Function Determination Program (SFDP)," ensures loss of safety function is detected and appropriate actions are taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other limitations, remedial actions, or compensatory actions may be identified as a result of the support system inoperability and corresponding exception to entering supported system Conditions and Required Actions. The SFDP implements the requirements of LCO 3.0.6. Cross train checks to identify a loss of safety function for those support systems that support multiple and redundant safety systems are required. The cross train check verifies that the supported systems of the redundant OPERABLE support system are OPERABLE, thereby ensuring safety function is retained. If this evaluation determines that a loss of safety function exists, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

LCO 3.0.7

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

(continued)

BASES

---

LCO 3.0.7  
(continued)

be changed to perform the special test or operation will remain in effect.

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

---

---

B 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

BASES

---

SRs SR 3.0.1 through SR 3.0.4 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.

---

SR 3.0.1 SR 3.0.1 establishes the requirement that SRs must be met during the MODES or other specified conditions in the Applicability for which the requirements of the LCO apply, unless otherwise specified in the individual SRs. This Specification is to ensure that Surveillances are performed to verify the OPERABILITY of systems and components, and that variables are within specified limits. Failure to meet a Surveillance within the specified Frequency, in accordance with SR 3.0.2, constitutes a failure to meet an LCO. Systems and components are assumed to be OPERABLE when the associated SRs have been met. Nothing in this Specification, however, is to be construed as implying that systems or components are OPERABLE when:

- a. The systems or components are known to be inoperable, although still meeting the SRs; or
- b. The requirements of the Surveillance(s) are known not to be met between required Surveillance performances.

Surveillances do not have to be performed when the unit is in a MODE or other specified condition for which the requirements of the associated LCO are not applicable, unless otherwise specified. The SRs associated with a test exception are only applicable when the test exception is used as an allowable exception to the requirements of a Specification.

Unplanned events may satisfy the requirements for a given SR. In this case, the unplanned event may be credited as fulfilling the performance of the SR. This allowance includes those SRs whose performance is normally precluded in a given MODE or other specified condition.

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

Upon completion of maintenance, appropriate post maintenance testing is required to declare equipment OPERABLE. This includes ensuring applicable Surveillances are not failed and their most recent

(continued)

BASES

SR 3.0.1  
(continued)

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

SR 3.0.2

SR 3.0.2 establishes the requirements for meeting the specified Frequency for Surveillances and any Required Action with a Completion Time that requires the periodic performance of the Required Action on a "once per . . ." interval. SR 3.0.2 permits a 25% extension of the interval specified in the Frequency. This extension facilitates Surveillance scheduling and considers plant operating conditions that may not be suitable for conducting the Surveillance (e.g., transient conditions or other ongoing Surveillance or maintenance activities).

The 25% extension does not significantly degrade the reliability that results from performing the Surveillance at its specified Frequency. This is based on the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the SRs. The exceptions to SR 3.0.2 are those Surveillances for which the 25% extension of the interval specified in the Frequency does not apply. These exceptions are stated in the individual Specifications. The requirements of regulations take precedence over the TS. Therefore, when a test interval is specified in the regulations, the interval cannot be extended by the TS, and the SR include a Note in the Frequency stating, "SR 3.0.2 is not applicable." An example of an exception when the test interval is not specified in the regulations is the Note in the Containment Leakage Rate Testing Program, "SR 3.0.2 is not applicable." This exception is provided because the program already includes extension of test interval.

As stated in SR 3.0.2, the 25% extension also does not apply to the initial portion of a periodic Completion Time that requires performance on a "once per ..." basis. The 25% extension applies to each performance after the initial performance. The initial performance of the Required Action, whether it is a particular Surveillance 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

(continued)

BASES

---

SR 3.0.2  
(continued)

components or accomplishes the function of the inoperable equipment in an alternative manner.

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

---

SR 3.0.3

SR 3.0.3 establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a Surveillance has not been completed within the specified Frequency. A delay period of up to 24 hours or up to the limit of the specified Frequency, whichever is less, 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 delay period provides adequate time to complete Surveillances that have been missed. This delay period permits the completion of a Surveillance before complying with Required Actions or other remedial measures that might preclude completion of the Surveillance.

The basis for this delay period includes consideration of unit conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, the safety significance of the delay in completing the required Surveillance, and the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the requirements. When a Surveillance with a Frequency based not on time intervals, but upon specified unit conditions or operational situations, is discovered not to have been performed when specified, SR 3.0.3 allows the full delay period of 24 hours to perform the Surveillance.

SR 3.0.3 also provides a time limit for completion of Surveillances that become applicable as a consequence of MODE changes imposed by Required Actions.

Failure to comply with specified Frequencies for SRs is expected to be an infrequent occurrence. Use of the delay period established by SR 3.0.3 is a flexibility which is not intended to be used as an operational convenience to extend Surveillance intervals.

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

(continued)

---

BASES

---

SR 3.0.3  
(continued)

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 applicable SRs must be met before entry into a MODE or other specified condition in the Applicability.

This Specification ensures that system and component OPERABILITY requirements and variable limits are met before entry into MODES or other specified conditions in the Applicability for which these systems and components ensure safe operation of the unit.

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

However, in certain circumstances, failing to meet an SR will not result in SR 3.0.4 restricting a MODE change or other specified condition change. When a system, subsystem, division, component, device, or variable is inoperable or outside its specified limits, the associated SR(s) are not required to be performed, per SR 3.0.1, which states that surveillances do not have to be performed on inoperable equipment. When equipment is inoperable, SR 3.0.4 does not apply to the associated SR(s) since the requirement for the SR(s) to be performed is removed. Therefore, failing to perform the Surveillance(s) within the specified Frequency does not result in an SR 3.0.4 restriction to changing MODES or other specified conditions of the Applicability. However, since the LCO is not met in this instance, LCO 3.0.4 will govern any restrictions that may (or may not) apply to MODE or other specified condition changes.

The provisions of SR 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of SR 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any unit shutdown.

The precise requirements for performance of SRs are specified such that exceptions to SR 3.0.4 are not necessary. The specific time frames and conditions necessary for meeting the SRs are specified in the Frequency, in the Surveillance, or both. This allows performance of Surveillances when the prerequisite condition(s) specified in a Surveillance procedure require entry into the MODE or other specified

(continued)

BASES

---

SR 3.0.4  
(continued)

condition in the Applicability of the associated LCO prior to the performance or completion of a Surveillance. A Surveillance that could not be performed until after entering the LCO Applicability, would have its Frequency specified such that it is not "due" until the specific conditions needed are met. Alternately, the Surveillance may be stated in the form of a Note as not required (to be met or performed) until a particular event, condition, or time has been reached. Further discussion of the specific formats of SRs' annotation is found in Section 1.4, Frequency.

SR 3.0.4 is only applicable when entering MODE 4 from MODE 5, MODE 3 from MODE 4, Mode 2 from MODE 3, or MODE 1 from MODE 2. Furthermore, SR 3.0.4 is applicable when entering any other specified condition in the Applicability only while operating in MODES 1, 2, 3, or 4.

The requirements of SR 3.0.4 do not apply in MODES 5 and 6, or in other specified conditions of the Applicability (unless in MODES 1, 2, 3, or 4) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken.

---

## B 3.1 REACTIVITY CONTROL SYSTEMS

### B 3.1.1 SHUTDOWN MARGIN (SDM)

#### BASES

---

##### BACKGROUND

According to GDC 26 (Ref. 1), the reactivity control systems must be redundant and capable of holding the reactor core subcritical when shut down under cold conditions. Maintenance of the SDM ensures that postulated reactivity events will not damage the fuel.

SDM requirements provide sufficient reactivity margin to ensure that acceptable fuel design limits will not be exceeded for normal shutdown and anticipated operational occurrences (AOOs). As such, the SDM defines the degree of subcriticality that would be obtained immediately following the insertion of all shutdown and control rods, assuming that the single rod cluster assembly of highest reactivity worth is fully withdrawn.

The system design requires that two independent reactivity control systems be provided, and that one of these systems be capable of maintaining the core subcritical under cold conditions. These requirements are provided by the use of movable control assemblies and soluble boric acid in the Reactor Coolant System (RCS). The Rod Control 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 Rod Control 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 Chemical and Volume Control System can control the soluble boron concentration to compensate for fuel depletion during operation and all xenon burnout reactivity changes and can maintain the reactor subcritical under cold conditions.

During power operation, SDM control is ensured, assuming that core reactivity is within design limit of LCO 3.1.2, by operating with the shutdown banks fully withdrawn and the control banks within the limits of LCO 3.1.6, "Control Bank Insertion Limits." When the unit is in the shutdown and refueling modes, the SDM requirements are met by means of adjustments to the RCS boron concentration.

---

##### APPLICABLE SAFETY ANALYSIS

The minimum required SDM is assumed as an initial condition in safety analyses. The safety analysis establishes an SDM that ensures specified acceptable fuel design limits are not exceeded for normal operation and AOOs, with the assumption of the highest worth rod stuck out on scram. For MODE 5, the primary safety analysis that

(continued)

## BASES

APPLICABLE  
SAFETY  
ANALYSIS  
(continued)

relies on the SDM limits is the boron dilution analysis.

The acceptance criteria for the SDM requirements are that specified acceptable fuel design limits are not exceeded. This is done by ensuring that:

- a. The reactor can be made subcritical from all operating conditions, transients, and Design Basis Events;
- b. The reactivity transients associated with postulated accident conditions are controllable within acceptable limits (departure from nucleate boiling ratio (DNBR), fuel centerline temperature limits for AOOs, and  $\leq 200$  cal/gm average fuel pellet enthalpy at the hot spot in irradiated fuel for the rod ejection accident, Ref. 5); and
- c. The reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

The most limiting accidents for the SDM requirements are the main steam line break (MSLB) and inadvertent boron dilution accidents, as described in the FSAR (Refs. 2 and 3). In addition to the limiting MSLB transient, the SDM requirement is also used in the analyses of the following events:

- a. Inadvertent boron dilution;
- b. An uncontrolled rod withdrawal from subcritical or low power condition; and
- c. Start of an inactive reactor coolant pump (RCP); and
- d. Rod ejection.

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 with RCS  $T_{avg}$  equal to 547°F. The positive reactivity addition from the moderator temperature decrease will terminate when the affected SG boils dry, thus terminating RCS heat removal and cooldown. Following the MSLB, a post trip return to power may occur; however, no fuel damage occurs as a result of the post trip return to power, and THERMAL POWER does not violate the Safety Limit (SL) requirement of SL 2.1.1.

(continued)

## BASES

APPLICABLE  
SAFETY  
ANALYSIS  
(continued)

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

The startup of an inactive RCP in MODES 1 or 2 is precluded. In MODE 3, the startup of an inactive RCP cannot 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 start is less than half the minimum required SDM. Startup of an idle RCP cannot, therefore, produce a return to power from the hot standby condition.

SDM satisfies Criterion 2 of 10CFR50.36(c)(2)(ii). Even though it is not directly observed from the control room, SDM is considered an initial condition process variable because it is periodically monitored to ensure that the unit is operating within the bounds of accident analysis assumptions.

## LCO

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

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

(continued)

BASES (continued)

---

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

---

ACTIONS	<p><u>A.1</u></p> <p>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.</p> <p>In the determination of the required combination of boration flow rate and boron concentration, there is no unique requirement that must be satisfied. Since it is imperative to raise the boron concentration of the RCS as soon as possible, the borated water source should be a highly concentrated solution, such as that normally found in the boric acid storage tank, or the refueling water storage tank. The operator should borate with the best source available for the plant conditions.</p>
---------	--

---

SURVEILLANCE REQUIREMENTS	<p><u>SR 3.1.1.1</u></p> <p>In MODES 1 and 2, SDM is verified by observing that the requirements of LCO 3.1.5 and LCO 3.1.6 are met. In the event that a rod is known to be untrippable, however, SDM verification must account for the worth of the untrippable rod as well as another rod of maximum worth.</p> <p>In MODES 2 (with <math>k_{\text{eff}} &lt; 1.0</math>), 3, 4, and 5, the SDM is verified by performing a reactivity balance calculation, considering the listed reactivity effects (SDM verification must account for the worth of the untrippable rod as well as another rod of maximum worth):</p> <ol style="list-style-type: none"> <li>a. RCS boron concentration;</li> <li>b. Control and shutdown rod position;</li> <li>c. RCS average temperature;</li> <li>d. Fuel burnup based on gross thermal energy generation;</li> <li>e. Xenon concentration;</li> <li>f. Samarium concentration; and</li> <li>g. Isothermal temperature coefficient (ITC).</li> </ol>
---------------------------	--

(continued)

BASES

---

SURVEILLANCE  
REQUIREMENTS

SR 3.1.1.1 (continued)

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

The Frequency of 24 hours is based on the generally slow change in required boron concentration and the low probability of an accident occurring without the required SDM. This allows time for the operator to collect the required data, which includes performing a boron concentration analysis, and complete the calculation.

---

REFERENCES

- 1 10 CFR 50, Appendix A, GDC 26.
  - 2 FSAR, Chapter 15, Section 15.4.2.1.
  - 3 FSAR, Chapter 15, Section 15.2.4.
  - 4 10 CFR 100.
  - 5 FSAR, Chapter 15, Section 15.4.6.1.6.
- 
-

## B 3.1 REACTIVITY CONTROL SYSTEMS

### B 3.1.2 Core Reactivity

#### BASES

---

##### BACKGROUND

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

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

In order to achieve the required fuel cycle energy output, the uranium enrichment, in the new fuel loading and in the fuel remaining from the previous cycle, provides excess positive reactivity beyond that required to sustain steady state operation throughout the cycle. When the reactor is critical at RTP and moderator temperature, the excess

(continued)

BASES

---

BACKGROUND  
(continued)

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 ensures plant operation is maintained within the assumptions of the safety analyses.

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

Design calculations 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 as well as providing inputs to the safety analysis.

The comparison between measured and predicted initial core reactivity provides a validation of 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 calculational 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 calculational model is not adequate for core

(continued)

---

BASES

---

APPLICABLE  
SAFETY  
ANALYSES  
(continued)

burnups beyond BOC, or that an unexpected change in core conditions has occurred.

The normalization of predicted RCS boron concentration to the measured value when deemed necessary shall be 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.

Core reactivity satisfies Criterion 2 of 10CFR50.36(c)(2)(ii).

---

LCO

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

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

---

APPLICABILITY

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

---

(continued)

BASES

---

APPLICABILITY  
(continued)

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

---

ACTIONS

A.1 and A.2

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

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

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

(continued)

---

BASES

---

ACTIONS  
(continued)

B.1

If the core reactivity cannot be restored to within the 1%  $\Delta k/k$  limit, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours. If the SDM for MODE 3 is not met, then the boration required by LCO 3.1.1 Required Action A.1 would occur. The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

---

SURVEILLANCE  
REQUIREMENTS

SR 3.1.2.1

Core reactivity is verified by periodic comparisons of measured and predicted RCS boron concentrations. The comparison is made, considering that other core conditions are fixed or stable, including control rod position, moderator temperature, fuel temperature, fuel depletion, xenon concentration, and samarium concentration. The Surveillance is performed prior to entering MODE 1 as an initial check on core conditions and design calculations at BOC. The SR is modified by a Note. The Note indicates that the normalization (adjustment, only if necessary) of predicted core reactivity to the measured value must take place within the first 60 effective full power days (EFPD) after each fuel loading. This allows sufficient time for core conditions to reach steady state, but prevents operation for a large fraction of the fuel cycle without establishing a benchmark for the design calculations. The required subsequent Frequency of 31 EFPD, following the initial 60 EFPD after entering MODE 1, is acceptable, based on the slow rate of core changes due to fuel depletion and the presence of other indicators (QPTR, AFD, etc.) for prompt indication of an anomaly.

---

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26, GDC 28, and GDC 29.
  2. FSAR, Chapter 15.
- 
-

## B 3.1 REACTIVITY CONTROL SYSTEMS

### B 3.1.3 Moderator Temperature Coefficient (MTC)

#### BASES

---

##### BACKGROUND

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

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

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

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

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

(continued)

BASES

BACKGROUND  
(continued)

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

APPLICABLE  
SAFETY  
ANALYSES

The acceptance criteria for the specified MTC are:

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

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

The consequences of accidents that cause core overheating must be evaluated when the MTC is positive. Such accidents include the rod withdrawal transient from either zero (Ref. 2) or RTP, loss of main feedwater flow, and loss of forced reactor coolant flow. The consequences of accidents that cause core overcooling must be evaluated when the MTC is negative. Such accidents include sudden feedwater flow increase and sudden decrease in feedwater temperature.

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

MTC values are bounded in reload safety evaluations assuming steady state conditions at BOC and EOC. An EOC measurement is conducted at conditions when the RCS boron concentration reaches a boron concentration equivalent to 300 ppm at an equilibrium, all rods out, RTP condition. The measured value may be extrapolated to project the EOC value, in order to confirm reload design predictions.

The most negative MTC value, equivalent to the most positive moderator density coefficient (MDC), was obtained by incrementally

(continued)

BASES

---

APPLICABLE  
SAFETY  
ANALYSES  
(continued)

correcting the MDC used in the FSAR analyses to nominal operating conditions. These corrections involved: (1) a conversion of the MDC used in the FSAR accident analyses to its equivalent MTC, based on the rate of change of moderator density with temperature at RATED THERMAL POWER conditions, and (2) adding margin to this value to account for the largest difference in MTC observed between an EOC, all rods withdrawn, RATED THERMAL POWER condition and an envelope of those most adverse conditions of moderator temperature and pressure, rods inserted to their insertion limits, axial power skewing, and xenon concentration that can occur in normal operation within Technical Specification limits and lead to a significantly more negative EOC MTC at RATED THERMAL POWER. These corrections transformed the MDC value used in the FSAR accident analyses into the limiting EOC MTC value. The 300 ppm surveillance limit MTC value represents a conservative value (with corrections for burnup and soluble boron) at a core condition of 300 ppm equilibrium boron concentration and is obtained by adding an allowance for burnup and soluble boron concentration changes to the limiting EOC MTC value.

MTC satisfies Criterion 2 of 10CFR50.36(c)(2)(ii). Even though it is not directly observed and controlled from the control room, MTC is considered an initial condition process variable because of its dependence on boron concentration.

---

LCO

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

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

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

---

(continued)

BASES

---

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

---

APPLICABILITY	<p>Technical Specifications place both LCO and SR values on MTC, based on the safety analysis assumptions described above.</p> <p>In MODE 1, the limits on MTC must be maintained to ensure that any accident initiated from THERMAL POWER operation will not violate the design assumptions of the accident analysis. In MODE 2 with the reactor critical, the upper limit 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. The lower MTC limit must be maintained in MODES 2 and 3, in addition to MODE 1, to ensure that cooldown accidents will not violate the assumptions of the accident analysis. In MODES 4, 5, and 6, this LCO is not applicable, since no Design Basis Accidents using the MTC as an analysis assumption are initiated from these MODES.</p>
---------------	--

---

ACTIONS	<p><u>A.1</u></p> <p>If the upper MTC limit is violated, administrative withdrawal limits for control banks must be established to maintain the MTC within its limits. The MTC becomes more negative with control bank insertion and decreased boron concentration. A Completion Time of 24 hours provides enough time for evaluating the MTC measurement and computing the required bank withdrawal limits.</p>
---------	--

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

B.1

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

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

(continued)

BASES

---

ACTIONS  
(continued)

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 brought to a MODE or condition in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least MODE 4 within 12 hours.

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

---

SURVEILLANCE  
REQUIREMENTS

SR 3.1.3.1

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 BOC limit prior to entering MODE 1 ensures that the limit will also be met at higher power levels.

The BOC MTC value for ARO will be inferred from isothermal temperature coefficient measurements obtained during the physics tests after refueling. The ARO value can be directly compared to the 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.

SR 3.1.3.2

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 ensure that the LCO limit will be met when the 300 ppm Surveillance criterion is met.

(continued)

---

BASES

---

SURVEILLANCE  
REQUIREMENTS

SR 3.1.3.2 (continued)

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

1. The SR is required to be performed once each cycle within 7 effective full power days (EFPDs) after reaching the equivalent of an equilibrium RTP all rods out (ARO) boron concentration of 300 ppm.
2. If the 300 ppm Surveillance limit is exceeded, it is possible that the EOC limit on MTC could be reached before the planned EOC. Because the MTC changes slowly with core depletion, the Frequency of 14 EFPD is sufficient to avoid exceeding the EOC limit.
3. The Surveillance limit for RTP boron concentration of 60 ppm is conservative. If the measured MTC at 60 ppm is less negative 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. 10 CFR 50, Appendix A, GDC 11.
  2. FSAR, Chapter 15.
  3. WCAP-9273-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985.
- 
-

## B 3.1 REACTIVITY CONTROL SYSTEMS

### B 3.1.4 Rod Group Alignment Limits

#### BASES

---

##### BACKGROUND

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

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

Mechanical or electrical failures may cause a control 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 have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

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

The RCCAs are divided among four control banks and four 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 and are moved in a staggered fashion, but always within one step of each other. All control banks contain two rod groups. Two shutdown banks (A and B) contain two rod groups and the remaining two shutdown banks (C and D) contain one rod group.

---

(continued)

BASES

---

BACKGROUND  
(continued)

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 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  $\pm 5/8$  inch). If a rod does not move one step for each demand pulse, the step counter will still count the pulse and incorrectly reflect the position of the rod.

The 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. To increase the reliability of the system, the inductive coils are connected alternately to data system A or B. Thus, if one data system fails, the DRPI will go on half accuracy. The DRPI system is capable of monitoring rod position within at least  $\pm 12$  steps with either full accuracy or half accuracy.

---

(continued)

BASES (continued)

---

APPLICABLE  
SAFETY  
ANALYSIS

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

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

Two types of misalignment are distinguished. During movement of a control or shutdown 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 rods to meet the SDM requirement, with the maximum worth rod stuck fully withdrawn.

Two types of analysis are performed in regard to static rod misalignment (Ref. 3). With control banks at their insertion limits, one type of analysis considers the case when any one rod is completely inserted into the core. The second type of analysis considers the case of a completely withdrawn single rod from bank D 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. 4).

The Required Actions in this LCO ensure 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 rod is allowed if the heat flux hot channel factor ( $F_Q(Z)$ ) and the nuclear enthalpy hot channel factor ( $F_{\Delta H}^N$ ) are verified to be within their limits in the COLR and the safety analysis is verified to remain valid. When a control rod is misaligned, the assumptions that are used to determine the rod

(continued)

BASES

---

APPLICABLE  
ANALYSIS  
(continued)

insertion limits, AFD limits, and quadrant power tilt limits are not preserved. Therefore, the limits may not preserve the design peaking factors, and  $F_Q(Z)$  and  $F_{\Delta H}^N$  must be verified directly by incore mapping. Bases Section 3.2 (Power Distribution Limits) contains more complete discussions of the relation of  $F_Q(Z)$  and  $F_{\Delta H}^N$  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 10CFR50.36(c)(2)(ii).

---

LCO

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

The requirement to maintain the rod alignment to within plus or minus 12 steps of their group step counter demand position 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.

The requirement to maintain rod alignment is met by comparing individual rod DRPI indication and bank demand position indication to be within plus or minus 12 steps. If one of these position indicators become inoperable, the conditions of this LCO are still met by compliance with LCO 3.1.7.

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.

---

(continued)

BASES (continued)

---

**APPLICABILITY** The requirements on RCCA OPERABILITY and alignment are applicable in MODES 1 and 2 because these are the only MODES in which neutron (or fission) power is generated, and the OPERABILITY (i.e., trippability) and alignment of rods have the potential to affect the safety of the plant. In MODES 3, 4, 5, and 6, the alignment limits do not apply because the control rods are typically fully inserted and the reactor is shut down and not producing fission power. In the shutdown MODES, the OPERABILITY of the shutdown and control rods has the potential to affect the required SDM, but this effect can be compensated for by an increase in the boron concentration of the RCS. See LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," for SDM in MODES 2 with  $k_{\text{eff}} < 1.0$ , 3, 4, and 5 and LCO 3.9.1, "Boron Concentration," for boron concentration requirements during refueling.

---

**ACTIONS**

A.1.1 and A.1.2

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

With an inoperable rod(s), this ACTION provides for verification of SDM, this is most simply accomplished by verifying rod insertion limits are met. Additionally, actions could include calculation of the current SDM and boration to meet limits specified in the COLR or proceed to MODE 3. These actions are consistent with those specified in LCO 3.1.5 and LCO 3.1.6.

A rod is considered trippable if it was demonstrated OPERABLE during the last performance of SR 3.1.4.2 and met the rod drop time criteria during the last performance of SR 3.1.4.3.

In this situation, SDM verification must account for the absence of the negative reactivity of the untrippable rod(s), as well as the rod of maximum worth.

A.2

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

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

(continued)

BASES

---

ACTIONS  
(continued)

B.1

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

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

B.2.1.1 and B.2.1.2

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

In many cases, realigning the remainder of the group to the misaligned rod may not be desirable. 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 fully inserted and control bank C must be inserted to approximately 100 steps.

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

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

For continued operation with a misaligned rod, reactor power must be reduced, SDM must periodically be verified within limits, hot channel factors ( $F_Q(Z)$  and  $F_{\Delta H}^N$ ) must be verified within limits, and the safety analyses must be re-evaluated to confirm continued operation is permissible. Reduction of power to 75% RTP ensures that local LHR increases due to a misaligned RCCA will not cause the core design criteria to be exceeded (Ref. 3). The Completion Time of 2 hours gives the operator sufficient time to accomplish an orderly power reduction without challenging the Reactor Protection System.

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

(continued)

BASES

---

ACTIONS

B.2.2, B.2.3, B.2.4, B.2.5, and B.2.6 (continued)

Verifying that  $F_Q(Z)$  and  $F_{\Delta H}^N$  are within the required limits ensures that current operation at 75% RTP with a rod misaligned is not resulting in power distributions that may invalidate safety analysis assumptions at full power. The Completion Time of 72 hours allows sufficient time to obtain flux maps of the core power distribution using the incore flux mapping system and to calculate  $F_Q(Z)$  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. The accident analyses of FSAR Chapter 15 are to be used to identify the appropriate design bases events requiring re-evaluation. A Completion Time of 5 days is sufficient time to obtain the required input data and to perform the analysis.

C.1

When Required Actions cannot be completed within their Completion Time, the unit must be brought to a MODE or Condition in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours, 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, for reaching MODE 3 from full power conditions in an orderly manner and without challenging the plant systems.

D.1.1 and D.1.2

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

Additionally, the requirements of LCO 3.1.5, "Shutdown Bank Insertion Limits," and LCO 3.1.6, "Control Bank Insertion Limits," apply if the misaligned rods are not within the required insertion limits.

(continued)

BASES

---

ACTIONS  
(continued)

D.2

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

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

---

SURVEILLANCE  
REQUIREMENTS

SR 3.1.4.1

Verification that individual rod positions are within alignment limits at a Frequency of 12 hours provides a history that allows the operator to detect a rod that is beginning to deviate from its expected position. 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.

SR 3.1.4.2

Verifying each rod is OPERABLE would require that each rod be tripped. However, in MODES 1 and 2, tripping each rod would result in radial or axial power tilts, or oscillations. Exercising each individual rod every 92 days provides confidence that all rods continue to be OPERABLE without exceeding the alignment limit, even if they are not regularly tripped. Moving each rod by 10 steps will not cause radial or axial power tilts, or oscillations, to occur. The 92 day Frequency takes into consideration other information available to the operator in the control room and SR 3.1.4.1, which is performed more frequently and adds to the determination of OPERABILITY of the rods. Between or during required performances of SR 3.1.4.2 (determination of rod OPERABILITY by movement), if a rod(s) is discovered to be immovable, but remains trippable, the rod(s) is considered to be OPERABLE. At any time, if a rod(s) is immovable, a determination of the trippability (OPERABILITY) of the rod(s) must be made, and appropriate action taken.

SR 3.1.4.3

Verification of rod drop times allows the operator to determine that the maximum rod drop time permitted is consistent with the assumed rod drop time used in the safety analysis. Measuring rod drop times prior

(continued)

BASES

---

SURVEILLANCE  
REQUIREMENTS

SR 3.1.4.3 (continued)

to reactor criticality, after reactor vessel head removal, ensures that the reactor internals and rod drive mechanism will not interfere with rod motion or rod drop time, and that no degradation in these systems has occurred that would adversely affect rod motion or drop time. This testing is performed with all RCPs operating and the average moderator temperature  $\geq 500^{\circ}\text{F}$  to simulate a reactor trip under actual conditions.

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

---

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10 and GDC 26.
  2. 10 CFR 50.46.
  3. FSAR, Chapter 15, Section 15.2.3.
  4. FSAR, Chapter 15, Section 4.2.3.
- 
-

## B 3.1 REACTIVITY CONTROL SYSTEMS

### B 3.1.5 Shutdown Bank Insertion Limits

#### BASES

---

##### BACKGROUND

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

The applicable criteria for these reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10, "Reactor Design," GDC 26, "Reactivity Control System Redundancy and Capability," GDC 28, "Reactivity Limits" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 2). Limits on 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 four control banks and four 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 four control banks contain two rod groups. Two shutdown banks (A and B) contain two rod groups and the remaining two shutdown banks (C and D) consist of a single group. See LCO 3.1.4, "Rod Group Alignment Limits," for control and shutdown rod OPERABILITY and alignment requirements, and LCO 3.1.7, "Rod Position Indication," for position indication requirements.

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

(continued)

BASES

---

BACKGROUND  
(continued)

without the core going critical. This provides available negative reactivity in the event of boration errors. 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 affect core power and burnup distribution, and add negative reactivity to shut down the reactor upon receipt of a reactor trip signal.

---

APPLICABLE  
SAFETY  
ANALYSIS

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

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

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

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

The shutdown bank insertion limits preserve an initial condition assumed in the safety analyses and, as such, satisfy Criterion 2 of 10CFR50.36(c)(2)(ii).

---

(continued)

BASES

---

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

The shutdown bank insertion limits are defined in the COLR.

---

APPLICABILITY The shutdown banks must be within their insertion limits, with the reactor in MODES 1 and 2. The applicability in MODE 2 begins at initial control bank withdrawal, during an approach to criticality, and continues throughout MODE 2, until all control bank rods are again fully inserted by reactor trip or by shutdown. This ensures that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM following a reactor trip. The shutdown banks do not have to be within their insertion limits in MODE 3, unless an approach to criticality is being made. In MODE 3, 4, 5, or 6, the shutdown banks are, typically, fully inserted in the core and contribute to the SDM. Refer to LCO 3.1.1 for SDM requirements in MODES 2 with  $k_{eff} < 1.0$ , 3, 4, and 5. LCO 3.9.1, "Boron Concentration," ensures adequate SDM in MODE 6.

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

---

ACTIONS A.1.1, A.1.2, and A.2

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

The allowed Completion Time of 2 hours provides an acceptable time for evaluating and repairing minor problems without allowing the plant to remain in an unacceptable condition for an extended period of time. Additionally, the requirements of LCO 3.1.4, "Rod Group Alignment Limits," apply if one or more shutdown rods are not within the required alignment limits.

(continued)

---

BASES

---

ACTIONS  
(continued)

B.1

If the shutdown banks cannot be restored to within their insertion limits within 2 hours, the unit must be brought to a MODE where the LCO is not applicable. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

---

SURVEILLANCE  
REQUIREMENTS

SR 3.1.5.1

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

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 in the control room for the purpose of monitoring the status of shutdown rods.

---

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10, GDC 26, and GDC 28.
  2. 10 CFR 50.46.
  3. FSAR, Chapter 15, Section 15.4.3.2.4.
- 
-

## B 3.1 REACTIVITY CONTROL SYSTEMS

### B 3.1.6 Control Bank Insertion Limits

#### BASES

---

##### BACKGROUND

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

The applicable criteria for these reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10, "Reactor Design," GDC 26, "Reactivity Control System Redundancy and Capability," GDC 28, "Reactivity Limits" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 2). Limits on 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 four control banks and four 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 and are moved in a staggered fashion, but always within one step of each other. Two shutdown banks (C and D) consist of a single group. See LCO 3.1.4, "Rod Group Alignment Limits," for control and shutdown rod OPERABILITY and alignment requirements, and LCO 3.1.7, "Rod Position Indication," for position indication requirements.

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

The COLR also indicates how the control banks are moved in an overlap pattern. Overlap is the distance traveled together by two control banks. The control banks are used for precise reactivity control of the reactor. The positions of the control banks can be controlled manually, or automatically by the Rod Control System. They are capable of adding reactivity very quickly (compared to borating or diluting).

---

(continued)

BASES

---

BACKGROUND  
(continued)

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

The shutdown and control bank insertion and alignment limits, AFD, and QPTR are process variables that together characterize and control the three dimensional power distribution of the reactor core. Additionally, the control bank insertion limits control the reactivity that could be added in the event of a rod ejection accident, and the shutdown and control bank insertion limits ensure the required SDM is maintained assuming LCO 3.1.2, "Core Reactivity" is met for core reactivity.

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 an RTS trip function.

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

There be no violations of:

Specified acceptable fuel design limits, or

Reactor Coolant System pressure boundary integrity; and

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

(continued)

BASES

---

APPLICABLE  
SAFETY  
ANALYSES  
(continued)

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 that 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 10CFR50.36(c)(2)(ii), in that they are initial conditions assumed in the safety analysis.

---

LCO

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

---

APPLICABILITY

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

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

---

ACTIONS

A.1.1, A.1.2, A.2, B.1.1, B.1.2, 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:

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

(continued)

BASES

---

ACTIONS

A.1.1, A.1.2, A.2, B.1.1, B.1.2, and B.2 (continued)

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

Similarly, if the control banks are found to be out of sequence or in the wrong overlap configuration, they must be restored to meet the limits. Failure of sequence or overlap support equipment does not require entering the ACTIONS as long as sequence and overlap limits are maintained.

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

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

Additionally, the requirements of LCO 3.1.4, "Rod Group Alignment Limits," apply if one or more control rods are not within the required alignment limits.

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 brought to MODE 3, where the LCO is not applicable. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

---

(continued)

BASES (continued)

---

SURVEILLANCE  
REQUIREMENTS

SR 3.1.6.1

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

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

SR 3.1.6.2

Verification of the control bank insertion limits at a Frequency of 12 hours is sufficient to ensure OPERABILITY and to detect control banks that may be approaching the insertion limits since, normally, very little rod motion occurs in 12 hours.

SR 3.1.6.3

When control banks are maintained within their insertion limits as checked by SR 3.1.6.2 above, it is unlikely that their sequence and overlap will not be in accordance with requirements provided in the COLR. The verification of compliance with the sequence and overlap limits specified in the COLR consists of an observation that the static rod positions of those control banks not fully withdrawn from the core are within the limits specified in the COLR. Bank sequence and overlap must also be maintained during rod movement, implicit within the LCO. A Frequency of 12 hours is consistent with the insertion limit check above in SR 3.1.6.2.

---

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10, GDC 26, GDC 28.
  2. 10 CFR 50.46.
  3. FSAR, Chapter 4, Section 4.3.2.4.
  4. FSAR, Chapter 4, Section 4.3.2.4.
  5. WCAP-9273-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985.
-

## B 3.1 REACTIVITY CONTROL SYSTEM

### B 3.1.7 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 3.1.7 is required to ensure OPERABILITY of the rod position indicators to determine rod positions and thereby ensure compliance with the rod alignment and insertion limits.

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

Mechanical or electrical failures may cause a rod to become inoperable or to become misaligned from its group. 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.

Limits on 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 rod drive mechanisms. The RCCAs are divided among four control banks and four shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control.

The axial position of shutdown rods and control rods are determined by two separate and independent systems: the 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  $\pm 5/8$  inch). If a rod does not move one step for each demand pulse,

(continued)

BASES

---

BACKGROUND  
(continued)

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 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. To increase the reliability of the system, the inductive coils are connected alternately to data system A or B. Thus, if one data system fails, the DRPI will go on half accuracy. The DRPI system is capable of monitoring rod position within at least  $\pm 12$  steps with either full accuracy or half accuracy.

---

APPLICABLE  
SAFETY  
ANALYSIS

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

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

---

LCO

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

- a. The DRPI System on either full accuracy or half accuracy indicates within 12 steps of the group step counter demand position as required by LCO 3.1.4, "Rod Group Alignment Limits"; and
- b. The Bank Demand Indication System has been reset in the fully inserted position, fully withdrawn position or to the DRPI System.

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

---

(continued)

BASES

---

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

These requirements ensure that 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 rods can be detected. Therefore, power peaking, ejected rod worth, and SDM can be controlled within acceptable limits.

---

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

---

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

A.1

When one DRPI per group fails, the position of the rod may still be determined indirectly by use of the movable incore detectors. The Required Action may also be ensuring, at least once per 8 hours, that  $F_Q$  satisfies LCO 3.2.1,  $F_{\Delta H}^N$  satisfies LCO 3.2.2, and SDM is within the limits provided in the COLR, provided the nonindicating rods have not been moved. Based on experience, normal power operation does not require excessive movement of banks. If a bank has been significantly moved, the Required Action of C.1 or C.2 below is required. Therefore, verification of RCCA position within the Completion Time of 8 hours is adequate for allowing continued full power operation, since the probability of simultaneously having a rod significantly out of position and an event sensitive to that rod position is small.

(continued)

BASES

---

ACTIONS  
(continued)

A.2

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

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

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

When more than one DRPI per group fail, additional actions are necessary to ensure that acceptable power distribution limits are maintained, minimum SDM is maintained, and the potential effects of rod misalignment on associated accident analyses are limited. Placing the Rod Control System in manual assures unplanned rod motion will not occur. Together with the indirect position determination available via movable incore detectors, this will minimize the potential for rod misalignment.

The immediate Completion Time for placing the Rod Control System in manual reflects the urgency with which unplanned rod motion must be prevented while in the Condition. Monitoring and recording reactor coolant  $T_{avg}$  help assure that significant changes in power distribution and SDM are avoided. The once per 1 hour Completion Time is acceptable because only minor fluctuations in RCS temperature are expected at steady state plant operating conditions.

The position of the rods can be determined indirectly by use of the movable incore detectors. The Required Action may also be satisfied by ensuring at least once per 8 hours that  $F_Q$  satisfies LCO 3.2.1,  $F_{\Delta H}^N$  satisfies LCO 3.2.2, and SHUTDOWN MARGIN is within the limits provided in the COLR, provided that the nonindicating rods have not moved. Verification of RCCA position once per 8 hours is adequate for allowing continued full power operation for a limited, 24 hour period, since the probability of simultaneously having a rod significantly out of position and an event sensitive to that rod position is small. The 24 hour Allowed Outage Time provides sufficient time to troubleshoot and restore the DRPI system to operation while avoiding the plant challenges associated with a shutdown without full rod position indication.

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

(continued)

BASES

---

ACTIONS  
(continued)

C.1 and C.2

These Required Actions clarify that when one or more rods with inoperable DRPIs have been moved in excess of 24 steps in one direction, since the position was last determined, the Required Actions of A.1 and A.2 or B.1 are still appropriate but must be initiated promptly under Required Action C.1 to begin indirectly verifying that these rods are still properly positioned, relative to their group positions. If, within 4 hours, the rod positions have not been determined, THERMAL POWER must be reduced to  $\leq 50\%$  RTP within 8 hours to avoid undesirable power distributions that could result from continued operation at  $> 50\%$  RTP, if one or more rods are misaligned by more than 24 steps. The allowed Completion Time of 4 hours provides an acceptable period of time to verify the rod positions using the movable incore detectors.

D.1.1 and D.1.2

With one demand position indicator per bank inoperable, the rod positions can be determined by the DRPI System. Since normal power operation does not require excessive movement of rods, verification by administrative means that the rod position indicators are OPERABLE and the most withdrawn rod and the least withdrawn rod are  $\leq 12$  steps apart within the allowed Completion Time of once every 8 hours is adequate.

D.2

Reduction of THERMAL POWER to  $\leq 50\%$  RTP puts the core into a condition where rod position is not significantly affecting core peaking factors (Ref. 3). The allowed Completion Time of 8 hours provides an acceptable period of time to verify the rod positions per Required Actions D.1.1 and D.1.2 or reduce power to  $\leq 50\%$  RTP.

E.1

If the Required Actions cannot be completed within the associated Completion Time, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours. The allowed Completion Time is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

---

(continued)

BASES (continued)

---

SURVEILLANCE  
REQUIREMENTS

SR 3.1.7.1

Verification that the DRPI agrees with the demand position within 12 steps ensures that the DRPI is operating correctly. Verification at 24, 48, 120, and 228 steps withdrawn for the control and shutdown banks provides assurance that the DRPI is operating correctly over the full range of indication.

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

---

REFERENCES

1. 10 CFR 50, Appendix A, GDC 13.
  2. FSAR, Chapter 15.
  3. WCAP-10216-P-A, Rev. 1A, "Relaxation of Constant Axial Offset Control and  $F_o$  Surveillance Technical Specification," February 1994.
- 
-

## B 3.1 REACTIVITY CONTROL SYSTEMS

### B 3.1.8 PHYSICS TESTS Exceptions - MODE 2

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

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

PHYSICS TESTS 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.

---

(continued)

BASES

---

BACKGROUND  
(continued)

The PHYSICS TESTS required for reload fuel cycles in MODE 2 typically include:

- a. Critical Boron Concentration,
- b. Control Rod Worth, and
- c. Isothermal Temperature Coefficient (ITC).

These tests may cause the operating controls and process variables to deviate from their LCO requirements during their performance.

---

APPLICABLE  
SAFETY  
ANALYSES

The fuel is protected by LCOs that preserve the initial conditions of the core assumed during the safety analyses. The methods for development of the LCOs that are excepted by this LCO are described in the 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.

Reload fuel cycle PHYSICS TESTS are performed in accordance with Technical Specification requirements, fuel vendor guidelines, and established industry practices. 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 3.1.3, "Moderator Temperature Coefficient (MTC)," LCO 3.1.4, LCO 3.1.5, LCO 3.1.6, and LCO 3.4.2 are suspended for PHYSICS TESTS, the fuel design criteria are preserved as long as the power level is limited to  $\leq 5\%$  RTP, the reactor coolant temperature is kept  $\geq 531^\circ\text{F}$ , and SDM is within the limits provided in the COLR.

The PHYSICS TESTS include measurement of core nuclear parameters or the exercise of control components that affect process variables. 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 COLR. 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 10 CFR 50.36(c)(2)(ii).

Reference 6 allows special test exceptions (STEs) 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.

---

BASES (continued)

---

LCO This LCO allows the reactor parameters of MTC and minimum temperature for criticality to be outside their specified limits. In addition, it allows selected control and shutdown rods to be positioned outside of their specified alignment and insertion limits. Operation beyond specified limits is permitted for the purpose of performing PHYSICS TESTS and poses no threat to fuel integrity, provided the SRs are met.

The requirements of LCO 3.1.3, LCO 3.1.4, LCO 3.1.5, LCO 3.1.6, and LCO 3.4.2 may be suspended during the performance of PHYSICS TESTS provided:

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

---

APPLICABILITY This LCO is applicable when performing low power PHYSICS TESTS. The applicable PHYSICS TESTS are performed in MODE 2 at HZP.

---

ACTIONS A.1 and A.2

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

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

B.1

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

C.1

When the RCS lowest operating loop's  $T_{\text{avg}}$  is  $< 531^{\circ}\text{F}$ , the appropriate action is to restore  $T_{\text{avg}}$  to within its specified limit. The allowed Completion Time of 15 minutes provides time for restoring  $T_{\text{avg}}$  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 an operating loop's temperature below  $531^{\circ}\text{F}$  could violate the assumptions for accidents analyzed in the safety analyses.

(continued)

BASES

---

ACTIONS  
(continued)

D.1

If the Required Actions cannot be completed within the associated Completion Time, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within an additional 15 minutes. The Completion Time of 15 additional minutes is reasonable, based on operating experience, for reaching MODE 3 in an orderly manner and without challenging plant systems.

---

SURVEILLANCE  
REQUIREMENTS

SR 3.1.8.1

A CHANNEL OPERATIONAL TEST is performed on each required power range and intermediate range nuclear instrument in MODE 2 by LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation." SR 3.1.8.1 verifies that the above surveillances are current, ensuring that the RTS is properly aligned to provide the required degree of core protection prior to initiation and during the performance of PHYSICS TESTS.

SR 3.1.8.2

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

SR 3.1.8.3

Verification that the THERMAL POWER is  $\leq 5\%$  RTP will ensure that the plant is not operating in a condition that could invalidate the safety analyses. Verification of the THERMAL POWER at a Frequency of 1 hour during the performance of the PHYSICS TESTS will ensure that the initial conditions of the safety analyses are not violated.

SR 3.1.8.4

Verification that the SDM is within limits specified in the COLR ensures that, for the specific RCCA and RCS temperature manipulations performed during PHYSICS TESTS, the plant is not operating in a condition that could invalidate the safety analysis assumptions.

The SDM for physics testing during tests where traditional SDM monitoring techniques are not adequate, is determined for the most restrictive test based on design calculations. Plant conditions are monitored during these tests to verify adequate SDM.

---

(continued)

BASES

---

SURVEILLANCE  
REQUIREMENTS

SR 3.1.8.4 (continued)

The Frequency of 24 hours is based on the generally slow change in required boron concentration and on the low probability of an accident occurring without the required SDM.

---

REFERENCES

1. 10 CFR 50, Appendix B, Section XI.
  2. 10 CFR 50.59.
  3. Regulatory Guide 1.68, Revision 2, August, 1978.
  4. Not used.
  5. WCAP-9273-NP-A, "Westinghouse Reload Safety Evaluation Methodology Report," July 1985.
  6. WCAP-11618, including Addendum 1, April 1989.
- 
-

## B 3.2 POWER DISTRIBUTION LIMITS

### B 3.2.1 Heat Flux Hot Channel Factor (F<sub>q</sub>(Z))

#### BASES

---

##### BACKGROUND

The purpose of the limits on the values of F<sub>q</sub>(Z) is to limit the local (i.e., pellet) peak power density. The value of F<sub>q</sub>(Z) varies along the axial height (Z) of the core.

F<sub>q</sub>(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<sub>q</sub>(Z) is a measure of the peak fuel pellet power within the reactor core.

During power operation, the global power distribution is limited by LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," which are directly and continuously measured process variables. These LCOs, along with LCO 3.1. 6, "Control Bank Insertion Limits," maintain the core limits on power distributions on a continuous basis.

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

F<sub>q</sub>(Z) is not directly measurable but is inferred from a power distribution map obtained with the movable incore detector system. The results of the power distribution map are analyzed to derive a measured value for F<sub>q</sub>(Z). These measurements are generally taken with the core at or near equilibrium conditions.

However, because this value represents an equilibrium condition, it does not include the variations in the value of F<sub>q</sub>(Z) that are present during nonequilibrium situations, such as load following.

To account for these possible variations, a transient F<sub>q</sub>(Z) is also calculated based on the steady state value of F<sub>q</sub>(Z). In this case, the steady state F<sub>q</sub>(Z) is adjusted by an elevation dependent factor, W(Z), that accounts for the calculated transient conditions.

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

---

##### APPLICABLE SAFETY ANALYSES

This LCO's principal effect is to preclude core power distributions that could lead to violation of the following fuel design criterion:

During a large break loss of coolant accident (LOCA), there is a high level of probability that the peak cladding temperature will not exceed 2200° F (Ref. 1).

(continued)

BASES

APPLICABLE  
SAFETY  
ANALYSES  
(continued)

Limits on F<sub>q</sub>(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 LOCA peak cladding temperature is typically most limiting.

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

F<sub>q</sub>(Z) satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

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

$$F_q(Z) \leq \frac{F_q^{RTP}}{P} K(Z) \text{ for } P > 0.5$$

$$F_q(Z) \leq \frac{F_q^{RTP}}{0.5} K(Z) \text{ for } P \leq 0.5$$

where: F<sub>q</sub><sup>RTP</sup> is the F<sub>q</sub>(Z) limit at RATED THERMAL POWER (RTP) provided in the COLR,

K(Z) is the F<sub>q</sub>(Z) normalization factor for core height provided in the COLR, and

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

The actual values of F<sub>q</sub><sup>RTP</sup> and K(Z) are given in the COLR.

For Relaxed Axial Offset Control operation, F<sub>q</sub>(Z) is approximated by F<sub>q</sub><sup>c</sup>(Z) and F<sub>q</sub><sup>w</sup>(Z). Thus, both F<sub>q</sub><sup>c</sup>(Z) and F<sub>q</sub><sup>w</sup>(Z) must meet the preceding limits on F<sub>q</sub>(Z).

(continued)

BASES

LCO  
(continued)

An F<sub>q</sub><sup>c</sup>(Z) evaluation requires obtaining an incore flux map in MODE 1. From the incore flux map results we obtain the measured value (F<sub>q</sub><sup>m</sup>(Z)) of F<sub>q</sub>(Z). The computed heat flux hot channel factor, F<sub>q</sub><sup>c</sup>(Z) is obtained by the equation:

$$F_{q}^{c}(Z) = F_{q}^{m}(Z) (1.03) (1.05)$$

where 1.03 is a factor that accounts for fuel manufacturing tolerances and 1.05 is a factor that accounts for flux map measurement uncertainty.

The expression for F<sub>q</sub><sup>w</sup>(Z) is:

$$F_{q}^{w}(Z) = F_{q}^{c}(Z) W(Z)$$

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.

Calculate the percent F<sub>q</sub>(Z) exceeds its limit by the following expression:

$$\left\{ \left( \text{maximum over } z \left[ \frac{F_{q}^{c}(z) \times W(z)}{\frac{F_{q}^{RTP}}{P} \times K(z)} \right] - 1 \right) \right\} \times 100 \text{ for } P \geq 0.5$$

$$\left\{ \left( \text{maximum over } z \left[ \frac{F_{q}^{c}(z) \times W(z)}{\frac{F_{q}^{RTP}}{0.5} \times K(z)} \right] - 1 \right) \right\} \times 100 \text{ for } P < 0.5$$

The F<sub>q</sub>(Z) limits define limiting values for core power peaking that, with a high level of probability, preclude 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. If F<sub>q</sub>(Z) cannot be maintained within the LCO limits, reduction of the core power is required.

Violating the LCO limits for F<sub>q</sub>(Z) may produce unacceptable

(continued)

BASES

---

LCO  
(continued)

consequences if a design basis event occurs while F<sub>o</sub>(Z) is outside its specified limits.

If the power distribution measurements are performed at a power level less than 100% RTP, then the F<sub>o</sub><sup>c</sup>(Z) and F<sub>o</sub><sup>w</sup>(Z) values that would result from measurements if the core was at 100% RTP should be inferred from the available information. A comparison of these inferred values with F<sub>o</sub><sup>RTP</sup> assures compliance with the LCO at all power levels.

APPLICABILITY

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

ACTIONS

A.1

Reducing THERMAL POWER by ≥1% RTP for each 1% by which F<sub>o</sub><sup>c</sup>(Z) exceeds its limit, maintains an acceptable absolute power density. F<sub>o</sub><sup>c</sup>(Z) is F<sub>o</sub><sup>m</sup>(Z) multiplied by factors which account for manufacturing tolerances and measurement uncertainties. F<sub>o</sub><sup>m</sup>(Z) is the measured value of F<sub>o</sub>(Z). The Completion Time of 15 minutes provides an acceptable time to reduce power in an orderly manner and without allowing the plant to remain in an unacceptable condition for an extended period of time.

The maximum allowable power level initially determined by Required Action A.1 may be affected by subsequent determinations of F<sub>o</sub><sup>c</sup>(Z) and would require power reductions within the 15 minutes of the F<sub>o</sub><sup>c</sup>(Z) determination, if necessary to comply with the decreased maximum allowable power level. Decreases in F<sub>o</sub><sup>c</sup>(Z) would allow increasing the maximum allowable power level and increasing power up to this revised limit.

A.2

A reduction of the Power Range Neutron Flux-High trip setpoints by ≥ 1% for each 1% by which F<sub>o</sub><sup>c</sup>(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

(continued)

BASES

---

ACTIONS

A.2 (continued)

time period and the preceding prompt reduction in THERMAL POWER in accordance with Required Action A.1. The maximum allowable Power Range Neutron Flux - High trip setpoints initially determined by Required Action A.3 may be affected by subsequent determinations of F<sub>o</sub><sup>c</sup>(Z) and would require Power Range Neutron Flux - High trip setpoint reductions within 72 hours of the F<sub>o</sub><sup>c</sup>(Z) determination, if necessary to comply with the decreased maximum allowable Power Range Neutron Flux - High trip setpoints. Decreases in F<sub>o</sub><sup>c</sup>(Z) would allow increasing the maximum allowable Power Range Neutron Flux - High trip setpoints.

A.3

Reduction in the Overpower ΔT trip setpoints by ≥ 1% for each 1% by which F<sub>o</sub><sup>c</sup>(Z) exceeds its limit, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 72 hours is sufficient considering the small likelihood of a severe transient in this time period, and the preceding prompt reduction in THERMAL POWER in accordance with Required Action A.1. The maximum allowable Overpower ΔT setpoints initially determined by Required Action A.4 may be affected by subsequent determinations of F<sub>o</sub><sup>c</sup>(Z) and would require Overpower ΔT setpoint reductions within 72 hours of the F<sub>o</sub><sup>c</sup>(Z) determination, if necessary to comply with the decreased maximum allowable Overpower ΔT trip setpoints. Decreases in F<sub>o</sub><sup>c</sup>(Z) would allow increasing the maximum allowable Overpower ΔT trip setpoints.

A.4

Verification that F<sub>o</sub><sup>c</sup>(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, ensures that core conditions during operation at higher power levels are consistent with safety analyses assumptions. Inherent in this action is identification of the cause of the out of limit condition, and the correction of the cause, to the extent necessary to allow safe operation at the higher power level. The allowable power level is determined by extrapolating F<sub>o</sub><sup>c</sup>(Z). SR 3.2.1.1 must be satisfied prior to increasing power above the extrapolated allowable power level or restoration of any reduced Reactor Trip System setpoints.

---

(continued )

BASES

---

ACTION  
(continued)

B.1

If it is found that the maximum calculated value of F<sub>o</sub>(Z) that can occur during normal maneuvers, F<sub>o</sub><sup>\*</sup>(Z), exceeds its specified limits, there exists a potential for F<sub>o</sub><sup>c</sup>(Z) to become excessively high if a normal operational transient occurs. Reducing both the positive and negative AFD limits by ≥ 1% for each 1% by which F<sub>o</sub><sup>\*</sup>(Z) exceeds its limit within the allowed Completion Time of 4 hours, restricts the axial flux distribution such that even if a transient occurred, core peaking factor limits are not exceeded.

C.1

If Required Actions A.1 through A.4 or B.1 are not met within their associated Completion Times, the plant must be placed in a mode or condition in which the LCO requirements are not applicable. This is done by placing the plant in at least MODE 2 within 6 hours.

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

---

SURVEILLANCE  
REQUIREMENTS

SR 3.2.1.1 and SR 3.2.1.2 are modified by a Note. The Note applies during power ascensions following a plant shutdown (leaving MODE 1). The note allows for power ascensions if the surveillances are not current. It states that THERMAL POWER may be increased until an equilibrium power level has been achieved at which a power distribution map can be obtained. This allowance is modified, however, by one of the Frequency conditions that requires verification that F<sub>o</sub><sup>c</sup>(Z) and F<sub>o</sub><sup>\*</sup>(Z) are within their specified limits after a power rise of more than 20% RTP over the THERMAL POWER at which they were last verified to be within specified limits. Because F<sub>o</sub><sup>c</sup>(Z) and F<sub>o</sub><sup>\*</sup>(Z) could not have previously been measured for a reload core, there is a second Frequency condition, applicable only for reload cores, that requires determination of these parameters before exceeding 75% RTP. This ensures that some determination of F<sub>o</sub><sup>c</sup>(Z) and F<sub>o</sub><sup>\*</sup>(Z) are made at a lower power level at which adequate margin is available before going to 100% RTP. Also, this Frequency condition, together with the Frequency condition requiring verification of F<sub>o</sub><sup>c</sup>(Z) and F<sub>o</sub><sup>\*</sup>(Z) following a power increase of more than 20%, ensures that they are verified within 24 hours from when equilibrium conditions are achieved

(continued)

---

BASES

SURVEILLANCE  
REQUIREMENTS  
(continued)

at RTP (or any other level for extended operation). Equilibrium conditions are achieved when the core is sufficiently stable such that the uncertainties associated with the measurement are valid. In the absence of these Frequency conditions, it is possible to increase power to RTP and operate for 31 days without verification of  $F_0^c(Z)$  and  $F_0^m(Z)$ . The Frequency condition is not intended to require verification of these parameters after every 20% increase in power level above the last verification. It only requires verification after a power level is achieved for extended operation that is 20% higher than that power at which  $F_0(Z)$  was last measured.

SR 3.2.1.1

Verification that  $F_0^c(Z)$  is within its specified limits involves increasing  $F_0^m(Z)$  to allow for manufacturing tolerance and measurement uncertainties in order to obtain  $F_0^c(Z)$ . Specifically,  $F_0^m(Z)$  is the measured value of  $F_0(Z)$  obtained from incore flux map results and  $F_0^c(Z) = F_0^m(Z) (1.03) (1.05)$  (Ref. 2).  $F_0^c(Z)$  is then compared to its specified limits.

The limit with which  $F_0^c(Z)$  is compared varies inversely with power above 50% RTP and directly with a function called  $K(Z)$  provided in the COLR.

Performing this Surveillance in MODE 1 prior to exceeding 75% RTP (and meeting the 100% RTP  $F_0(Z)$  limit) provides assurance that the  $F_0^c(Z)$  limit is met when RTP is achieved, because peaking factors generally decrease as power level is increased.

If THERMAL POWER has been increased by  $\geq 20\%$  RTP since the last determination of  $F_0^c(Z)$ , another evaluation of this factor is required 24 hours after achieving equilibrium conditions at this higher power level to ensure that  $F_0^c(Z)$  values are being reduced sufficiently with power increase to stay within the LCO limits.

The Frequency of 31 EFPD is adequate to monitor the change of power distribution with core burnup because such changes are slow and well controlled when the plant is operated in accordance with the Technical Specifications (TS).

(continued)

BASES

---

SURVEILLANCE  
REQUIREMENTS  
(continued)

SR 3.2.1.2

Because flux maps are taken in equilibrium conditions, the variations in power distribution resulting from normal operational maneuvers are not present in the flux map data. These variations are, however, conservatively calculated by considering a wide range of unit maneuvers in normal operation. The maximum peaking factor increase over steady state values, calculated as a function of core elevation, Z, is called W(Z). Multiplying the measured total peaking factor, F<sub>q</sub><sup>c</sup>(Z), by W(Z) gives the maximum F<sub>q</sub>(Z) calculated to occur in normal operation, F<sub>q</sub><sup>w</sup>(Z).

The limit with which F<sub>q</sub><sup>w</sup>(Z) is compared varies inversely with power and directly with the function K(Z) provided in the COLR.

The W(Z) curve is provided in the COLR for discrete core elevations. Flux map data are typically taken for 30 to 75 core elevations. F<sub>q</sub><sup>w</sup>(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 a Note that may require that more frequent surveillances be performed. When F<sub>q</sub><sup>w</sup>(Z) is determined, an evaluation of the expression below is required to account for any increase to F<sub>q</sub><sup>c</sup>(Z) that may occur and cause the F<sub>q</sub>(Z) limit to be exceeded before the next required F<sub>q</sub>(Z) evaluation.

If the two most recent F<sub>q</sub>(Z) evaluations show an increase in the expression

$$\text{maximum over } z \left[ \frac{F_q^c(Z)}{K(Z)} \right]$$

it is required to meet the F<sub>q</sub>(Z) limit with the last F<sub>q</sub><sup>w</sup>(Z) increased by a factor ≥ 2 percent which is specified in the COLR, or to evaluate F<sub>q</sub>(Z) more frequently, each 7 EFPD. These alternative requirements

(continued)

BASES

---

SURVEILLANCE  
REQUIREMENTS

SR 3.2.1.2 (continued)

prevent F<sub>q</sub>(Z) from exceeding its limit for any significant period of time without detection. Performing the Surveillance in MODE 1 prior to exceeding 75% RTP or at a reduced power at any other time, and meeting the 100% RTP F<sub>q</sub>(Z) limit, provides assurance that the F<sub>q</sub>(Z) limit will be met when RTP is achieved, because peaking factors are generally decreased as power level is increased.

F<sub>q</sub>(Z) is verified at power levels  $\geq$  20% RTP above the THERMAL POWER of its last verification, 24 hours after achieving equilibrium conditions to ensure that F<sub>q</sub>(Z) is within its limit at higher power levels.

The Surveillance Frequency of 31 EFPD is normally 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<sub>q</sub>(Z) evaluations.

The Frequency of 31 EFPD is adequate to monitor the change of power distribution because such a change is sufficiently slow, when the plant is operated in accordance with the TS, to preclude adverse peaking factors between 31 day surveillances.

---

REFERENCES

1. 10 CFR 50.46, 1974.
  2. 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 ( $F_{\Delta H}^N$ )

#### 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 ensures that local conditions in the fuel rods and coolant channels do not challenge core integrity at any location during normal operation, operational transients, and any transient condition arising from events of moderate frequency analyzed in the safety analyses.

$F_{\Delta H}^N$  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,  $F_{\Delta H}^N$  is a measure of the maximum total power produced in a fuel rod.  $F_{\Delta H}^N$  is sensitive to fuel loading patterns, bank insertion, and fuel burnup.

$F_{\Delta H}^N$  is not directly measurable but is inferred from a power distribution map obtained with the movable incore detector system. Specifically, the results of the power distribution map are analyzed to determine  $F_{\Delta H}^N$ . This factor is calculated at least every 31 EFPD. However, during power operation, the global power distribution is monitored by LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," which address directly and continuously measured process variables. Compliance with these LCOs, along with the LCOs governing shutdown and control rod insertion and alignment, maintains the core limits on power distribution on a continuous basis.

The 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. All DNB limited transient events are assumed to begin with an  $F_{\Delta H}^N$  value that satisfies the LCO requirements.

---

(continued)

BASES
 

---

 BACKGROUND  
 (continued)

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

---

 APPLICABLE  
 SAFETY  
 ANALYSES

Limits on  $F_{\Delta H}^N$  preclude core power distributions that exceed the following fuel design limits:

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

The limits on  $F_{\Delta H}^N$  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 applicable to a specific DNBR correlation. This value provides a high degree of assurance that the hottest fuel rod in the core does not experience a DNB condition.

The allowable  $F_{\Delta H}^N$  limit increases with decreasing power level. This relationship between power and  $F_{\Delta H}^N$  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  $F_{\Delta H}^N$  in the analysis. Likewise, all transients that may be DNB limited are assumed to begin with a limiting initial  $F_{\Delta H}^N$  as a function of power level defined by the  $F_{\Delta H}^N$  limit equation in the COLR.

---

 (continued)

BASES

<p>APPLICABLE SAFETY ANALYSES (continued)</p>	<p>The LOCA safety analysis also uses <math>F_{\Delta H}^N</math> as an input parameter. The Nuclear Heat Flux Hot Channel Factor (<math>F_Q(Z)</math>) 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].</p> <p>The fuel is protected in part by compliance with Technical Specifications, which ensure that the initial conditions assumed in the safety and accident analyses remain valid. The following LCOs ensure this: LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," LCO 3.1. 6, "Control Bank Insertion Limits," LCO 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor <math>F_{\Delta H}^N</math>," and LCO 3.2.1, "Heat Flux Hot Channel Factor (<math>F_Q(Z)</math>)."</p> <p><math>F_{\Delta H}^N</math> and <math>F_Q(Z)</math> 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 AFD, QPTR, and Bank Insertion Limits.</p> <p><math>F_{\Delta H}^N</math> satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).</p>
<p>LCO</p>	<p><math>F_{\Delta H}^N</math> shall be maintained within the limits of the relationship provided in the COLR.</p> <p>The <math>F_{\Delta H}^N</math> limit is representative of the coolant flow channel with the maximum enthalpy rise. This channel has the highest probability for a DNB condition.</p> <p>A power multiplication factor in this equation includes an additional allowance for higher radial peaking factors from reduced thermal feedback and greater control rod insertion at low power levels. The limiting value of <math>F_{\Delta H}^N</math> is allowed to increase by a cycle-dependent factor, <math>PF_{\Delta H}</math>, specified in the COLR for reduction in THERMAL POWER.</p> <p>If the power distribution measurements are performed at a power level less than 100% RTP, then the <math>F_{\Delta H}^N</math> values that would result from measurements if the core was at 100% RTP should be inferred from the available information. A comparison of these inferred values with <math>F_{\Delta H}^{NRTP}</math> assures compliance with the LCO at all power levels.</p>

BASES

APPLICABILITY	<p>The <math>F_{\Delta H}^N</math> 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.</p>
ACTIONS	<p><u>A.1.1</u></p> <p>With <math>F_{\Delta H}^N</math> exceeding its limit, the unit is allowed 4 hours to restore <math>F_{\Delta H}^N</math> to within its limits. This restoration may, for example, involve realigning any misaligned rods or reducing power enough to bring <math>F_{\Delta H}^N</math> within its power dependent limit. When the <math>F_{\Delta H}^N</math> limit is exceeded, the DNBR limit is not likely to be violated in steady state operation, because events that could significantly perturb the <math>F_{\Delta H}^N</math> 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 occurs. Thus, the allowed Completion Time of 4 hours provides an acceptable time to restore <math>F_{\Delta H}^N</math> to within its limits without allowing the plant to remain in an unacceptable condition for an extended period of time. The restoration of the peaking factor to within its limits by power reduction or control rod movement does not restore compliance with the LCO. Thus, this condition can not be exited until a valid surveillance demonstrates compliance with the LCO.</p> <p>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 is not reduced because this Required Action is completed within the 4 hour time period, Required Action A.2 nevertheless requires another measurement and calculation of <math>F_{\Delta H}^N</math> within 24 hours in accordance with SR 3.2.2.1. However, if power is reduced below 50% RTP, Required Action A.3 requires that another determination of <math>F_{\Delta H}^N</math> must be done prior to exceeding 50% RTP, prior to exceeding 75% RTP, and within 24 hours after reaching or exceeding 95% RTP; however, THERMAL POWER does not have to be reduced to comply with these requirements. In addition, Required Action A.2 is performed if power ascension is delayed past 24 hours.</p>

(continued)

BASES

---

ACTIONS  
(continued)

A.1.2.1 and A.1.2.2

If the value of  $F_{\Delta H}^N$  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% RTP in accordance with Required Action A.1.2.1 and reduce the Power Range Neutron Flux—High to  $\leq 55\%$  RTP in accordance with Required Action A.1.2.2. Reducing THERMAL POWER to < 50% RTP increases the DNB margin and does not likely cause the DNBR limit to be violated in steady state operation. The reduction in trip setpoints ensures that continuing operation remains at an acceptable low power level with adequate DNBR margin. The allowed Completion Time of 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. The Completion Times of 4 hours for Required Actions A.1.1 and A.1.2.1 are not additive.

The allowed Completion Time of 72 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; however, for extended operations at the reduced power level, the reduced trip setpoints are required to protect against events involving positive reactivity excursions. This is a sensitive operation that may inadvertently actuate the Reactor Protection System.

A.2

Once actions have been taken to restore  $F_{\Delta H}^N$  to within its limits per Required Action A.1.1, or the power level has been reduced to < 50% RTP per Required Action A.1.2.1, an incore flux map (SR 3.2.2.1) must be obtained and the measured value of  $F_{\Delta H}^N$  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  $F_{\Delta H}^N$ .

(continued)

BASES

---

ACTIONS  
(continued)

A.3

Verification that  $F_{\Delta H}^N$  is within its specified limits after an out of limit occurrence ensures that the cause that led to exceeding the  $F_{\Delta H}^N$  limit is identified, to the extent necessary, and corrected, and that subsequent operation proceeds within the LCO limit. This Action demonstrates that the  $F_{\Delta H}^N$  limit is within the LCO limits prior to exceeding 50% RTP, again prior to exceeding 75% RTP, and within 24 hours after THERMAL POWER is  $\geq 95\%$  RTP. SR 3.2.2.1 must be satisfied prior to increasing power above the extrapolated allowable power level or restoration of any reduced Reactor Trip System setpoints. When  $F_{\Delta H}^N$  is measured at reduced power levels, the allowable power level is determined by evaluating  $F_{\Delta H}^N$  for higher power levels.

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

B.1

When Required Actions A.1.1 through A.3 cannot be completed within their required Completion Times, the plant must be placed in a mode in which the LCO requirements are not applicable. This is done by placing the plant in at least MODE 2 within 6 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience regarding the time required to reach MODE 2 from full power conditions in an orderly manner and without challenging plant systems.

---

SURVEILLANCE  
REQUIREMENTS

SR 3.2.2.1

SR 3.2.2.1 is modified by a Note. The Note applies during power ascensions following a plant shutdown (leaving MODE 1). The Note allows for power ascensions if the surveillances are not current. It states that THERMAL POWER may be increased until an equilibrium power level has been achieved at which a power distribution map can be obtained. Equilibrium conditions are achieved when the core is sufficiently stable such that uncertainties associated with the measurement are valid.

The value of  $F_{\Delta H}^N$  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  $F_{\Delta H}^N$  from the measured flux distributions. The limit of  $F_{\Delta H}^N$  in the COLR allows for 4% measurement uncertainties.

(continued)

BASES

---

SURVEILLANCE  
REQUIREMENTS

SR 3.2.2.1 (continued)

After each refueling,  $F_{\Delta H}^N$  must be determined in MODE 1 prior to exceeding 75% RTP. This requirement ensures that  $F_{\Delta H}^N$  limits are met at the beginning of each fuel cycle. Performing this Surveillance in MODE 1 prior to exceeding 75% RTP, or at a reduced power level at any other time, and meeting the 100% RTP  $F_{\Delta H}^N$  limit, provides assurance that the  $F_{\Delta H}^N$  limit is met when RTP is achieved, because peaking factors generally decrease as power level is increased.

The 31 EFPD Frequency is acceptable because the power distribution changes relatively slowly over this amount of fuel burnup. Accordingly, this Frequency is short enough that the  $F_{\Delta H}^N$  limit cannot be exceeded for any significant period of operation.

---

REFERENCES

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

## B 3.2 POWER DISTRIBUTION LIMITS

### B 3.2.3 Axial Flux Difference (AFD)

#### BASES

---

##### BACKGROUND

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

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

The AFD is monitored on an automatic basis using the plant 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 if the AFD for two or more OPERABLE excore channels is outside its specified limits.

Although the RAOC defines limits that must be met to satisfy safety analyses, typically an operating scheme, is used to control axial power distribution in day to day operation (Ref. 1). This requires that the AFD be controlled within a band around a burnup dependent target to minimize the variation of axial peaking factors and axial xenon distribution during unit maneuvers.

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

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 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 tentatively wide AFD limits. One dimensional axial power distribution calculations are then performed to demonstrate that normal operation power shapes are acceptable for the LOCA and 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 ensure 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 that are used as initial conditions in the analyses of Condition II, III, or IV events. Compliance with these limits ensures that the fuel cladding integrity is maintained for these postulated accidents. The most important Condition IV event is the LOCA. The most important Condition III event is the complete loss of forced RCS flow accident. The most important Condition II events are uncontrolled bank withdrawal and boration or dilution accidents. Condition II accidents are used to confirm the adequacy of the Overpower  $\Delta T$  and Overttemperature  $\Delta T$  trip setpoints.

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

LCO

The shape of the power profile in the axial (i.e., the vertical) direction is largely under the control of the operator through the manual operation of the control banks or automatic motion of control banks. The automatic motion of the control banks is in response to temperature deviations resulting from manual operation of the Chemical and Volume Control System to change boron concentration or from turbine load 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 power range detectors. The AFD is defined as the difference in normalized flux signals between the top

(continued)

BASES

LCO (continued)	<p>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 % <math>\Delta</math> flux or % <math>\Delta</math>I.</p> <p>The AFD limits are provided in the COLR. The AFD limits for RAOC do not depend on the target flux difference. However, the target AFD may be used to minimize changes in the axial power distribution.</p> <p>Violating this LCO on the AFD could produce unacceptable consequences if a Condition II, III, or IV event occurs while the AFD is outside its specified limits.</p>
APPLICABILITY	<p>The AFD requirements are applicable in MODE 1 greater than or equal to 50% RTP when the combination of THERMAL POWER and core peaking factors are of primary importance in safety analysis.</p> <p>For AFD limits developed using RAOC methodology, the value of the AFD does not affect the limiting accident consequences with THERMAL POWER &lt; 50% RTP and for lower operating power MODES.</p>
ACTIONS	<p><u>A.1</u></p> <p>As an alternative to restoring the AFD to within its specified limits, Required Action A.1 requires a THERMAL POWER reduction to &lt; 50% RTP. This places the core in a condition for which the value of the AFD is not important in the applicable safety analyses. A Completion Time of 30 minutes is reasonable, based on operating experience, to reach 50% RTP without challenging plant systems.</p>
SURVEILLANCE REQUIREMENTS	<p><u>SR 3.2.3.1</u></p> <p>This Surveillance verifies that the AFD, as indicated by each OPERABLE NIS excore channel, is within its specified limits. The Surveillance Frequency of 7 days is adequate considering that the AFD is monitored by a computer and any deviation from requirements is alarmed.</p>
REFERENCES	<ol style="list-style-type: none"> <li>1. WCAP-8403 (nonproprietary), "Power Distribution Control and Load Following Procedures," Westinghouse Electric Corporation, September 1974.</li> <li>2. WCAP-10216-P-A, Revision 1A, Relaxation of Constant Axial Offset Control, F<sub>o</sub> Surveillance Technical Specification, February 1994 (Westinghouse Proprietary).</li> <li>3. FSAR, Chapter 4.3.2.2.4.</li> </ol>

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 Quadrant Power Tilt Ratio (QPTR)

BASES

---

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

The power density at any point in the core must be limited so that the fuel design criteria are maintained. Together, LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.2.4, and LCO 3.1. 6, "Control 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 that violate the following fuel design criteria:

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

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

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

(continued)

BASES

<p>APPLICABLE SAFETY ANALYSES (continued)</p>	<p>In MODE 1, the <math>F_{\Delta H}^N</math> and <math>F_{\alpha}(Z)</math> limits must be maintained to preclude core power distributions from exceeding design limits assumed in the safety analyses.</p> <p>The QPTR satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).</p>
<p>LCO</p>	<p>The QPTR limit of 1.02, above which corrective action is required, provides a margin of protection for both the DNB ratio and linear heat generation rate contributing to excessive power peaks resulting from X-Y plane power tilts. A limiting QPTR of 1.02 can be tolerated before the margin for uncertainty in <math>F_{\alpha}(Z)</math> and <math>(F_{\Delta H}^N)</math> is possibly challenged.</p>
<p>APPLICABILITY</p>	<p>The QPTR limit must be maintained in MODE 1 with THERMAL POWER &gt; 50% RTP to prevent core power distributions from exceeding the design limits.</p> <p>Applicability in MODE 1 <math>\leq</math> 50% RTP and in other MODES is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the reactor coolant to require the implementation of a QPTR limit on the distribution of core power. The QPTR limit in these conditions is, therefore, not important. Note that the <math>F_{\Delta H}^N</math> and <math>F_{\alpha}(Z)</math> LCOs still apply, but allow progressively higher peaking factors at 50% RTP or lower.</p>
<p>ACTIONS</p>	<p><u>A.1</u></p> <p>With the QPTR exceeding its limit, a power level reduction of 3% 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 a power reduction may cause a change in the tilted condition.</p> <p>The maximum allowable THERMAL POWER level initially determined by Required Action A.1 may be affected by subsequent determinations of QPTR. Increases in QPTR would require a THERMAL POWER reduction within 2 hours of QPTR determination, if necessary to comply with the decreased maximum allowable THERMAL POWER level. Decreases in QPTR would allow raising the maximum allowable THERMAL POWER level and increasing THERMAL POWER up to this revised limit.</p> <p><u>A.2</u></p> <p>After completion of Required Action A.1, the QPTR may still exceed its limits. Any additional changes in the QPTR are detected by requiring a check of the QPTR once per 12 hours thereafter. If the QPTR</p>

(continued)

BASES

---

ACTIONS

A.2 (continued)

continues to increase, THERMAL POWER has to be reduced accordingly. A 12 hour Completion Time is sufficient because any additional change in QPTR would be relatively slow.

A.3

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

A.4

Although  $F_{\Delta H}^N$  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 that requires an investigation and evaluation that is accomplished by examining the incore power distribution. Specifically, the core peaking factors and the incore quadrant tilt must be evaluated because they are the factors that best characterize the core power distribution. This

(continued)

BASES

ACTIONS

A.4 (continued)

evaluation is required to ensure that, before increasing THERMAL POWER to above the limit of Required Action A.1, the reactor core conditions are consistent with the assumptions in the safety analyses.

A.5

If the QPTR remains above the 1.02 limit and a evaluation of the safety analysis is completed and shows that safety requirements are met, the excore detectors are normalized to restore QPTR to within limit prior to increasing THERMAL POWER to above the limit of Required Action A.1. This is done to detect any subsequent significant changes in QPTR.

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

A.6

Once the excore detectors are normalized to restore QPTR to within limit (i.e., Required Action A.5 is performed), it is acceptable to return to full power operation. However, as an added check that the core power distribution at RTP is consistent with the safety analysis assumptions, Required Action A.6 requires verification that  $F_q(Z)$  and  $F_{\Delta H}^N$  are within their specified limits within 24 hours of achieving equilibrium conditions. Equilibrium conditions are achieved when the core is sufficiently stable at the intended operating conditions to support flux mapping. As an added precaution, if the peaking factor verification cannot be performed within 24 hours due to non-equilibrium core conditions, a maximum time of 48 hours is allowed for the completion of the verification.

This Completion Time is intended to allow adequate time to increase THERMAL POWER to above the limit of Required Action A.1, while not permitting the core to remain with unconfirmed power distributions for extended periods of time.

(continued)

BASES

ACTIONS

A.6 (continued)

Required Action A.6 is modified by a Note that states that the peaking factor surveillances must be completed when the excore detectors have been normalized to restore QPTR to within limit (i.e., Required Action A.5). The intent of this Note is to have the peaking factor surveillances performed at operating power levels, which are only required if the excore detectors were normalized to restore QPTR to within limit per Required Action A.5.

B.1

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

SURVEILLANCE  
REQUIREMENTS

SR 3.2.4.1

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

Input from a Power Range Neutron Flux channel is considered to be operable if the upper and lower detector currents are obtainable. The remaining portion of the channel (the electronics required to provide the channel input to the QPTR alarm) need not be operable.

This Surveillance verifies that the QPTR, as indicated by the Nuclear Instrumentation System (NIS) excore channels, is within its limits. The Frequency of 7 days takes into account other information and alarms available to the operator in the control room.

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

SR 3.2.4.2

This Surveillance is modified by a Note, which states that it is not required until 12 hours after the input from one or more Power Range Neutron Flux channels is inoperable and the THERMAL POWER is  $> 75\%$  RTP.

(continued)

BASES

---

SURVEILLANCE  
REQUIREMENTS

SR 3.2.4.2 (continued)

With an NIS power range channel inoperable, tilt monitoring for a portion of the reactor core becomes degraded. Large tilts are likely detected with the remaining channels, but the capability for detection of small power tilts in some quadrants is decreased. Performing SR 3.2.4.2 at a Frequency of 12 hours provides an accurate alternative means for ensuring that any tilt remains within its limits.

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

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

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

---

REFERENCES

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

## B 3.3 INSTRUMENTATION

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

#### BASES

---

##### BACKGROUND

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

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

The LSSS, as defined in 10 CFR 50.36, are defined in this specification as the Allowable Values, and 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 more than once during the unit life, the acceptable limits are:

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

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

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

(continued)

---

BASES

---

BACKGROUND  
(continued)

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 Digital 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 unit 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 (CRDMs) and allows the rod cluster control assemblies (RCCAs), or "rods," to fall into the core and shut down the reactor. The bypass breakers allow testing of the RTBs at power.

Field Transmitters or Sensors

To meet the design demands for redundancy and reliability, more than one, and often as many as four, field transmitters or sensors are used to measure unit parameters. To account for the calibration tolerances and instrument drift, which are assumed to occur between calibrations, statistical allowances are provided in the Trip Setpoint and 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 unit 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

(continued)

BASES

---

BACKGROUND

Signal Process Control and Protection System (continued)

in the FSAR (References 1, 2, 3, 9, 10, & 11). 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, except in the case of the seismic, turbine stop valve position, auto stop oil pressure, 12 kV bus and RCP breaker inputs which do not go through signal conditioning. Channel separation is maintained up to and through the input bays. However, not all unit parameters require four channels of sensor measurement and signal processing. Some unit parameters provide input only to the SSPS, while others provide input to the SSPS, the main control board, the unit computer, and one or more control systems.

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

Generally, if a parameter is used for input to the SSPS and a control function, four channels with a two-out-of-four logic are sufficient to provide the required reliability and redundancy. In the case of the Digital Feedwater Control System (DFWCS), the median/signal select (MSS) feature prevents control/protection interaction even though there are only three inputs and 2-out-of-3 logic. The circuit must be able to withstand both an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation. Again, a single failure will neither cause nor prevent the protection function actuation. These requirements are described in IEEE-279-1971 (Ref. 4). The actual number of channels required for each unit parameter is specified in Reference 1.

Two logic channels are required to ensure no single random failure of a logic channel will disable the RTS. The logic channels are designed such that testing required while the reactor is at power may be accomplished without causing a trip. The process Protection System is designed to permit any one channel to be tested and maintained at power in a bypass mode. If a channel has been bypassed for any purpose, the bypass is continuously indicated in the control room as required by applicable codes and standards. As an alternative to testing in the bypass mode, testing in the trip mode is also possible and permitted.

(continued)

BASES

BACKGROUND  
(continued)

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 two sided tolerance band for CHANNEL CALIBRATION tolerance. The calibration tolerance, after conversion, should correspond to the rack comparator setting accuracy defined in the latest setpoint study.

The Trip Setpoints used in the bistables are based on the analytical limits stated in Reference 1. The selection of these Trip Setpoints is such that adequate protection is provided when all sensor and processing time delays are taken into account. To allow for calibration tolerances, instrumentation uncertainties, instrument drift, and severe environment errors for those RTS channels that must function in harsh environments as defined by 10 CFR 50.49 (Ref. 5), the Trip Setpoints and Allowable Values specified in Table 3.3.1-1 in the accompanying LCO 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 WCAP-11082, Rev. 5, "Westinghouse Setpoint Methodology for Protection Systems Diablo Canyon Units 1 & 2, 24 Month Fuel Cycle Evaluation," January 1997 (Ref. 17) and calculation NSP-1-20-13F Rev 1 (Ref. 18) and NSP-2-20-13F Rev 1 (Ref. 19). Interlock setpoints are Nominal Values provided in the PLS (Westinghouse Precautions Limitations and Setpoints) and their allowable values are calculated in Calculation J-110 Rev 5 (Ref. 20). The actual nominal Trip Setpoint entered into the bistable is more conservative than that specified by the Allowable Value to account for Rack Drift and Rack Measuring and Test Equipment uncertainties. 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.

Rack drift in excess of the Allowable Value exhibits the behavior that the rack has not met its allowance. Since there is a small statistical chance that this will happen, an infrequent excessive drift is expected. Rack or sensor drift in excess of the allowance that is more than occasional may be indicative of more serious problems and warrants further investigation. In the event a channel's setpoint is found nonconservative with respect to the specified Trip Setpoint, but more conservative than the Allowable Value, the setpoint must be adjusted consistent with the Trip Setpoint value. When a channel's Trip Setpoint is nonconservative with respect to the Allowable Value, declare the channel inoperable and apply the applicable ACTION statement until the channel is returned to OPERABLE status with its Setpoint adjusted consistent with the Trip Setpoint value.

(continued)

BASES

BACKGROUND

Trip Setpoints and Allowable Values (continued)

Setpoints in accordance with the Allowable Value ensure that SLs are not violated during AOOs (and that the consequences of DBAs will be acceptable, providing the unit is operated from within the LCOs at the onset of the AOO or DBA and the equipment functions as designed). Note that in the accompanying LCO 3.3.1, the Allowable Values of Table 3.3.1-1 are the LSSS as defined in 10 CFR 50.36.

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, or in the case of the Power Range channels the test signal is added to 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 Trip Setpoints and Allowable Values listed in Table 3.3.1-1 are based on the methodology described in Reference 17, 18, 19, and 20, 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. The inequality sign only indicates conservative direction. The as-left value will be within a two-sided calibration tolerance band on either side of the nominal value. This also applies to the Overtemperature  $\Delta T$  and Overpower  $\Delta T K$  values per reference 16. All field sensors and signal processing equipment for these channels are assumed to operate within the allowances of these uncertainty magnitudes.

Trip Setpoints may be administratively redefined in the conservative direction for several reasons including startup, testing, process error accountability, or even a conservative response for equipment malfunction or inoperability. Some trip functions have historically been redefined at the beginning of each cycle for purposes of startup testing, e.g. Power Range Neutron Flux High and Overtemperature  $\Delta T$ . Calibration to within the defined calibration tolerance of an administratively redefined, conservative Trip Setpoint is acceptable. Redefinition at full power conditions for these functions is expected and acceptable.

Solid State Protection System

The SSPS equipment is used for the decision logic processing of outputs from the signal processing equipment bistables. To meet the redundancy requirements, two trains of SSPS, each performing the same functions, are provided. If one train is taken out of service for maintenance or test purposes, the second train will provide reactor trip and/or ESF actuation for the unit. If both trains are taken out of service

(continued)

BASES

BACKGROUND

Solid State Protection System (continued)

or placed in test, a reactor trip will result. Each train is packaged in its own cabinet for physical and electrical separation to satisfy separation and independence requirements. The system has been designed to trip in the event of a loss of power, directing the unit to a safe shutdown condition.

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

The bistable outputs from the signal processing equipment or relay contact input (RCP breaker, 12kV UV/UF, seismic, etc.) are sensed by the SSPS equipment and combined into logic matrices that represent combinations indicative of various unit transients. If a required logic matrix combination is completed, the system will initiate a reactor trip or send actuation signals via master and slave relays to those components whose aggregate Function best serves to alleviate the condition and restore the unit to a safe condition. Examples are given in the Applicable Safety Analyses, LCO, and Applicability sections of this Bases.

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 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 unit is at power. During normal operation the output from the SSPS is a voltage signal that 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 RTBs 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 reactor trip breaker is also equipped with an automatic shunt trip device that is energized to trip the breaker open upon receipt of a reactor trip signal from the SSPS. Either the undervoltage coil or the shunt trip mechanism is sufficient by itself, thus providing a diverse trip mechanism.

(continued)

BASES

BACKGROUND	<p><u>Reactor Trip Switchgear</u> (continued)</p> <p>The decision logic matrix Functions are described in the functional diagrams included in Reference 1. In addition to the reactor trip or ESF, these diagrams also describe the various "permissive interlocks" that are associated with unit conditions. Each train has a built in testing device that can automatically test the decision logic matrix Functions 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.</p>
<p>APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY</p>	<p>The RTS functions to maintain the applicable limits during all AOOs and mitigates the consequences of DBAs in all MODES in which the Rod Control System is capable of rod withdrawal or one or more rods are not fully inserted.</p> <p>Each of the analyzed accidents and transients can be detected by one or 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 are qualitatively credited in the safety analysis and the NRC staff approved licensing basis for the unit. These RTS trip Functions may provide protection for conditions that do not require dynamic transient analysis to demonstrate Function performance. They may also serve as backups to RTS trip Functions that were credited in the accident analysis.</p> <p>The LCO requires all instrumentation performing an RTS Function, listed in Table 3.3.1-1 in the accompanying LCO, to be OPERABLE. Failure of any instrument renders the affected channel(s) inoperable and reduces the reliability of the affected Functions.</p> <p>The LCO generally requires OPERABILITY of four or three channels in each instrumentation Function, two channels of Manual Reactor Trip in each logic Function, and two trains in each Automatic Trip Logic Function. Generally 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. In the case of the Digital Feedwater Control System, the Median Signal Select feature prevents control/protection interaction even though there are only three inputs and a 2-out-of-3 logic. 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</p>

(continued)

BASES

APPLICABLE  
SAFETY  
ANALYSES  
LCO, and  
APPLICABILITY  
(continued)

two-out-of-three configuration are generally required when there is no potential for control system and protection system interaction that could simultaneously create a need for RTS trip and disable one RTS channel. The two-out-of-three and two-out-of-four configurations allow one, channel to be tripped during maintenance or testing without causing a reactor trip. Specific exceptions to the above general philosophy exist and are discussed below.

Reactor Trip System Functions

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

1. Manual Reactor Trip

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

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

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

(continued)

BASES

APPLICABLE  
SAFETY  
ANALYSES,  
LCO, and  
APPLICABILITY  
(continued)

2. Power Range Neutron Flux

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

a. Power Range Neutron Flux—High

The Power Range Neutron Flux-High trip Function ensures that protection is provided, from all power levels, against a positive reactivity excursion leading to fuel damage. Reactivity excursions can be caused by rod withdrawal or inadvertent CVCS malfunction, or for example, by sudden changes in RCS coolant temperature such as a feedwater system malfunction (Ref. 12).

The LCO requires all four of the Power Range Neutron Flux—High channels to be OPERABLE (2-out-of-4 coincidence).

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

b. Power Range Neutron Flux—Low

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

(continued)

BASES

---

APPLICABLE  
SAFETY  
ANALYSES,  
LCO, and  
APPLICABILITY

b. Power Range Neutron Flux—Low (continued)

The LCO requires all four of the Power Range Neutron Flux—Low channels to be OPERABLE (2-out-of-4 coincidence).

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

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

3. Power Range Neutron Flux Rate

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

a. Power Range Neutron Flux—High Positive Rate

The Power Range Neutron Flux—High Positive Rate trip Function ensures that protection is provided against rapid increases in neutron flux that are characteristic of an RCCA drive rod housing rupture and the accompanying ejection of the RCCA. This Function complements 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.

The LCO requires all four of the Power Range Neutron Flux—High Positive Rate channels to be OPERABLE (2-out-of-4 coincidence).

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

(continued)

BASES

APPLICABLE  
SAFETY  
ANALYSES  
LCO, and  
APPLICABILITY

a. Power Range Neutron Flux—High Positive Rate (continued)

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 SDM in the event of an REA. In MODE 6, no rods are withdrawn and the SDM is increased during refueling operations. The reactor vessel head is also removed or the closure bolts are detensioned preventing any pressure buildup. In addition, the NIS power range detectors cannot detect neutron levels present in this mode.

b. Power Range Neutron Flux—High Negative Rate

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

The LCO requires all four Power Range Neutron Flux—High Negative Rate channels to be OPERABLE (2-out-of-4 coincidence).

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

(continued)

BASES

---

APPLICABLE  
SAFETY  
ANALYSES,  
LCO, and  
APPLICABILITY  
(continued)

4. Intermediate Range Neutron Flux

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

The LCO requires two channels of Intermediate Range Neutron Flux to be OPERABLE (1-out-of-2 coincidence). Two OPERABLE channels are sufficient to ensure no single random failure will disable this trip Function.

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

In MODE 1 below the P-10 setpoint, and in MODE 2 above the P-6 setpoint, when there is a potential for an uncontrolled RCCA bank rod withdrawal accident during reactor startup, the Intermediate Range Neutron Flux trip must be OPERABLE. Above the P-10 setpoint, the Power Range Neutron Flux—High Setpoint trip and the Power Range Neutron Flux—High Positive Rate trip provide core protection for a rod withdrawal accident. In MODE 2 below the P-6 setpoint, the Intermediate Range Neutron Flux trip is not required because the Source Range Neutron Flux trip function provides core protection for reactivity accidents. Although this mode applicability is different from the CTS, where the Applicability was Mode 1 below P-10 and Mode 2, no changes to plant operations or procedures are expected. Since the Intermediate Range channels must be OPERABLE prior to entering the new Applicability per LCO 3.0.4, and the time frame for non-applicability is small and more difficult to control administratively than by the current Mode 2 limitation, DCPD will conservatively consider the Applicability as Mode 1 below P-10 and Mode 2. 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.

(continued)

BASES

APPLICABLE  
SAFETY  
ANALYSES,  
LCO and  
APPLICABILITY  
(continued)

4. Intermediate Range Neutron Flux (continued)

The reactor cannot be started up in this condition. The core also has the required SDM 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 SDM. Also, the NIS intermediate range detectors cannot detect neutron levels present in this MODE.

5. Source Range Neutron Flux

The LCO requirement for the Source Range Neutron Flux trip Function ensures that protection is provided against an uncontrolled RCCA bank rod withdrawal accident from a subcritical condition during startup. This trip Function provides redundant protection to the Power Range Neutron Flux—Low trip Function. In MODES 3, 4, and 5, administrative controls also prevent the uncontrolled withdrawal of rods. The NIS source range detectors are located external to the reactor vessel and measure neutrons leaking from the core. The NIS source range detectors do not provide any inputs to control systems. The source range trip is the only RTS automatic protection function required in MODES 2 below P-6, 3, 4, and 5 with the Rod Control System capable of rod withdrawal or one or more rods not fully inserted. Therefore, the functional capability at the Trip Setpoint is assumed to be available.

The LCO requires two channels of Source Range Neutron Flux to be OPERABLE. Two OPERABLE channels are sufficient to ensure no single random failure will disable this trip Function. The LCO also requires one channel of the Source Range Neutron Flux to be OPERABLE in MODE 3, 4, or 5 with RTBs open or the control rods incapable of withdrawal. In this case, the source range Function is to provide control room indication. The outputs of the Function to RTS logic are not required OPERABLE in MODE 6 or when the RTBs are open or all rods are fully inserted and the Rod Control System is incapable of withdrawal.

The Source Range Neutron Flux Function provides protection for control rod withdrawal from subcritical, boron dilution and control rod ejection events. The Function also provides neutron flux indication in the control room.

In MODE 2 when below the P-6 setpoint during a reactor startup, the Source Range Neutron Flux trip must be OPERABLE (1-out-of-2 coincidence). 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.

(continued)

BASES

---

APPLICABLE  
SAFETY  
ANALYSES,  
LCO, and  
APPLICABILITY

5. Source Range Neutron Flux (continued)

Above the P-6 setpoint, the NIS source range neutron flux trip may be manually blocked and the high voltage to the detectors may be de-energized. Below the P-6 setpoint, the source range neutron flux trip is automatically reinstated and the high voltage to the detectors is automatically energized. In MODES 3, 4, and 5 with the reactor shut down, but with the Rod Control System capable of rod withdrawal or one or more rods not fully inserted, the Source Range Neutron Flux trip Function must also be OPERABLE (1-out-of-2 coincidence) to provide core protection against a rod withdrawal accident. If the Rod Control 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 an uncontrolled boron dilution. The requirements for the NIS source range detectors in MODE 6 are addressed in LCO 3.9.3, "Nuclear Instrumentation."

6. Overtemperature  $\Delta T$

The Overtemperature  $\Delta T$  trip Function is provided to ensure that the design limit DNBR is met. This trip Function also limits the range over which the Overpower  $\Delta T$  trip Function must provide protection and it protects against vessel exit bulk boiling and ensures that the exit quality is within the limits defined by the DNBR correlation. The inputs to the Overtemperature  $\Delta T$  trip include pressure, coolant temperature, axial power distribution, and reactor power as indicated by loop  $\Delta T$  assuming full reactor coolant flow. Protection from violating the DNBR limit is assured for those transients that are slow with respect to delays from the core to the measurement system. The Overtemperature  $\Delta T$  trip Function uses each loop's  $\Delta T$  as a measure of reactor power and is compared with a setpoint that is automatically varied with the following parameters:

- reactor coolant average temperature—the Trip Setpoint is varied to correct for changes in coolant density and specific heat capacity with changes in coolant temperature;
- pressurizer pressure—the Trip Setpoint is varied to correct for changes in system pressure; and
- axial power distribution— $f(\Delta I)$ , the Trip Setpoint is varied to account for imbalances in the axial power distribution as detected by the NIS upper and lower power range detectors.

(continued)

BASES

---

APPLICABLE  
SAFETY  
ANALYSES,  
LCO, and  
APPLICABILITY

6. Overtemperature  $\Delta T$  (continued)

If axial peaks are greater than the design limit, as indicated by the difference between the upper and lower NIS power range detectors, the Trip Setpoint is reduced in accordance with Note 1 of Table 3.3.1-1.

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

$\Delta T_0$ , as used in the overtemperature and overpower  $\Delta T$  trips, represents the 100 percent RTP value of  $\Delta T$  as measured for each loop. For the initial startup of a refueled core,  $\Delta T_0$  is initially assumed to be the same as the measured  $\Delta T$  value from the previous cycle until  $\Delta T$  is once again measured at full power. Accurate determination of the loop specific  $\Delta T$  values are made quarterly when performing the incore/excore recalibration at steady-state conditions (i.e., power distribution conditions not affected by xenon or other transient conditions). The indicated  $\Delta T$  variation between loops is due to the difference between hot leg temperatures and hot leg temperature measurement biases. The hot leg temperature variance between loops is primarily caused by asymmetrical flow in the upper plenum, and the difference in hot leg temperature measurement biases primarily caused by differences in hot leg temperature streaming error between loops. The loop  $\Delta T$ s change with burn up which result from the change in the hot leg streaming biases as the radial power distribution changes.

The Overtemperature  $\Delta T$  trip Function is calculated for each loop as described in Note 1 of Table 3.3.1-1. Trip occurs if Overtemperature  $\Delta T$  is indicated in two loops. The pressure and temperature signals are used for other control functions; thus the actuation logic must be able to withstand an input failure to the

(continued)

BASES

---

APPLICABLE  
SAFETY  
ANALYSES,  
LCO, and  
APPLICABILITY

6. Overtemperature  $\Delta T$  (continued)

control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation. Note that this Function also provides a signal to generate a turbine runback prior to reaching the Trip Setpoint. A turbine runback will reduce turbine power and reactor power. A reduction in power will normally alleviate the Overtemperature  $\Delta T$  condition and may prevent a reactor trip.

The LCO requires all four channels of the Overtemperature  $\Delta T$  trip Function to be OPERABLE. Note that the Overtemperature  $\Delta T$  Function receives input from channels shared with other RTS Functions.

Failures that affect multiple Functions require entry into the Conditions applicable to all affected Functions. In MODE 1 or 2, the Overtemperature  $\Delta T$  trip must be OPERABLE to prevent DNB (2-out-of-4 coincidence). In MODE 3, 4, 5, or 6, this trip Function does not have to be OPERABLE because the reactor is not operating and there is insufficient heat production to be concerned about DNB.

7. Overpower  $\Delta T$

The Overpower  $\Delta T$  trip Function ensures that protection is provided to ensure the integrity of the fuel (i.e., no fuel pellet melting and less than 1% cladding strain) under all possible overpower conditions for Condition I and II events (Ref. 12). This trip Function also limits the required range of the Overtemperature  $\Delta T$  trip Function and provides a backup to the Power Range Neutron Flux—High Setpoint trip. The Overpower  $\Delta T$  trip Function ensures that the allowable heat generation rate (kW/ft) of the fuel is not exceeded. The Overpower  $\Delta T$  trip also provides protection to mitigate the consequences of small steamline breaks, as reported in WCAP-9226, Ref. 16, and steamline breaks with coincident control rod withdrawal (Ref. 3). It uses the  $\Delta T$  of each loop as a measure of reactor power with a setpoint that is automatically varied with the following parameters:

- reactor coolant average temperature—the Trip Setpoint is varied to correct for changes in coolant density and specific heat capacity with changes in coolant temperature; and
- rate of change of reactor coolant average temperature—including dynamic compensation for the delays between the core and the temperature measurement system.

(continued)

---

BASES

---

APPLICABLE  
SAFETY  
ANALYSES,  
LCO, and  
APPLICABILITY

7. Overpower  $\Delta T$  (continued)

$\Delta T_0$ , as used in the overtemperature and overpower  $\Delta T$  trips, represents the 100 percent RTP value of  $\Delta T$  as measured for each loop. For the initial startup of a refueled core,  $\Delta T_0$  is initially assumed to be the same as the measured  $\Delta T$  value from the previous cycle until  $\Delta T$  is once again measured at full power. Accurate determination of the loop specific  $\Delta T$  values are made quarterly when performing the incore/excore recalibration at steady-state conditions (i.e., power distribution conditions not affected by xenon or other transient conditions). The indicated  $\Delta T$  variation between loops is due to the difference between hot leg temperatures and hot leg temperature measurement biases. The hot leg temperature variance between loops is primarily caused by asymmetrical flow in the upper plenum, and the difference in hot leg temperature measurement biases is primarily caused by differences in hot leg temperature streaming error between loops. The loop  $\Delta T$ s change with burn up which result from the change in the hot leg streaming biases as the radial power distribution changes.

The Overpower  $\Delta T$  trip Function is calculated for each loop as per Note 2 of Table 3.3.1-1. Trip occurs if Overpower  $\Delta T$  is indicated in two loops. The temperature signals are used for other control functions; thus, 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 trip setpoint. A turbine runback will reduce turbine power and reactor power. A reduction in power will normally alleviate the Overpower  $\Delta T$  condition and may prevent a reactor trip.

The LCO requires four channels of the Overpower  $\Delta T$  trip Function to be OPERABLE (2-out-of-4 coincidence). Note that the Overpower  $\Delta T$  trip Function receives input channels shared with other RTS Functions. Failures that affect multiple Functions require entry into the Conditions applicable to all affected Functions.

In MODE 1 or 2, the Overpower  $\Delta T$  trip Function must be OPERABLE. These are the only times that enough heat is generated in the fuel to be concerned about the heat generation

(continued)

---

BASES

---

APPLICABLE  
SAFETY  
ANALYSES,  
LCO, and  
APPLICABILITY  
(continued)

rates and overheating of the fuel. In MODE 3, 4, 5, or 6, this trip Function does not have to be OPERABLE because the reactor is not operating and there is insufficient heat production to be concerned about fuel overheating and fuel damage.

8. Pressurizer Pressure

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

a. Pressurizer Pressure—Low

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

The LCO requires four channels of Pressurizer Pressure—Low to be OPERABLE (2-out-of-4 coincidence).

In MODE 1, when DNB is a major concern, the Pressurizer Pressure—Low trip must be OPERABLE. This trip Function is automatically enabled on increasing power by the P-7 Low Pressure Permissive 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, there is insufficient heat production to be concerned about DNB.

b. Pressurizer Pressure—High

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

The LCO requires four channels of the Pressurizer Pressure—High to be OPERABLE (2-out-of-4 coincidence).

The Pressurizer Pressure—High LSSS is selected to be below the pressurizer safety valve actuation pressure and above the power operated relief valve (PORV) setting. This setting

(continued)

BASES

---

APPLICABLE  
SAFETY  
ANALYSES,  
LCO, and  
APPLICABILITY  
(continued)

b. Pressurizer Pressure—High (Continued)

minimizes challenges to safety valves while avoiding unnecessary reactor trip for those pressure increases that can be controlled by the PORVs.

In MODE 1 or 2, the Pressurizer Pressure—High trip must be OPERABLE to help prevent RCS overpressurization and minimize challenges to the relief and safety valves. In MODE 3, 4, 5, or 6, the Pressurizer Pressure—High trip Function does not have to be OPERABLE because transients that could cause an overpressure condition will usually be slow to occur. Therefore, the operator will have sufficient time to evaluate unit conditions and take corrective actions. Two low temperature overpressure protection system channels provide overpressure protection with the PORVs when below the low temperature cut-off specified in the pressure and temperature limits report (PTLR).

9. Pressurizer Water Level—High

The Pressurizer Water Level—High trip Function provides a backup signal for the Pressurizer Pressure—High trip and also provides protection against water relief through the pressurizer safety valves. These valves are designed to pass steam in order to achieve their design energy removal rate. A reactor trip is actuated prior to the pressurizer becoming water solid. The LCO requires three channels of Pressurizer Water Level—High to be OPERABLE. The pressurizer level channels are used as input to the Pressurizer Level Control System. A fourth channel is not required to address control/protection interaction concerns. The level channels do not actuate the safety valves, and the high pressure reactor trip is set below the safety valve setting. Therefore, with the slow rate of charging available, pressure overshoot due to level channel failure cannot cause the safety valve to lift before reactor high pressure trip.

In MODE 1, when there is a potential for overfilling the pressurizer, the Pressurizer Water Level—High trip must be OPERABLE (2-out-of-3 coincidence). 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 that could raise the pressurizer water level will be slow and the operator will have sufficient time to evaluate unit conditions and take corrective actions.

(continued)

BASES

---

APPLICABLE  
SAFETY  
ANALYSES,  
LCO, and  
APPLICABILITY

10. Reactor Coolant Flow—Low

The Reactor Coolant Flow—Low trip Function ensures that protection is provided against violating the DNBR limit due to low flow in two or more RCS loops while avoiding reactor trips due to normal variations in loop flow.

Each loop has three flow detectors to monitor flow. The flow signals are not used for any control system input.

The LCO requires three Reactor Coolant Flow—Low channels per loop to be OPERABLE (2-out-of-3 coincidence in one loop).

In MODE 1 above the P-7 setpoint and below the P-8 setpoint, the Reactor Coolant Flow—Low trip must be OPERABLE. Below the P-7 setpoint, all reactor trips on low flow are automatically blocked since there is insufficient heat production to be concerned about DNB. 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. Reactor Coolant Pump (RCP) Breaker Position

The RCP Breaker Position 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, 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 Trip Setpoint is reached.

The LCO requires one RCP Breaker Position channel per RCP to be OPERABLE (2-out-of-4 coincidence). One OPERABLE channel is sufficient for this Function because the RCS Flow—Low trip alone provides sufficient protection of unit SLs for loss of flow events. The RCP Breaker Position trip serves only to anticipate the low flow trip, minimizing the thermal transient associated with loss of an RCP.

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 LSSS.

(continued)

BASES

---

APPLICABLE  
SAFETY  
ANALYSES,  
LCO, and  
APPLICABILITY  
(continued)

11. Reactor Coolant Pump (RCP) Breaker Position (Continued)

In MODE 1 above the P-7 setpoint, the RCP Breaker Position trip must be OPERABLE. Below the P-7 setpoint, all reactor trips on loss of flow are automatically blocked since there is insufficient heat production to be concerned about DNB. Above the P-7 setpoint, the reactor trip on loss of flow in two RCS loops is automatically enabled.

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 all RCS loops. The voltage to each bus is monitored by one relay on each RCP for two relays per bus (1/2 coincidence on 2/2 buses). Above the P-7 setpoint, a loss of voltage detected on both RCP buses, i.e. a complete loss of flow event, will initiate a reactor trip. For this event, the under voltage trip Function will generate a reactor trip before the Reactor Coolant Flow—Low Trip Setpoint is reached. Time delays are incorporated into the Undervoltage RCPs channels to prevent reactor trips due to momentary electrical power transients.

The LCO requires two Undervoltage RCPs channels per bus to be OPERABLE (2 pumps per Bus with one channel per pump).

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 there is insufficient heat production to be concerned about DNB. Above the P-7 setpoint, the reactor trip on loss of flow in all four RCS loops is automatically enabled.

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 flow in two or more RCS loops from a major network frequency disturbance. An underfrequency condition will slow down the pumps, thereby reducing their coastdown time following a pump trip. An adequate coastdown time is required so that reactor heat can be removed immediately after reactor trip. The frequency of each RCP bus is monitored. Above the P-7 setpoint, a loss of frequency detected by two relays on one RCP bus will initiate a reactor trip. This trip Function will generate a reactor trip before the Reactor Coolant Flow—Low Trip Setpoint is reached.

(continued)

BASES

---

APPLICABLE  
SAFETY  
ANALYSES,  
LCO, and  
APPLICABILITY

13. Underfrequency Reactor Coolant Pumps (Continued)

Time delays are incorporated into the Underfrequency RCPs channels to prevent reactor trips due to momentary electrical power transients.

The LCO requires three Underfrequency RCPs channels per bus to be OPERABLE.

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 there is insufficient heat production to be concerned about DNB. Above the P-7 setpoint, the reactor trip on loss of flow in two or more RCS loops is automatically enabled.

14. Steam Generator Water Level—Low Low

- a. The SG Water Level—Low Low trip Function ensures that protection is provided against a loss of heat sink in the event of a loss of feedwater flow to one or more SGs. The SGs are the heat sink for the reactor. In order to act as a heat sink, the SGs must contain a minimum amount of water. A narrow range low low level in any SG 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 ESFAS function of starting the AFW pumps on low low SG level.

The LCO requires three channels of SG Water Level—Low Low per SG and four channels of RCS  $\Delta T$ (1/loop) to be OPERABLE. The installation of the median signal selector (MSS) and four channels of RCS  $\Delta T$  (1/loop) effectively eliminates the possibility that a single random failure could cause a control system action that results in a condition requiring protection action, and also prevent proper operation of a protection system channel designed to protect against the condition. Together with the MSS, the Steam Flow Arbitrator (SFA) eliminates the possibility that failure of the instrument tap shared between one narrow-range level channel and one steam flow channel on each steam generator will cause a transient that would require protective action by any of the level channels. The MSS prevents the

(continued)

BASES

---

APPLICABLE  
SAFETY  
ANALYSES,  
LCO, and  
APPLICABILITY

14. Steam Generator Water Level—Low Low (continued)

resulting failed high narrow range level signal from causing a level transient via the level portion of the DFWCS. The SFA prevents the resulting failed low steam flow signal from causing a level transient via the feed forward mass balance portion of the DFWCS.

Thus, the MSS prevents interaction between the feedwater control and reactor protection systems in accordance with the requirements of IEEE 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations." Removal of this interaction eliminates the need for the low feedwater flow reactor trip. The MSS will functionally separate steam generator narrow range level protection channels (low-low steam generator water level trip) to provide compliance with IEEE 279-1971 and satisfy the original design basis.

Additionally, since no adverse control system action can result from a single failed protection system instrument channel (including the shared instrument tap), a second random protection system failure, as would otherwise be required by IEEE 279-1971, need not be considered. This trip is actuated on two out of three low-low water level signals occurring in any steam generator. If a low-low water level condition is detected in one steam generator, signals shall be generated to trip the reactor and start the motor driven auxiliary feedwater pumps. If a low-low water level condition is detected in two or more steam generators, a signal is generated to start the turbine driven auxiliary feedwater pump as well.

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

(continued)

BASES

---

APPLICABLE  
SAFETY  
ANALYSES,  
LCO, and  
APPLICABILITY  
(continued)

14. Steam Generator Water Level—Low Low (continued)

- b. The signals to actuate reactor trip and start auxiliary feedwater pumps may be delayed through the use of a Trip Time Delay (TTD) system for reactor power levels below 50% of RTP. Low-low water level in any protection set in any steam generator will generate a signal which starts an elapsed time trip delay timer.

The allowable trip time delay is based upon the prevailing power level at the time the low-low level trip setpoint is reached. The Time Delay value used is determined as directed under Note 3. If power level rises after the trip time delay setpoints have been determined, the trip time delay is re-determined (i.e., decreased) according to the increase in power level. However, the trip time delay is not changed if the power level decreases after the delay has been determined. The use of this delay allows added time for natural steam generator level stabilization or operator intervention to avoid an inadvertent protection system actuation.

15. Steam Generator Water Level—Low, Coincident With Steam Flow/Feedwater Flow Mismatch - Not used.

16. Turbine Trip

a. Turbine Trip—Low Auto Stop Oil Pressure

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

(continued)

BASES

---

APPLICABLE  
SAFETY  
ANALYSES,  
LCO, and  
APPLICABILITY

16. Turbine Trip (continued)

The LCO requires three channels of Turbine Trip—Low Autostop Oil Pressure to be OPERABLE in MODE 1 above P-9 (2-out-of-3 coincidence).

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 Auto Stop Oil Pressure trip Function does not need to be OPERABLE.

b. Turbine Trip—Turbine Stop Valve Closure

The Turbine Trip—Turbine Stop Valve Closure trip Function anticipates the loss of heat removal capabilities of the secondary system following a turbine trip. 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. Any turbine trip from a power level below the P-9 setpoint, less than or equal to a maximum setpoint of 50 percent power, will not actuate a reactor trip. This trip Function will not and is not required to operate in the presence of a single channel failure. The unit is designed to withstand a complete loss of load and not sustain core damage or challenge the RCS pressure limitations. 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 Auto Stop 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 above P-9.

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

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

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

(continued)

BASES

---

APPLICABLE  
SAFETY  
ANALYSES,  
LCO, and  
APPLICABILITY

17. Safety Injection Input from Engineered Safety Feature Actuation System (continued)

The SI Input from ESFAS ensures that if a reactor trip has not already been generated by the RTS, the ESFAS automatic actuation logic will initiate a reactor trip upon any signal that initiates SI. This is a condition of acceptability for the small break LOCA but rod insertion is not credited for the large break LOCA (Ref. 3).

However, other transients and accidents take credit for varying levels of ESF performance and rely upon rod insertion, except for the most reactive rod that is assumed to be fully withdrawn, to ensure reactor shutdown. Therefore, a reactor trip is initiated every time an SI signal is present.

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

The LCO requires two trains of SI Input from ESFAS to be OPERABLE in MODE 1 or 2 (1-out-of-2 coincidence).

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. 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

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

a. Intermediate Range Neutron Flux, P-6

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

(continued)

BASES

---

APPLICABLE  
SAFETY  
ANALYSES,  
LCO, and  
APPLICABILITY

a. Intermediate Range Neutron Flux, P-6 (continued)

- on increasing power, the P-6 interlock allows the manual block of the NIS Source Range, Neutron Flux reactor trip and allows the high voltage to be de-energized. This prevents a premature block of the source range trip and allows the operator to ensure that the intermediate range is OPERABLE prior to leaving the source range, and
- on decreasing power, the P-6 interlock automatically energizes the NIS source range detectors and enables the NIS Source Range Neutron Flux reactor trip.

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 (1-out-of-2 coincidence).

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.

b. Low Power Reactor Trips Block, P-7

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

- (1) on increasing power, the P-7 interlock automatically enables reactor trips on the following Functions:
  - Pressurizer Pressure—Low;
  - Pressurizer Water Level—High;
  - Reactor Coolant Flow—Low (Low Flow in two or more RCS Loops);
  - 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).

(continued)

BASES

---

APPLICABLE  
SAFETY  
ANALYSES,  
LCO, and  
APPLICABILITY

b. Low Power Reactor Trips Block, P-7 (continued)

The reactor trips provide protection against violating the DNBR limit. Below the P-7 setpoint, the RCS is capable of providing sufficient natural circulation without any RCP running.

- (2) on decreasing power, the P-7 interlock automatically blocks reactor trips on the following Functions:
- Pressurizer Pressure—Low;
  - Pressurizer Water Level—High;
  - Reactor Coolant Flow—Low (Low Flow in two or more RCS Loops);
  - RCP Breaker Position (Two Loops);
  - Undervoltage RCPs; and
  - Underfrequency RCPs.

Trip Setpoint and Allowable Value are not applicable to the P-7 interlock because it is a logic Function and thus has no parameter with which to associate an LSSS. The P-7 train is operable if the P-10 and P-13 interlocks are in their required states based on plant conditions.

The P-7 interlock is a logic Function with train and not channel identity. Therefore, the LCO requires one channel per train of Low Power Reactor Trips Block, P-7 interlock to be OPERABLE in MODE 1 (1-out-of-2 coincidence).

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

c. Power Range Neutron Flux, P-8

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

(continued)

BASES

---

APPLICABLE  
SAFETY  
ANALYSES,  
LCO, and  
APPLICABILITY

c. Power Range Neutron Flux, P-8 (continued)

The LCO requires four channels of Power Range Neutron Flux, P-8 interlock to be OPERABLE in MODE 1 (2-out-of-4 coincidence).

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

d. Power Range Neutron Flux, P-9

The Power Range Neutron Flux, P-9 interlock is actuated at less than or equal to 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 Auto Stop 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 may challenge the pressurizer PORVs due to the mismatch between reactor power and the capacities of the Steam Dump and Reactor Control Systems. 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.

The LCO requires four channels of Power Range Neutron Flux, P-9 interlock to be OPERABLE in MODE 1 (2-out-of-4 coincidence).

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 significant load rejection beyond the capacity of the Steam Dump System.

e. Power Range Neutron Flux, P-10

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

(continued)

BASES

---

APPLICABLE  
SAFETY  
ANALYSES,  
LCO, and  
APPLICABILITY

e. Power Range Neutron Flux, P-10 (continued)

- on increasing power, the P-10 interlock allows the operator to manually block the Intermediate Range Neutron Flux reactor trip. Note that blocking the reactor trip also blocks the signal to prevent automatic and manual rod withdrawal;
- on increasing power, the P-10 interlock allows the operator to manually block the Power Range Neutron Flux—Low reactor trip;
- on increasing power, the P-10 interlock automatically provides a back up signal to block the Source Range Neutron Flux reactor trip, and also to de-energize the NIS source range detectors high voltage and allows manual block of the IR rod stop;
- the P-10 interlock provides one of the two inputs to the P-7 interlock;
- on decreasing power, the P-10 interlock automatically enables the Power Range Neutron Flux—Low reactor trip and the Intermediate Range Neutron Flux reactor trip (and rod stop); and
- on decreasing power, the P-10 interlock automatically defeats the block of the source range neutron flux trip and with P-6 energizes the source range high voltage.

The LCO requires four channels of Power Range Neutron Flux, P-10 interlock to be OPERABLE in MODE 1 or 2 (2-out-of-4).

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

(continued)

BASES

---

APPLICABLE  
SAFETY  
ANALYSES,  
LCO, and  
APPLICABILITY  
(continued)

f. Turbine Impulse Chamber Pressure, P-13

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

The LCO requires two channels of Turbine Impulse Chamber Pressure, P-13 interlock to be OPERABLE in MODE 1 (1-out-of-2-coincidence).

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

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, the trip logic, and all trip breakers associated with a single RTS logic train that are racked in, closed, and capable of supplying power to the Rod Control System. Thus, the train may consist of the main breaker, bypass breaker, or main breaker and bypass breaker, depending upon the system configuration. Two OPERABLE trains ensure no single random failure can disable the RTS trip capability.

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

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 Rod Control System, or declared inoperable under Function 19 above. OPERABILITY of both trip mechanisms on each breaker ensures that no single trip mechanism failure will prevent opening any breaker on a valid signal.

(continued)

BASES

---

APPLICABLE  
SAFETY  
ANALYSES,  
LCO, and  
APPLICABILITY

20. Reactor Trip Breaker Undervoltage and Shunt Trip Mechanisms  
(continued)

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

21. Automatic Trip Logic

The LCO requirement for the RTBs (Functions 19 and 20) and Automatic Trip Logic (Function 21) ensures that means are provided to interrupt the power to 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 unit is at power. The reactor trip signals generated by the RTS Automatic Trip Logic cause the RTBs and associated bypass breakers to open and shut down the reactor.

The LCO requires two trains of RTS Automatic Trip Logic to be OPERABLE (1-out-of-2 coincidence). Having two OPERABLE channels ensures that random failure of a single logic channel will not prevent reactor trip.

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

22. Seismic Trip

The seismic trip system operates to shut down reactor operations should ground accelerations exceed a preset level in any of the three orthogonal directions monitored (one vertical, two horizontal).

Three triaxial sensors (accelerometers) are anchored to the containment base in three separate locations 120 degrees apart. Each senses acceleration in three mutually orthogonal directions. Output signals are generated when ground accelerations exceed the preset level. These signals are transmitted to the Trains A and

(continued)

BASES

---

APPLICABLE  
SAFETY  
ANALYSES,  
LCO, and  
APPLICABILITY

22. Seismic Trip (continued)

B Solid State Protection System (SSPS). If two of the three sensors in any direction produce simultaneous outputs, the logic produces trains A and B reactor trip signals.

This trip function must be OPERABLE in MODE 1 or 2 when the reactor is critical and must be capable to shut down the reactor in the event of an earthquake. Three channels in these directions are required to be OPERABLE.

---

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

---

ACTIONS

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

In the event a channel's Trip 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(s) entered for the protection Function(s) affected. When the Required Channels in Table 3.3.1-1 are specified (e.g., on a per steam line, per loop, per SG), then the Condition may be entered separately for each steam line, loop, SG, etc., as appropriate.

In the event a channel's setpoint is found nonconservative with respect to the specified Trip Setpoint, but more conservative than the Allowable Value, the setpoint must be adjusted consistent with the Trip Setpoint value. When a channel's Trip Setpoint is nonconservative with respect to the Allowable Value, declare the channel inoperable and apply the applicable ACTION statement until the channel is returned to OPERABLE status with its Setpoint adjusted consistent with the Trip Setpoint value.

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

A.1

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

(continued)

BASES

---

ACTIONS

A.1 (continued)

Actions for the protection functions affected. The Completion Times are those from the referenced Conditions and Required Actions.

B.1, and B.2

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

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

If the Manual Reactor Trip Function cannot be restored to OPERABLE status within the allowed 48 hour Completion Time, the unit must be brought to a MODE in which the requirement does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 additional hours (54 hours total time). The 6 additional hours to reach MODE 3 is reasonable, based on operating experience, to exit the applicability from full power operation in an orderly manner and without challenging unit systems. With the unit in MODE 3, Condition C is entered if the Manual Reactor Trip Function has not been restored and the Rod Control System is capable of rod withdrawal or one more rods are not fully inserted.

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

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

- Manual Reactor Trip;
- RTBs;
- RTB Undervoltage and Shunt Trip Mechanisms; and
- Automatic Trip Logic.

This action addresses the train orientation of the SSPS for these Functions. With one channel or train inoperable, the inoperable channel or train must be restored to OPERABLE status within 48 hours. If the affected Function(s) cannot be restored to OPERABLE status within the allowed 48 hour Completion Time, the unit must be placed in a MODE in which the requirement does not apply. To achieve this status action must be initiated within the same 48 hours to fully insert all rods, and the Rod Control System must be rendered

(continued)

BASES

---

ACTIONS

C.1, C.2.1, and C.2.2 (continued)

incapable of rod withdrawal within the next hour (e.g., by de-energizing all CRDMs, by opening the RTBs, or by de-energizing the motor generator (MG) sets). The additional hour for the latter provides sufficient time to accomplish the action in an orderly manner. With the rods fully inserted and the Rod Control System rendered incapable of rod withdrawal, these Functions are no longer required.

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

Condition C is modified by a Note stating that while this LCO is not met for Functions 19, 20, or 21 in MODE 5 making the Rod Control System capable of rod withdrawal is not permitted. This note is in addition to the requirements of LCO 3.0.4 which preclude the transition from either MODE 3 or MODE 4 to MODE 3 or MODE 4 with the Rod control System capable of rod withdrawal or all rods not fully inserted for Functions 19, 20, or 21 with one channel or train inoperable.

D.1.1, D.1.2, D.2.1, D.2.2, and D.3

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

The NIS power range detectors provide input to the Rod Control System and, therefore, have a two-out-of-four trip logic. A known inoperable channel must be placed in the tripped condition. This results in a partial trip condition requiring only one-out-of-three logic for actuation. The 6 hours allowed to place the inoperable channel in the tripped condition is justified in WCAP-10271-P-A (Ref. 7).

In addition to placing the inoperable channel in the tripped condition, THERMAL POWER must be reduced to  $\leq 75\%$  RTP within 12 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.

As an alternative to the above actions, the inoperable channel can be placed in the tripped condition within 6 hours and the QPTR monitored once every 12 hours as per SR 3.2.4.2, including the SR 3.2.4.2 note, for QPTR verification. Calculating QPTR every 12 hours compensates for the lost monitoring capability due to the inoperable NIS power range channel and allows continued unit operation at power levels  $> 75\%$  RTP. The 6 hour Completion Time and the 12 hour Frequency are consistent with LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)."

(continued)

BASES

---

ACTIONS

D.1.1, D.1.2, D.2.1, D.2.2, and D.3 (continued)

As an alternative to the above Actions, the plant must be placed in a MODE where this Function is no longer required OPERABLE. Twelve hours are allowed to place the plant in MODE 3. This is a reasonable time, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging plant systems. If Required Actions cannot be completed within their allowed Completion Times, LCO 3.0.3 must be entered.

The Required Actions have been modified by a Note that allows placing the inoperable channel in the bypass condition for up to 4 hours while performing routine surveillance testing of other channels. The Note also allows placing the inoperable channel in the bypass condition to allow setpoint adjustments of other channels when required to reduce the setpoint in accordance with other Technical Specifications. In accordance with WCAP 10271, very specific circumstances are related to the use of this bypass condition. Since the NIS channels are not designed with Bypass-capable logic that meets the requirements of IEEE 279, the provisions for bypass only apply to a specific type of channel failure. To apply, the channel must fail in such a way that it does not trip the bistables. With this type of failure, the channel may be returned to service and considered "bypassed" under this Note. Specifically, the bypass condition is the state when a failed channel is taken out of the forced "tripped" state and placed in operation. Due to the failed nature of the channel, the channel cannot be assumed to be OPERABLE, and is therefore considered to be in a state of bypass when the channel failure is such that its bistables are not tripped. The provisions of WCAP 10271 specifically prohibit the use of jumpers or lifted leads to bypass these channels. In this configuration, a second channel can be tested or setpoints adjusted with the channel in the tripped mode without completing reactor trip logic. The 4 hour time limit is justified in Reference 7.

Required Action D.2.2 has been modified by a Note which only requires SR 3.2.4.2 to be performed if the Power Range Neutron Flux input to QPTR becomes inoperable. The performance of SR 3.2.4.2 per ACTION D.2.2 is subject to the SR 3.2.4.2 note. Failure of a component in the Power Range Neutron Flux Channel which renders the High Flux Trip Function inoperable may not affect the capability to monitor QPTR. As such, determining QPTR using this movable incore detectors once per 12 hours may not be necessary.

(continued)

BASES

---

ACTIONS  
(Continued)

E.1 and E.2

Condition E applies to the following reactor trip Functions:

- Power Range Neutron Flux—Low;
- Overtemperature  $\Delta T$ ;
- Overpower  $\Delta T$ ;
- Power Range Neutron Flux—High Positive Rate;
- Power Range Neutron Flux—High Negative Rate;
- Pressurizer Pressure—High; and
- SG Water Level—Low Low.

A known inoperable channel must be placed in the tripped condition within 6 hours. Placing the channel in the tripped condition results in a partial trip condition requiring only one-out-of-two logic for actuation of the two-out-of-three trips 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 placed in the trip condition within the specified Completion Time, the unit must be placed in a MODE where these Functions are not required OPERABLE. An additional 6 hours is allowed to place the unit in MODE 3. Six hours is a reasonable time, based on operating experience, to place the unit in MODE 3 from full power in an orderly manner and without challenging unit systems.

(continued)

---

BASES

---

ACTIONS

E.1 and E.2 (continued)

The Required Actions have been modified by a Note that allows placing the inoperable channel, or one additional channel in the bypassed condition for Functions 6, 7, and 8b for up to 4 hours while performing routine surveillance testing of the other channels. The NOTE allows only the inoperable channel for FUNCTIONS 2b, 3a, 3b and 14.a to be bypassed for surveillance testing of other channels. In accordance with WCAP 10271, very specific circumstances are related to the use of this bypass condition for RTS Functions 2.b, 3.a, and 3.b. Since these channels are not designed with Bypass-capable logic that meets the requirements of IEEE 279, the provisions for bypass only apply to a specific type of channel failure. To apply, the channel must fail in such a way that it does not trip the bistables. With this type of failure, the channel may be returned to service and considered "bypassed" under this Note. Specifically, the bypass condition is the state when a failed channel is taken out of the forced "tripped" state and placed in operation. Due to the failed nature of the channel, the channel cannot be assumed to be OPERABLE, and is therefore considered to be in a state of bypass when the channel failure is such that its bistables are not tripped. The provisions of WCAP 10271 specifically prohibit the use of jumpers or lifted leads to bypass these channels. In this configuration, a second channel can be tested with the channel in the tripped mode without completing reactor trip logic. The 4 hour time limit is justified in Reference 7.

F.1 and F.2

Condition F applies to the Intermediate Range Neutron Flux trip when THERMAL POWER is above the P-6 setpoint and below the P-10 setpoint and one channel is inoperable. Above the P-6 setpoint and below the P-10 setpoint, the NIS intermediate range detector performs the monitoring Functions. If THERMAL POWER is greater than the P-6 setpoint but less than the P-10 setpoint, 24 hours is allowed to reduce THERMAL POWER below the P-6 setpoint or increase to THERMAL POWER above the P-10 setpoint. The NIS Intermediate Range Neutron Flux channels must be OPERABLE when the power level is above the capability of the source range, P-6, and below the capability of the power range, P-10. If THERMAL POWER is greater than the P-10 setpoint, the NIS power range detectors perform the monitoring and protection functions and the intermediate range is not required. The Completion Times allow for a slow and controlled power adjustment above P-10 or below P-6 and take into account the

(continued)

BASES

---

ACTIONS  
(continued)

F.1 and F.2 (Continued)

redundant capability afforded by the redundant OPERABLE channel, the overlap of the power range detectors, and the low probability of its failure during this period. This action does not require the inoperable channel to be tripped because the Function uses one-out-of-two logic. Tripping one channel would trip the reactor. Thus, the Required Actions specified in this Condition are only applicable when channel failure does not result in reactor trip.

G.1 and G.2

Condition G applies to two inoperable Intermediate Range Neutron Flux trip channels in MODE 2 when THERMAL POWER is above the P-6 setpoint and below the P-10 setpoint. Required Actions specified in this Condition are only applicable when channel failures do not result in reactor trip. Above the P-6 setpoint and below the P-10 setpoint, the NIS intermediate range detector performs the monitoring Functions. With no intermediate range channels OPERABLE, the Required Actions are to suspend operations involving positive reactivity additions immediately. This will preclude any power level increase since there are no OPERABLE Intermediate Range Neutron Flux channels. The operator must also reduce THERMAL POWER below the P-6 setpoint within two hours. Below P-6, the Source Range Neutron Flux channels will be able to monitor the core power level. The Completion Time of 2 hours will allow a slow and controlled power reduction to less than the P-6 setpoint and takes into account the low probability of occurrence of an event during this period that may require the protection afforded by the NIS Intermediate Range Neutron Flux trip.

H.1 - Not used

I.1

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 unit in this Condition, below P-6, the NIS source range performs the monitoring and protection functions. With one of the two channels inoperable, operations involving positive reactivity additions shall be suspended immediately.

This will preclude any power escalation. With only one source range channel OPERABLE, core protection is severely reduced and any actions that add positive reactivity to the core must be suspended immediately.

(continued)

BASES

---

ACTIONS  
(continued)

J.1

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 Rod Control System capable of rod withdrawal or one or more rods not fully inserted. With the unit in this Condition, below P-6, the NIS source range performs the monitoring and protection functions. With both source range channels inoperable, the RTBs must be opened immediately. With the RTBs open, the core is in a more stable condition and the unit enters Condition L.

K.1, K.2.1, and K.2.2

Condition K applies to one inoperable source range channel in MODE 3, 4, or 5 with the Rod Control System capable of rod withdrawal or one or more rods not fully inserted. With the unit in this Condition, below P-6, the NIS source range performs the protection 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, action must be initiated within the same 48 hours to fully insert all rods. 1 additional hour is allowed to place the Rod Control System in a condition incapable of rod withdrawal (e.g., by de-energizing all CRDMs, by opening the RTBs, or by de-energizing the motor generator (MG) sets). Once these ACTIONS are completed the core is in a more stable condition. The allowance of 48 hours to restore the channel to OPERABLE status, and the additional hour to place the Rod Control System in a condition incapable of rod withdrawal, are justified in Reference 7.

L.1, L.2, and L.3

Condition L applies when the required number of OPERABLE Source Range Neutron Flux channels is not met in MODE 3, 4, or 5 with the RTBs open or with the Rod Control System incapable of rod withdrawal and all rods fully inserted. With the unit in this Condition, the NIS source range performs a monitoring function. With less than the required number of source range channels OPERABLE, operations involving positive reactivity additions shall be suspended immediately. This will preclude any power escalation.

Also, the SDM must be verified within 1 hour and once every 12 hours thereafter as per SR 3.1.1.1, SDM verification. With no source range channels OPERABLE, core protection is severely reduced. Verifying the SDM within 1 hour allows sufficient time to perform the calculations and determine that the SDM requirements are met. The SDM must

(continued)

BASES

---

ACTIONS

L.1, L.2, and L.3 (Continued)

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 unit conditions will change slowly.

M.1 and M.2

Condition M applies to the following reactor trip Functions:

- Pressurizer Pressure — Low;
- Pressurizer Water Level — High;
- Reactor Coolant Flow — Low;
- RCP Breaker Position ;
- Undervoltage RCPs; and
- Underfrequency RCPs.

With one channel inoperable, the inoperable channel must be placed in the tripped condition within 6 hours. For the Pressurizer Pressure - Low, Pressurizer Water Level - High, Undervoltage RCPs, and Underfrequency RCPs trip Functions, placing the channel in the tripped condition when above the P-7 setpoint results in a partial trip condition requiring only one additional channel to initiate a reactor trip. For the Reactor Coolant Flow - Low trip Function, placing the channel in the tripped condition results in a partial trip condition requiring only one additional channel to initiate a reactor trip above the P-7 and P-8 setpoints. These Functions do not have to be OPERABLE below the P-7 setpoint because there are no loss of flow trips below the P-7 setpoint. The 6 hours allowed to place the channel in the tripped condition is justified in Reference 7. An additional 6 hours is allowed to reduce THERMAL POWER to below P-7 if the inoperable channel cannot be restored to OPERABLE status or placed in trip within the specified Completion Time. The Reactor Coolant Flow - Low reactor trip function goes from 1 of 4 logic to 2 of 4 logic below the P-8 setpoint; however, the Required Action must take the plant below the P-7 setpoint, if an inoperable channel is not tripped within 6 hours, due to the shared components between this function and the Reactor Coolant Flow - Low trip function.

(continued)

---

BASES

ACTIONS  
(continued)

M.1 and M.2 (Continued)

Allowance of this time interval takes into consideration the redundant capability provided by the remaining redundant OPERABLE channel, and the low probability of occurrence of an event during this period that may require the protection afforded by the Functions associated with Condition M.

The Required Actions have been modified by a Note that allows placing the inoperable channel, or one additional channel for Function 8a, in the bypassed condition for up to 4 hours while performing routine surveillance testing of the other channels. The Note allows only the inoperable channel for Functions 9, 10, 12 and 13 to be bypassed for surveillance testing of other channels. Function 11 may not be bypassed since its logic is not 2 of 4 or 2 of 3, therefore, single failure would not be maintained. In accordance with WCAP 10271, very specific circumstances are related to the use of this bypass condition for RTS Functions 12 and 13. Since these channels are not designed with Bypass-capable logic that meets the requirements of IEEE 279, the provisions for bypass only apply to a specific type of channel failure. To apply, the channel must fail in such a way that it does not trip the bistables. With this type of failure, the channel may be returned to service and considered "bypassed" under this Note. Specifically, the bypass condition is the state when a failed channel is taken out of the forced "tripped" state and placed in operation. Due to the failed nature of the channel, the channel cannot be assumed to be OPERABLE, and is therefore considered to be in a state of bypass when the channel failure is such that its bistables are not tripped. The provisions of WCAP 10271 specifically prohibit the use of jumpers or lifted leads to bypass these channels. In this configuration, a second channel can be tested with the channel in the tripped mode without completing reactor trip logic. The 4 hour time limit is justified in Reference 7.

N.1 and N.2 - Not used

O.1 and O.2

Condition O applies to Turbine Trip on Low Auto-Stop Oil Pressure. With one channel inoperable, the inoperable channel must be 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.

(continued)

BASES

---

ACTIONS  
(continued)

O.1 and O.2 (Continued)

The Required Actions have been modified by a Note that allows placing an inoperable channel in the bypassed condition for up to 4 hours while performing routine surveillance testing of the other channels. In accordance with WCAP 10271, very specific circumstances are related to the use of this bypass condition for RTS Function 16. Since this channel is not designed with Bypass-capable logic that meets the requirements of IEEE 279, the provisions for bypass only apply to a specific type of channel failure. To apply, the channel must fail in such a way that it does not trip the bistables. With this type of failure, the channel may be returned to service and considered "bypassed" under this Note. Specifically, the bypass condition is the state when a failed channel is taken out of the forced "tripped" state and placed in operation. Due to the failed nature of the channel, the channel cannot be assumed to be OPERABLE, and is therefore considered to be in a state of bypass when the channel failure is such that its bistables are not tripped. The provisions of WCAP 10271 specifically prohibit the use of jumpers or lifted leads to bypass this channel. In this configuration, a second channel can be tested with the channel in the tripped mode without completing reactor trip logic. The 4 hour time limit is justified in Reference 7.

P.1 and P.2

Condition P applies to Turbine Trip on Turbine Stop Valve Closure. With one or more channels inoperable, the inoperable channel must be placed in the trip condition within 6 hours. For the Turbine Trip on Turbine Stop Valve Closure function, where four-of-four channels are required to initiate a reactor trip; hence more than one channel may be placed in trip. If the channel(s) 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(s) in the tripped condition and the 4 hours allowed for reducing power are justified in Reference 7.

Q.1 and Q.2

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 Functions. With one train inoperable, 6 hours are allowed to restore the train to OPERABLE status (Required Action Q.1) or the unit must be placed in MODE 3 within the next 6 hours. The Completion Time of 6 hours (Required

(continued)

---

BASES

---

ACTIONS

Q.1 and Q.2 (Continued)

Action Q.1) is reasonable considering that in this Condition, the remaining OPERABLE train is adequate to perform the safety function and given the low probability of an event during this interval. The Completion Time of 6 hours (Required Action Q.2) is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging unit systems.

The Required Actions have been modified by a Note that allows bypassing one train up to 4 hours for surveillance testing, provided the other train is OPERABLE.

R.1 and R.2

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 unit 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 unit in MODE 3 results in Condition C entry if one RTB train is inoperable.

The Required Actions have been modified by two Notes. Note 1 allows one train to be bypassed for up to 2 hours for surveillance testing, provided the other train is OPERABLE. Note 2 allows one RTB to be bypassed only for the time required for performing maintenance on undervoltage or shunt trip mechanisms per Condition U if the other RTB train is OPERABLE. The time limits are justified in Reference 7.

S.1 and S.2

Condition S applies to the P-6 and P-10 interlocks. With one or more channels inoperable, the associated interlock must be verified by observation of the associated permissive annunciator window to be in its required state for the existing unit condition within 1 hour or the unit must be placed in MODE 3 within the next 6 hours. Verifying the interlock status manually accomplishes the interlock's Function. The Completion Time of 1 hour is based on operating experience and the minimum amount of time allowed for manual operator actions. 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 unit 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.

(continued)

BASES

---

ACTIONS  
(continued)

T.1 and T.2

Condition T applies to the P-7, P-8, P-9, and P-13 interlocks. With one or more channel(s) inoperable, the associated interlock must be verified by observation of the associated permissive annunciator window to be in its required state for the existing unit condition within 1 hour or the unit must be placed in MODE 2 within the next 6 hours. These actions are conservative for the case where power level is being raised. Verifying the interlock status manually accomplishes the interlock's Function. The Completion Time of 1 hour is based on operating experience and the minimum amount of time allowed for manual operator actions. 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 unit systems.

U.1 and U.2

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 unit in MODE 3 within the next 6 hours (54 hours total time). 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 unit systems.

With the unit in MODE 3, Condition C is entered if the inoperable trip mechanism has not been restored and the Rod Control System is capable of rod withdrawal or one or more rods are not fully inserted. The affected RTB shall not be bypassed while one of the diverse features is inoperable except for the time required to perform maintenance to restore the inoperable trip mechanism to OPERABLE status, consistent with Ref. 13.

The Completion Time of 48 hours for Required Action U.1 is reasonable considering that in this Condition there is one remaining diverse feature for the affected RTB, and one OPERABLE RTB capable of performing the safety function and given the low probability of an event occurring during this interval.

V.1 - Not used

---

(continued)