NUREG–1430
Vol. 2, Rev. 1

Standard Technical Specifications
Babcock and Wilcox Plants

Bases (Sections 2.0–3.3)

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PREFACE

This NUREG contains the improved Standard Technical Specifications (STS) for Babcock and Wilcox (B&W) plants. Revision 1 incorporates the cumulative changes to Revision 0, which was published in September 1992. The changes reflected in Revision 1 resulted from the experience gained from license amendment applications to convert to these improved STS or to adopt partial improvements to existing technical specifications. This NUREG is the result of extensive public technical meetings and discussions between the Nuclear Regulatory Commission (NRC) staff and various nuclear power plant licensees, Nuclear Steam Supply System (NSSS) Owners Groups, specifically the B&W Owners Group (BWOG), NSSS vendors, and the Nuclear Energy Institute (NEI). The improved STS were developed based on the criteria in the Final Commission Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors, dated July 22, 1993 (58 FR 39132). Licensees are encouraged to upgrade their technical specifications consistent with those criteria and conforming, to the extent practical and consistent with the licensing basis for the facility, to Revision 1 to the improved STS. The Commission continues to place the highest priority on requests for complete conversions to the improved STS. Licensees adopting portions of the improved STS to existing technical specifications should adopt all related requirements, as applicable, to achieve a high degree of standardization and consistency.

(continued)

B 2.0
SAFETY LIMITS

B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core SLs

BASES

BACKGROUND

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

The restrictions of this SL prevent overheating of the fuel and cladding and possible cladding perforation that would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady state peak linear heat rate (LHR) below the level at which 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.

(continued)

BASES

BACKGROUND (continued)

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

APPLICABLE SAFETY ANALYSES

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

a. There must be at least 95% probability at a 95% confidence level (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 fuel centerline melting.

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

Automatic enforcement of these reactor core SLs is provided by the following:

a. RCS High Pressure trip;

b. RCS Low Pressure trip;

c. Nuclear Overpower trip;

d. RCS Variable Low Pressure trip;

e. Reactor Coolant Pump to Power trip;

f. Nuclear Overpower RCS Flow and Axial Power Imbalance trip; and

g. MSSVs.

The SL represents a design requirement for establishing the RPS trip setpoints identified previously.

(continued)

BASES (continued)

SAFETY LIMITS

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

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

Operation within these limits is ensured by compliance with the AXIAL POWER IMBALANCE protective limits preserved by their corresponding RPS setpoints in LCO 3.3.1, "Reactor Protection System (RPS) Instrumentation," as specified in the COLR. The AXIAL POWER IMBALANCE protective limits are separate and distinct from the AXIAL POWER IMBALANCE operating limits defined by LCO 3.2.3, "AXIAL POWER IMBALANCE Operating Limits." The AXIAL POWER IMBALANCE operating limits in LCO 3.2.3, also specified in the COLR, preserve initial conditions of the safety analyses but are not reactor core SLs.

APPLICABILITY

SL 2.1.1.1, SL 2.1.1.2, and SL 2.1.1.3 only apply in MODES 1 and 2 because these are the only MODES in which the reactor is critical. Automatic protection functions are required to be OPERABLE during MODES 1 and 2 to ensure operation within the reactor core SLs. The MSSVs, or automatic protection actions, serve to prevent RCS heatup to reactor core SL conditions or to initiate a reactor trip function, which forces the unit into MODE 3. Setpoints for the reactor trip functions are specified in LCO 3.3.1.

(continued)

BASES

APPLICABILITY (continued)

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.

2.2.1 and 2.2.2

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

The allowed Completion Time of 1 hour recognizes the importance of bringing the plant to a MODE of operation where these SLs are not applicable and reduces the probability of fuel damage.

2.2.5

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

2.2.6

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

2.2.7

If SL 2.1.1.1, SL 2.1.1.2, or SL 2.1.1.3 is violated, a Licensee Event Report shall be prepared and submitted within 30 days to the NRC in accordance with 10 CFR 50.73 (Ref. 4). A copy of the report shall also be submitted to the senior

(continued)

BASES

SAFETY LIMIT VIOLATIONS

2.2.7 (continued)

management of the nuclear plant, and the utility Vice President—Nuclear Operations.

2.2.8

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

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.
2. FSAR, Section [ ].
3. 10 CFR 50.72.
4. 10 CFR 50.73.

B 2.0 SAFETY LIMITS (SLs)

B 2.1.2 Reactor Coolant System (RCS) Pressure SL

BASES

BACKGROUND

According to 10 CFR 50, Appendix A, GDC 14, "Reactor Coolant Pressure Boundary," and GDC 15, "Reactor Coolant System Design" (Ref. 1), the reactor coolant pressure boundary (RCPB) design conditions are not to be exceeded during normal operation nor during anticipated operational occurrences (AOOs). GDC 28, "Reactivity Limits" (Ref. 1), specifies that reactivity accidents including rod ejection do not result in damage to the RCPB greater than limited local yielding.

The design pressure of the RCS is 2500 psig. During normal operation and AOOs, the RCS pressure is kept from exceeding the design pressure by more than 10% in order to remain in accordance with Section III of the ASME Code (Ref. 2). Hence, the safety limit is 2750 psig. To ensure system integrity, all RCS components are hydrostatically tested at 125% of design pressure prior to initial operation, according to the ASME Code requirements. Inservice operational hydrotesting at 100% of design pressure is also required whenever the reactor vessel head has been removed or if other pressure boundary joint alterations have occurred. Following inception of unit operation, RCS components shall be pressure tested, in accordance with the requirements of ASME Code, Section XI (Ref. 3).

APPLICABLE SAFETY ANALYSES

The RCS pressurizer safety valves, operating in conjunction with the Reactor Protection System trip settings, ensure that the RCS pressure SL will not be exceeded.

The RCS pressurizer safety valves are sized to prevent system pressure from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME Code for Nuclear Power Plant Components (Ref. 2). The transient that is most influential for establishing the required relief capacity, and hence the valve size requirements and lift settings, is a rod withdrawal from low power. During the transient, no control actions are assumed except that the safety valves on the secondary plant are assumed to open

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

when the steam pressure reaches the secondary plant safety valve settings, and nominal feedwater supply is maintained.

The overpressure protection analyses (Ref. 4) and the safety analyses (Ref. 5) 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 turbine bypass valves;

c. Control system runback of reactor and turbine power; and

d. 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 2750 psig.

Overpressurization of the RCS can result in a breach of the RCPB. If such a breach occurs in conjunction with a fuel cladding failure, fission products could enter the containment atmosphere, raising concerns relative to limits on radioactive releases specified in 10 CFR 100, "Reactor Site Criteria" (Ref. 7).

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 during overpressurization events. The SL is not applicable in MODE 6 because the reactor vessel head closure bolts are not fully tightened, making it unlikely that the RCS can be pressurized.

(continued)

BASES (continued)

SAFETY LIMIT VIOLATIONS

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

2.2.3

If the RCS pressure SL is violated when the reactor is in MODE 1 or 2, the requirement is to restore compliance 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 7).

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

2.2.4

If the RCS pressure SL is exceeded in MODE 3, 4, or 5, RCS pressure must be restored to within the SL value within 5 minutes.

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

2.2.5

If the RCS pressure SL is violated, the NRC Operations Center must be notified within 1 hour, in accordance with 10 CFR 50.72 (Ref. 8).

(continued)

BASES

SAFETY LIMIT VIOLATIONS (continued)

2.2.6

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

2.2.7

If the RCS pressure SL is violated, a Licensee Event Report shall be prepared and submitted within 30 days to the NRC, in accordance with 10 CFR 50.73 (Ref. 9). A copy of the report shall also be provided to the senior management of the nuclear plant, and the utility Vice President—Nuclear Operations and the [offsite reviewers specified in Specification 5.2.2] ["Offsite Review and Audit"].

2.2.8

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

REFERENCES

1. 10 CFR 50, Appendix A, GDC 14, GDC 15, and GDC 28, 1988.
2. ASME Boiler and Pressure Vessel Code, Section III, Article NB-7000.
3. ASME Boiler and Pressure Vessel Code, Section XI, Article IW-5000.
4. BAW-10043, May 1972.
5. FSAR, Section [14].
6. ASME USAS B31.1, Standard Code for Pressure Piping, 1967.

(continued)

BASES

REFERENCES (continued)

7. 10 CFR 100.

8. 10 CFR 50.72.

9. 10 CFR 50.73.

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

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:

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

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

There are two basic types of Required Actions. The first type of Required Action specifies a time limit 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

(continued)

BASES

LCO 3.0.2 (continued)

ACTIONS.) The second type of Required Action specifies the remedial measures that permit continued operation of the unit that is not further restricted by the Completion Time. In this case, compliance with the Required Actions provides an acceptable level of safety for continued operation.

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

The nature of some Required Actions of some Conditions necessitates that, once the Condition is entered, the Required Actions must be completed even though the associated Conditions no longer exist. The individual LCO's ACTIONS specify the Required Actions where this is the case. 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. 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. Alternatives that would not result in redundant equipment being inoperable should be used instead. Doing so limits the time both subsystems/trains of a safety function are inoperable and limits the time other conditions exist which 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.

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BASES (continued)

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 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.

(continued)

BASES

LCO 3.0.3 (continued)

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.

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 3.7.14, "Fuel Storage Pool Water Level." LCO 3.7.14 has an Applicability of "During movement of irradiated fuel

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BASES

LCO 3.0.3 (continued)

assemblies in fuel storage pool." Therefore, this LCO can be applicable in any or all MODES. If the LCO and the Required Actions of LCO 3.7.14 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 3.7.14 of "Suspend movement of irradiated fuel assemblies in 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.

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

(continued)

BASES

LCO 3.0.4 (continued)

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. 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 MODES 1, 2, 3, or 4. The requirements of LCO 3.0.4 do not apply in MODES 5 and 6, or in other specified conditions of the Applicability (unless in MODES 1, 2, 3, or 4) because the ACTIONS or individual specifications sufficiently define the remedial measures to be taken. [In some cases (e.g., ..) 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.

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

(continued)

BASES

LCO 3.0.5 (continued)

the applicable Required Action(s)) to allow the performance of SRs to demonstrate:

a. The OPERABILITY of the equipment being returned to service; or

b. The OPERABILITY of other equipment.

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

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

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

LCO 3.0.6

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.

(continued)

BASES

LCO 3.0.6 (continued)

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 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 in 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 remaining OPERABLE support systems are OPERABLE, thereby ensuring safety function is retained. If this evaluation determines that a loss of safety function exists, the appropriate Conditions and

(continued)

BASES

LCO 3.0.6 (continued)

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 LCOs [3.1.9, 3.1.10, 3.1.11, and 3.4.19] allow specified Technical Specification (TS) requirements to be changed to permit performances of these special tests and operations, which otherwise could not be performed 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 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 to be not 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 Special Test Exception (STE) LCO are only applicable when the STE LCO is used as an allowable exception to the requirements of a Specification.

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

(continued)

BASES

SR 3.0.1 (continued)

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 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. An example of where SR 3.0.2 does not apply is a Surveillance with a Frequency of "in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions." The requirements of regulations take precedence over the TS. The TS cannot in and of themselves extend a test interval specified in the regulations.

(continued)

BASES

SR 3.0.2 (continued)

Therefore, there is a Note in the Frequency stating, "SR 3.0.2 is not applicable."

As stated in SR 3.0.2, the 25% extension also does not apply to the initial portion of a periodic Completion Time 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 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 an 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

(continued)

BASES

SR 3.0.3 (continued)

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 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

(continued)

BASES

SR 3.0.4 (continued)

failure to exercise the good practice of restoring systems or components to OPERABLE status before entering an associated MODE or other specified condition in the Applicability.

However, in certain circumstances, failing to meet an SR will not result in SR 3.0.4 restricting a MODE change or other specified condition change. When a system, subsystem, division, component, device, or variable is inoperable or outside its specified limits, the associated SR(s) are not required to be performed, per SR 3.0.1, which states 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 entry into 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.

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 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

(continued)

BASES

SR 3.0.4 (continued)

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

The reactivity control systems must be redundant and capable of holding the reactor core subcritical when shut down under cold conditions GDC 26 (Ref. 1). 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). In MODES 3, 4, and 5, the SDM defines the degree of subcriticality that would be obtained immediately following the insertion of all safety and regulating rods, assuming the single CONTROL ROD 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 CONTROL RODS can compensate for the reactivity effects of the fuel and water temperature changes accompanying power level changes over the range from full load to no load. In addition, the CONTROL RODS, together with the Chemical Addition and Makeup System, provide SDM during power operation and are 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 Addition and Makeup System can compensate for fuel depletion, during operation and all xenon burnout reactivity changes, and maintain the reactor subcritical under cold conditions.

During power operation, SDM control is ensured by operating with the safety rods fully withdrawn (LCO 3.1.5, "Safety Rod Insertion Limits") and the regulating rods within the limits of LCO 3.2.1, "Regulating Rod 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. Adjusted SDM limits defined in the COLR preclude recriticality in the event of a main steam line break (MSLB) in MODE 3, 4, or 5 when high steam generator levels exist.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

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

The acceptance criteria for SDM requirements are that specified acceptable fuel design limits are maintained. The SDM requirements must ensure 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 with acceptable limits (departure from nucleate boiling ratio (DNBR), fuel centerline temperature limits for AOOs, and ≤ 280 cal/gm energy deposition for the rod ejection accident); and

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

The most limiting accident for the SDM requirements is based on an MSLB, as described in the accident analysis (Ref. 2).

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

a. Inadvertent boron dilution;

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

c. Startup of an inactive reactor coolant pump;

d. Rod ejection; and

e. Return to criticality if an MSLB occurs during high steam generator level operations in MODE 3, 4, or 5.

The basis for the shutdown requirement when high steam generator levels exist is the heat removal potential in the

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

secondary system fluid and the negative reactivity added via MTC. At any given initial primary system temperature and its associated secondary system pressure, the secondary system liquid levels can be equated to a final primary system temperature assuming the entire mass is boiled. The resulting RCS temperature determines the required SDM.

SDM satisfies Criterion 2 of the NRC Policy Statement.

LCO

Shutdown boron concentration requirements assume the highest worth rod is stuck in the fully withdrawn position to account for a postulated inoperable or untrippable rod prior to reactor shutdown.

SDM is a core design condition that can be ensured through CONTROL ROD positioning (control and shutdown groups) and through the soluble boron concentration.

The MSLB (Ref. 2) accident is the most limiting analysis that establishes 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 limits (Ref. 3).

To compensate for the potential heat removal associated with an MSLB accident when high steam generator levels exist during secondary system chemistry control and steam generator cleaning, the initial SDM in the core must be adjusted. The figure in the COLR represents a series of initial conditions that ensure the core will remain subcritical following an MSLB accident from those conditions.

APPLICABILITY

In MODES 3, 4, and 5, the SDM requirements are applicable to provide sufficient negative reactivity to meet the assumptions of the safety analysis discussed above. The figure in the COLR is used to define the SDM when high steam generator levels exist during secondary system chemistry control and steam generator cleaning in MODES 3, 4, and 5. In MODES 1 and 2, SDM is ensured by complying with LCO 3.1.5

(continued)

BASES

APPLICABILITY (continued)

and LCO 3.2.1. In MODE 6, the shutdown reactivity requirements are given in LCO 3.9.1, "Boron Concentration."

ACTIONS

A.1

If the SDM requirements are not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. It is assumed that boration will be continued until the SDM requirements are met. If the SDM is below the limit for the steam generator level and RCS temperature specified in the COLR, RCS boration must be continued until the limit specified in the COLR is met.

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

In determining the boration flow rate, the time in core life must be considered. For instance, the most difficult time in core life to increase the RCS boron concentration is at the beginning of cycle, when the boron concentration may approach or exceed 2000 ppm. Assuming that a value of [1]% $\Delta k/k$ must be recovered and a boration flow rate is [ ] gpm, it is possible to increase the boron concentration of the RCS by 100 ppm in approximately 35 minutes. If a boron worth of 10 pcm/ppm is assumed, this combination of parameters will increase the SDM by [1]% $\Delta k/k$. These boration parameters of [ ] gpm and [ ] ppm represent typical values and are provided for the purpose of offering a specific example.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1

The SDM is verified by performing a reactivity balance calculation, considering the listed reactivity effects:

a. RCS boron concentration;

b. Regulating rod position;

c. RCS average temperature;

d. Fuel burnup based on gross thermal energy generation;

e. Xenon concentration;

f. Samarium concentration; and

g. Isothermal temperature coefficient (ITC).

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

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

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26.

2. FSAR, Chapter [14].

3. 10 CFR 100, "Reactor Site Criteria."

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.2 Reactivity Balance

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, the reactivity balance is used as a measure of the predicted versus measured core reactivity during power operation. The periodic confirmation of core reactivity is necessary to ensure that safety analyses of design basis transients and accidents remain valid. A large reactivity difference could be the result of unanticipated changes in fuel, CONTROL ROD, or burnable poison worth, or operation at conditions not consistent with those assumed in the predictions of core reactivity. These 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

(continued)

BASES

BACKGROUND (continued)

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

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

Design calculations and safety analyses are performed for each fuel cycle for the purpose of predetermining reactivity

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

behavior and the RCS boron concentration requirements for reactivity control during fuel depletion.

The comparison between measured and predicted initial core reactivity provides a normalization for the calculational models used to predict core reactivity. If the measured and predicted RCS boron concentrations for identical core conditions at beginning of cycle (BOC) do not agree, then the assumptions used in the reload cycle design analysis or the 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, which is developed during fuel depletion, may be an indication that the calculational model is not adequate for core burnups beyond BOC, or that an unexpected change in core conditions has occurred.

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

Reactivity balance satisfies Criterion 2 of the NRC Policy Statement.

LCO

Long term core reactivity behavior is a result of the core physics design and cannot be easily controlled, once the core design is fixed. During operation, therefore, the conditions of the LCO can only be ensured through measurement and tracking, and appropriate actions taken as necessary. Large differences between actual and predicted core reactivity may indicate that the assumptions of the Design Basis Accident (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 of ± 1% $\Delta k/k$ has been established, based on engineering judgment. A ± 1% $\Delta k/k$ deviation in reactivity

(continued)

BASES

LCO (continued)

from that predicted is larger than expected for normal operation and should therefore be evaluated.

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

APPLICABILITY

In MODES 1 and 2 during fuel cycle operation with $k_{eff} \geq 1$, the limits on core reactivity must be maintained 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 shutdown and changes to core reactivity due to fuel depletion cannot occur.

In MODE 6, fuel loading results in a continually changing core reactivity. Boron concentration requirements (LCO 3.9.1, "Refueling Boron Concentration") ensure that fuel movements are performed within the bounds of the safety analysis, and an SDM demonstration is required during the first startup following operations that could have altered core reactivity (e.g., fuel movement or CONTROL ROD replacement or shuffling).

ACTIONS

A.1 and A.2

Should an anomaly develop between measured and predicted core reactivity, an evaluation of the core design and safety analysis must be performed. Core conditions are evaluated to determine their consistency with input to design

(continued)

ACTIONS

A.1 and A.2 (continued)

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 72 hours 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 may be renormalized, and power operation may continue. If operational restrictions or additional surveillance requirements are necessary to ensure the reactor core is acceptable for continued operation, then they must be defined.

The required Completion Time of 72 hours is adequate for preparing operating restrictions or surveillances that may be required to allow continued reactor operation.

B.1

If the core reactivity cannot be restored to within the 1% $\Delta k/k$ limit, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours. If the SDM for MODE 3 is not met, then boration required by Required Action A.1 of LCO 3.1.1 would occur. The allowed

(continued)

BASES

ACTIONS

B.1 (continued)

Completion Time of 6 hours is reasonable, based on operating experience to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1

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 positions, moderator temperature, fuel temperature, fuel depletion, xenon concentration, and samarium concentration. The Surveillance is performed prior to entering MODE 1 as an initial check on core conditions and design calculations at BOC. A Note is included in the SR to indicate that the normalization of predicted core reactivity to the measured value must take place within the first 60 effective full power days (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 reactivity changes due to fuel depletion and the presence of other indicators (QPT, etc.) for prompt indication of an anomaly. Another Note is included in the SRs to indicate that the performance of the Surveillance is not required for entry into MODE 2.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26, GDC 28, and GDC 29.

2. FSAR, Chapter [14].

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. The same characteristic is true when the MTC is positive and coolant temperature decreases occur.

MTC values are predicted at selected burnups during the safety evaluation analysis and are confirmed to be acceptable by measurements. Both initial and reload cores are designed so that the beginning of cycle (BOC) MTC is less than zero when THERMAL POWER is 95% RTP or greater. The actual value of the MTC is dependent on core characteristics, such as fuel loading and reactor coolant soluble boron concentration. The core design may require additional burnable absorbers 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 the MTC does not exceed the EOC limit.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

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

The acceptance criteria for the specified MTC are:

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

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

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

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

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

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

MTC satisfies Criterion 2 of the NRC Policy Statement.

LCO

LCO 3.1.3 requires the MTC to be within specified limits in the COLR to ensure the core operates within the assumptions of the accident analysis. During the reload core safety evaluation, the MTC is analyzed to determine that its values remain within the bounds of the original accident analysis during operation. The LCO establishes a maximum positive value that can not be exceeded. The limit of +0.9E-4 (% $\Delta k/k$)/°F on positive MTC, when THERMAL POWER is < 95% RTP, ensures that core overheating accidents will not violate the accident analysis assumptions. The requirement for a negative MTC, when THERMAL POWER is ≥ 95% RTP, ensures that core operation will be stable. The negative MTC limit for EOC specified in the COLR ensures that core overcooling accidents will not violate the accident analysis assumptions.

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

APPLICABILITY

In MODE 1, the limits on MTC must be maintained to ensure that any accident initiated from THERMAL POWER operation will not violate the design assumptions of the accident analysis. In MODE 2, the limits must also be maintained to ensure that startup and subcritical accidents, such as the uncontrolled CONTROL ROD assembly or group withdrawal, will not violate the assumptions of the accident analysis. In MODES 3, 4, 5, and 6, this LCO is not applicable, since no Design Basis Accidents (DBAs) using the MTC as an analysis

(continued)

BASES

APPLICABILITY (continued)

assumption are initiated from these MODES. However, the variation of MTC with temperature in MODES 3, 4, and 5 for DBAs initiated in MODES 1 and 2 is accounted for in the subject accident analysis. The variation of MTC with temperature assumed in the safety analysis, is accepted as valid once the BOC and middle of cycle measurements are used for normalization.

ACTIONS

A.1

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

SURVEILLANCE REQUIREMENTS

The following two SRs for measurement of the MTC at the beginning and end of each fuel cycle provide for confirmation of the limiting MTC values. The MTC changes slowly from most positive (least negative) to most negative value during fuel cycle operation, as the RCS boron concentration is reduced with fuel depletion.

SR 3.1.3.1

The requirement for measurement, prior to initial operation above 5% RTP, satisfies the confirmatory check on the most positive (least negative) MTC value.

SR 3.1.3.2

The requirement for measurement, within 7 effective full power days (EFPD) after reaching an equilibrium boron concentration of 300 ppm for RTP, satisfies the confirmatory

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.3.2 (continued)

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

The SR is modified by two Notes. Note 1 indicates performance of SR 3.1.3.2 is not required prior to entering MODE 1 or 2. Although this Surveillance is applicable in MODES 1 and 2, the reactor must be critical before the Surveillance can be completed. Therefore, entry into the applicable MODE, prior to accomplishing the Surveillance, is necessary.

Note 2 indicates that SR 3.1.3.2 may be repeated, and shutdown must occur, prior to exceeding the minimum allowable boron concentration at which MTC is projected to exceed the lower limit. The minimum allowable boron concentration is obtained from the EOC MTC versus boron concentration slope with appropriate conservatisms. Thus, the projected EOC MTC is evaluated before the lower limit is actually reached.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 11.
2. FSAR, Chapter [14].
3. FSAR, Section [ ].
4. FSAR, Section [ ].

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.4 CONTROL ROD Group Alignment Limits

BASES

BACKGROUND

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

The applicable criteria for these design requirements are 10 CFR 50, Appendix A, GDC 10, "Reactor Design," and 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 within design power peaking limits and the core design requirement of a minimum SDM.

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

CONTROL RODS are moved by their CONTROL ROD drive mechanisms (CRDMs). Each CRDM moves its rod ¾ inch for one revolution of the leadscrew, but at varying rates depending on the signal output from the Control Rod Drive Control System (CRDCS).

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

(continued)

BASES

BACKGROUND (continued)

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

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

The relative position indicator transducer is a potentiometer that is driven by electrical pulses from the CRDCS. There is one counter for each CONTROL ROD drive. Individual rods in a group all receive the same signal to move; therefore, the counters for all rods in a group should indicate the same position. The Relative Position Indicator System is considered highly precise (one rotation of the leadscrew is ¾ inch in rod motion). If a rod does not move for each demand pulse, the counter will still count the pulse and incorrectly reflect the position of the rod.

The Absolute Position Indicator System provides a highly accurate indication of actual CONTROL ROD position, but at a lower precision than relative position indicators. This system is based on inductive analog signals from a series of reed switches spaced along a tube with a center to center distance of 3.75 inches.

APPLICABLE SAFETY ANALYSES

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

a. There shall be no violations of:

   1. specified acceptable fuel design limits, or
   2. Reactor Coolant System (RCS) pressure boundary damage; and

b. The core must remain subcritical after accident transients.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

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

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

Continued operation of the reactor with a misaligned or dropped CONTROL ROD is allowed if the $F_Q(Z)$ and the $F^N_{\Delta H}$ are verified to be within their limits in the COLR. When a CONTROL ROD is misaligned, the assumptions that are used to determine the regulating rod insertion limits, APSR insertion limits, AXIAL POWER IMBALANCE limits, and QPT limits are not preserved. Therefore, the limits may not preserve the design peaking factors, and $F_Q(Z)$ and $F^N_{\Delta H}$ must be verified directly by incore mapping. Bases Section 3.2, Power Distribution Limits, contains a more complete discussion of the relation of $F_Q(Z)$ and $F^N_{\Delta H}$ to the operating limits.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The CONTROL ROD group alignment limits satisfy Criterion 2 of the NRC Policy Statement.

LCO

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

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

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

APPLICABILITY

The requirements on CONTROL ROD OPERABILITY and alignment are applicable in MODES 1 and 2 because these are the only MODES in which neutron (or fission) power is generated, and the OPERABILITY (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 bottomed, and the reactor is shut down and not producing fission power. In the shutdown MODES, the OPERABILITY of the safety and regulating rods has the potential to affect the required SDM, but this effect can be compensated for by an increase in the boron concentration of the RCS. See LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," for SDM in MODES 3, 4, and 5, and

(continued)

BASES

APPLICABILITY (continued)

LCO 3.9.1, "Boron Concentration," for boron concentration requirements during refueling.

ACTIONS

A.1

Alignment of the inoperable or misaligned CONTROL ROD may be accomplished by either moving the single CONTROL ROD to the group average position, or by moving the remainder of the group to the position of the single inoperable or misaligned CONTROL ROD. Either action can be used to restore the CONTROL RODS to a radially symmetric pattern. However, this must be done without violating the CONTROL ROD group sequence, overlap, and insertion limits of LCO 3.2.1, "Regulating Rod Insertion Limits," given in the COLR. THERMAL POWER must also be restricted, as necessary, to the value allowed by the insertion limits of LCO 3.2.1. The required Completion Time of 1 hour is acceptable because local xenon redistribution during this short interval will not cause a significant increase in LHR. This option is not available if a safety rod is misaligned, since the limits of LCO 3.1.5, "Safety Rod Insertion Limits," would be violated.

A.2.1.1

Compliance with Required Actions A.2.1.1 through A.2.5 allows for continued power operation with one CONTROL ROD inoperable but trippable, or misaligned from its group average position. These Required Actions comprise the final alternate for Condition A.

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

(continued)

BASES

ACTIONS (continued)

A.2.1.2

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

A.2.2

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

A.2.3

Reduction of the nuclear overpower trip setpoint to ≤ 70% ALLOWABLE THERMAL POWER, after THERMAL POWER has been reduced to 60% ALLOWABLE THERMAL POWER, maintains both core protection and an operating margin at reduced power similar to that at RTP. The required Completion Time of 10 hours allows the operator 8 additional hours after completion of the THERMAL POWER reduction in Required Action A.2.2 to adjust the trip setpoint.

A.2.4

The existing CONTROL ROD configuration must not cause an ejected rod to exceed the limit of 0.65% $\Delta k/k$ at RTP or 1.00% $\Delta k/k$ at zero power (Ref. 6). This evaluation may require a computer calculation of the maximum ejected rod worth based on nonstandard configurations of the CONTROL ROD groups. The evaluation must determine the ejected rod worth for the remainder of the fuel cycle to ensure a valid

(continued)

BASES

ACTIONS

A.2.4 (continued)

evaluation, should fuel cycle conditions at some later time become more bounding than those at the time of the rod misalignment. The required Completion Time of 72 hours is acceptable because LHRs are limited by the THERMAL POWER reduction and sufficient time is provided to perform the required evaluation.

A.2.5

Performance of SR 3.2.5.1 provides a determination of the power peaking factors using the Incore Detector System. Verification of the $F_Q(Z)$ and $F^N_{\Delta H}$ from an incore power distribution map is necessary to ensure that excessive local LHRs will not occur due to CONTROL ROD misalignment. This is necessary because the assumption that all CONTROL RODS are aligned (used to determine the regulating rod insertion, AXIAL POWER IMBALANCE, and QPT limits) is not valid when the CONTROL RODS are not aligned. The required Completion Time of 72 hours is acceptable because LHRs are limited by the THERMAL POWER reduction and adequate time is allowed to obtain an incore power distribution map.

B.1

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

C.1.1

More than one trippable CONTROL ROD becoming inoperable or misaligned, or both inoperable but trippable and misaligned from their group average position, is not expected and may violate the minimum SDM requirement. Therefore, SDM must be evaluated. Ensuring the SDM meets the minimum requirement

(continued)

BASES

ACTIONS

C.1.1 (continued)

within 1 hour allows the operator adequate time to determine the SDM.

C.1.2

Restoration of the required SDM requires increasing the RCS boron concentration to provide negative reactivity. RCS boration must occur as described in Bases Section 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 for aligning the required valves and starting the boric acid pumps. Boration will continue until the required SDM is restored.

C.2

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

Therefore, it is prudent to place the reactor in MODE 3. LCO 3.1.4 does not apply in MODE 3 since excessive power peaking cannot occur and the minimum required SDM is ensured. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES

ACTIONS (continued)

D.1.1 and D.1.2

When one or more rods are untrippable, the 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 to restore SDM.

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

D.2

If the untrippable 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 plant 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 rods are aligned within [6.5]% of their group average height limits at a 12 hour Frequency allows the operator to detect a rod that is beginning to deviate from its expected position. If the asymmetric CONTROL ROD alarm is inoperable, a Frequency of 4 hours is reasonable to prevent large deviations in CONTROL ROD alignment from occurring without detection. The specified Frequency takes into account other rod position information that is continuously available to the operator in the control room, so that during actual rod motion, deviations can immediately be detected.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.4.2

Verifying each CONTROL ROD is OPERABLE would require that each rod be tripped. However, in MODES 1 and 2, tripping each CONTROL ROD could result in radial tilts. Exercising each individual CONTROL ROD every 92 days provides increased confidence that all rods continue to be OPERABLE without exceeding the alignment limit, even if they are not regularly tripped. Moving each CONTROL ROD by 3% 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 required performances of SR 3.1.4.2 (determination of CONTROL ROD OPERABILITY by movement), if a CONTROL ROD(S) is discovered to be immovable, but is determined to be trippable and aligned, the CONTROL ROD(S) is considered to be OPERABLE. At any time, if a CONTROL ROD(S) is immovable, a determination of the trippability (OPERABILITY) of the CONTROL ROD(S) must be made, and appropriate action taken.

SR 3.1.4.3

Verification of rod drop time 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. The rod drop time given in the safety analysis is 1.4 seconds to ⅔ insertion. Using the identical rod drop curve gives a value of [1.66] seconds to ¾ insertion. The latter value is used in the Surveillance because the zone reference lights are located at 25% insertion intervals. The zone reference lights will activate at ¾ insertion to give an indication of the rod drop time and rod location. Measuring rod drop times, prior to reactor criticality after reactor vessel head removal and after CONTROL ROD drive system maintenance or modification, ensures that the reactor internals and CRDM will not interfere with CONTROL ROD motion or rod drop time. 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.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.4.3 (continued)

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

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10 and GDC 26.
2. 10 CFR 50.46.
3. FSAR, Chapter [14].
4. FSAR, Section [ ].
5. FSAR, Section [ ].
6. FSAR, Section [ ].

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.5 Safety Rod Insertion Limit

BASES

BACKGROUND

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

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

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

The safety groups can be fully withdrawn without the core going critical. This provides available negative reactivity in the event of boration errors. The safety groups are controlled manually by the control room operator. During normal full power operation, the safety groups are fully withdrawn. The safety groups must be completely withdrawn from the core prior to withdrawing any regulating groups

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BASES

BACKGROUND (continued)

during an approach to criticality. The safety groups remain in the fully withdrawn position until the reactor is shut down. They add negative reactivity to shut down the reactor upon receipt of a reactor trip signal.

APPLICABLE SAFETY ANALYSES

On a reactor trip, all rods (safety groups and regulating groups), except the most reactive rod, are assumed to insert into the core. The safety groups shall be at their fully withdrawn limits and available to insert the maximum amount of negative reactivity on a reactor trip signal. The regulating groups may be partially inserted in the core as allowed by LCO 3.2.1, "Regulating Rod Insertion Limits." The safety group and regulating rod insertion limits are established to ensure that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM (see LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") following a reactor trip from full power. The combination of regulating groups and safety groups (less the most reactive rod, which is assumed to be fully withdrawn) is sufficient to take the reactor from full power conditions at rated temperature to zero power and to maintain the required SDM at rated no load temperature (Ref. 3). The safety group insertion limit also limits the reactivity worth of an ejected safety rod.

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

a. There shall be no violations of:

    1. specified acceptable fuel design limits, or
    2. RCS pressure boundary integrity; and

b. The core must remain subcritical after accident transients.

The safety rod insertion limits satisfy Criteria 2 and 3 of the NRC Policy Statement.

(continued)

BASES (continued)

LCO

The safety groups must be fully withdrawn any time the reactor is critical or approaching criticality. This ensures that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM following a reactor trip.

APPLICABILITY

The safety groups must be within their insertion limits with the reactor in MODES 1 and 2. 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. Refer to LCO 3.1.1 for SDM requirements in MODES 3, 4, and 5. LCO 3.9.1, "Boron Concentration," ensures adequate SDM in MODE 6.

This LCO has been modified by a Note indicating the LCO requirement is suspended during SR 3.1.4.2. This SR verifies the freedom of the rods to move, and requires the safety group to move below the LCO limits, which would normally violate the LCO.

ACTIONS

A.1, A.2.1.1, A.2.1.2, and A.2.2

When one safety rod is not fully withdrawn, 1 hour is allowed to fully withdraw the rod. This is necessary because the available SDM may be reduced with one of the safety rods not within insertion limits.

Alternatively, the rod may be declared inoperable within the same 1 hour time frame. This requires entry into LCO 3.1.4, "CONTROL ROD Group Alignment Limits." In addition, since the rod may be inserted farther than the group average insertion for a long time, SDM must be evaluated. Ensuring the SDM meets the minimum requirement within 1 hour is adequate to determine that further degradation of the SDM is not occurring.

Restoration of the required SDM requires increasing the boron concentration, since the CONTROL ROD may remain misaligned and not be providing its normal negative reactivity on tripping. RCS boration must occur as described in Bases Section 3.1.1. The required Completion Time of 1 hour for initiating boration is reasonable, based

(continued)

BASES

ACTIONS

A.1, A.2.1.1, A.2.1.2, and A.2.2 (continued)

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 for aligning the required valves and starting the boric acid pumps. Boration will continue until the required SDM is restored.

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

B.1.1 and B.1.2

When more than one safety rod is 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 to restore SDM.

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

B.2

If more than one safety rod is inoperable 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 each safety rod is fully withdrawn ensures the rods are available to provide reactor shutdown capability.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.5.1 (continued)

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

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10 and GDC 26.
2. 10 CFR 50.46.
3. FSAR, Section [ ].

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.6 AXIAL POWER SHAPING ROD (APSR) Alignment Limits

BASES

BACKGROUND

The OPERABILITY of the APSRs and rod misalignment are initial condition assumptions in the safety analysis that directly affect core power distributions. The applicable criteria for these power distribution design requirements are 10 CFR 50, Appendix A, GDC 10, "Reactor Design," and GDC 26, "Reactivity Limits" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 2).

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

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

CONTROL RODS and APSRs are moved by their CONTROL ROD drive mechanisms (CRDMs). Each CRDM moves its rod ¾ inch for one revolution of the leadscrew at varying rates depending on the signal output from the Rod Control System.

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

LCO 3.1.6 is conservatively based on use of black (Ag-In-Cd) APSRs and bounds use of gray (Inconel) APSRs. The reactivity worth of black APSRs is greater than that of gray APSRs; thus the impact of black APSR misalignment on the core power distribution is greater.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

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

a. Specified acceptable fuel design limits; and

b. Reactor Coolant System (RCS) pressure boundary integrity.

Two types of misalignment or inoperability are distinguished. During movement of an APSR group, one rod may stop moving while the other rods in the group continue. This condition may cause excessive power peaking. The second type of misalignment occurs when one rod drops partially or fully into the reactor core. This event causes an initial power reduction, followed by a return towards the original power, due to positive reactivity feedback from the negative moderator temperature coefficient. Increased peaking during the power increase may result in excessive local linear heat rates (LHRs). The accident analysis and reload safety evaluations define APSR insertion limits that ensure that if an APSR is stuck in or dropped in, the increase in local LHR is within the design limits. The Required Action statement in the LCO provides a conservative approach to ensure that continued operation remains within the bounds of the safety analysis (Ref. 4).

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

The APSR alignment limits satisfy Criterion 2 of the NRC Policy Statement.

LCO

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

The limit for individual APSR misalignment is [6.5]% (9 inches) deviation from the group average position. This

(continued)

BASES

LCO (continued)

value is established based on the distance between reed switches, with additional allowances for uncertainty in the absolute position indicator amplifiers, group maximum or minimum synthesizer, and asymmetric alarm or fault detector outputs. The position of an inoperable rod is not included in the calculation of the rod group's average position.

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

APPLICABILITY

The requirements on APSR OPERABILITY and alignment are applicable in MODES 1 and 2, when the APSRs are not fully withdrawn because these are the only MODES in which neutron (or fission) power is generated, and the OPERABILITY and alignment of rods have the potential to affect the safety of the plant. OPERABILITY and alignment of the APSRs are not required when they are fully withdrawn because they do not influence core power peaking. In MODES 3, 4, 5, and 6, the alignment limits do not apply because the reactor is shut down and not producing fission power, and excessive local LHRs cannot occur from APSR misalignment.

ACTIONS

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

A.1 and A.2

An alternate to realigning a single misaligned APSR to the group average position is to align the remainder of the APSR group to the position of the misaligned or inoperable APSR, while maintaining APSR insertion, in accordance with the limits in the COLR. This restores the alignment requirements. Deviations up to 2 hours will not cause significant xenon redistribution to occur. Required Action A.1 assumes the APSR group movement does not cause the limits of LCO 3.2.2, "AXIAL POWER SHAPING ROD (APSR) Insertion Limits," to be exceeded. For this reason,

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

Required Action A.1 is only practical for instances where small movements of the APSR group are sufficient to re-establish APSR alignment.

The reactor may continue in operation with the APSR misaligned if further movement of the APSR group is prohibited, so that the misalignment does not increase and cause the limits on AXIAL POWER IMBALANCE to be exceeded. The required Completion Time of up to 2 hours will not cause significant xenon redistribution to occur.

B.1

The plant must be brought to a MODE in which the LCO does not apply if the Required Actions and associated Completion Times cannot be met. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours. The Completion Time of 6 hours is reasonable, based on operating experience, for reaching MODE 3 from RTP in an orderly manner and without challenging plant systems. In MODE 3, APSR group alignment limits are not required because the reactor is not generating THERMAL POWER and excessive local LHRs cannot occur from APSR misalignment.

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1

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

(continued)

BASES (continued)

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10 and GDC 26.
2. 10 CFR 50.46.
3. FSAR, Section [ ].
4. FSAR, Section [ ].

B 3.1 REACTIVITY CONTROL

B 3.1.7 Position Indicator Channels

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 CONTROL ROD and APSR position indicators, and thereby ensure compliance with the CONTROL ROD and APSR alignment and insertion limits.

The OPERABILITY, including position indication, of the safety and regulating rods is an initial condition assumption in all safety analyses that assume rod insertion upon reactor trip. Maximum rod misalignment for the safety rods, regulating rods, and APSRs is assumed in the safety analysis, which directly affect core power distributions and assumptions of available SDM.

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

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

Two methods of CONTROL ROD and APSR position indication are provided in the CONTROL ROD Drive Control System. The two means are by absolute position indicator and relative position indicator transducers. The absolute position indicator transducer consists of a series of magnetically operated reed switches mounted in a tube parallel to the CONTROL ROD drive mechanism (CRDM) motor tube extension.

(continued)

BASES

BACKGROUND (continued)

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

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

CONTROL ROD position indicating readout devices located in the control room consist of single CRA position meters on a wall mounted position indication panel and four group average position meters on the console. A selector switch permits either relative or absolute position indication to be displayed on all of the single rod meters. Indicator lights are provided on the single CRA meter panel to indicate when each CRA is fully withdrawn, fully inserted,

(continued)

BASES

BACKGROUND (continued)

enabled, or transferred, and whether a CRA position asymmetry alarm condition is present. Indicators on the console show full insertion, full withdrawal, and enabled for motion for each CONTROL ROD group. Identical instrumentation and devices exist for the APSR group. The consequence of continued operation with an inoperable absolute position indicator or relative position indicator channel is a decreased reliability in determining CONTROL ROD position. Therefore, the potential for operation in violation of design peaking factors or SDM is increased.

APPLICABLE SAFETY ANALYSES

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

LCO

LCO 3.1.7 specifies that one absolute position indicator channel and one relative position indicator channel be OPERABLE for each CONTROL ROD and APSR.

The agreement between the relative position indicator channel and the absolute position indicator channel, within the limit given in the COLR, indicates that relative position indicators are adequately calibrated and can be

(continued)

BASES

LCO (continued)

used for indication of the measurement of CONTROL ROD group position. A deviation of less than the allowable limit, given in the COLR, in position indication for a single CONTROL ROD or APSR, ensures confidence that the position uncertainty of the corresponding CONTROL ROD group or APSR group is within the assumed values used in the analysis that specifies CONTROL ROD group and APSR insertion limits.

These requirements ensure that CONTROL ROD position indication during power operation and PHYSICS TESTS is accurate, and that design assumptions are not challenged. OPERABILITY of the position indicator channels ensures that inoperable, misaligned, or mispositioned CONTROL RODS or APSRs can be detected. Therefore, power peaking and SDM can be controlled within acceptable limits.

APPLICABILITY

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

ACTIONS

A.1

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

(continued)

BASES

ACTIONS (continued)

B.1.1

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

B.1.2

To allow continued operation, the rods with inoperable absolute position indicator channels are maintained at the zone reference indicator position. In addition, the affected rods are maintained within the limits of LCO 3.1.5 (when the affected rod is a safety rod); LCO 3.2.1 (when the affected rod is a regulating rod); or LCO 3.2.2 (when the affected rod is an APSR). This Required Action ensures safety rods remain fully withdrawn, and that regulating rods and APSRs remain aligned within their insertion limits. The required Completion Time of 8 hours is reasonable for allowing the operator adequate time to determine the affected rods are in compliance with these LCOs. Continuing to verify the rod positions every 8 hours thereafter is reasonable for ensuring that rod alignment and insertion are not changing, and provides the operator adequate time to correct any deviation that may occur. Continuing the verification every 8 hours thereafter in the applicable condition is acceptable, based on the fact that during normal power operation excessive movement of the groups is not required. Also, if the rod is out of position during this 8 hour period, the simultaneous occurrence of an event sensitive to the rod position has a small probability.

(continued)

BASES

ACTIONS (continued)

B.2.1

If the absolute position indicator is inoperable for one or more rods, the position of the rod is monitored by the relative position indicator channel for each affected rod. However, the relative position indicator channel is not as reliable a method of monitoring rod position as the absolute position indicator because it counts electrical pulse steps. The fixed incore system can be used to indirectly determine the absolute position of the affected rod. The fixed incore instrumentation can provide a continual update of CONTROL ROD position, therefore this method can be used to allow continued operation of the reactor with a manual CONTROL ROD movement, while maintaining verification of CONTROL ROD insertion and alignment. Required Action B.2.1. restricts rod motion by placing the groups with nonindicating rods in manual control; thus, even if the rod fails to move in alignment with the group, misalignment is limited. The required Completion Time of 8 hours provides the operator adequate time for placing the rods in manual control, and is consistent with the required Completion Time for Required Action B.1.1. If the rod is out of position during this 8 hour period, the simultaneous occurrence of an event sensitive to the rod position has a small probability.

B.2.2

Continuing to verify the rod positions every 8 hours is reasonable for ensuring that rod alignment and insertion are not changing, and provides the operator adequate time to correct any deviation that may occur. The additional Completion Time of 1 hour after motion of nonindicating rods, which exceeds 15 inches in one direction since the last determination of the rod's position, ensures that the rod with inoperable position indication will not be misaligned for a significant period of time, in the event the rod is moved. The specified Completion Times are acceptable because the simultaneous occurrence of a mispositioned rod and an event sensitive to the rod position has a small probability.

(continued)

BASES

ACTIONS (continued)

C.1

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

SURVEILLANCE REQUIREMENTS

SR 3.1.7.1

Verification is required that the Absolute Position Indicator channels and Relative Position Indicator channels agree within the limit given in the COLR. This verification ensures that the Relative Position Indicator channels, which are regarded as the potentially less reliable means of position indication, remain OPERABLE and accurate. The required Frequency of 12 hours is adequate for verifying that no degradation in system OPERABILITY has occurred. If the asymmetric CONTROL ROD alarm is inoperable, then the Surveillance is performed every 4 hours. This required Frequency is adequate for ensuring that the CONTROL RODS and APSRs do not exceed their alignment limits.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 13.
2. FSAR, Section [14.1.2.2], Section [14.1.2.3], Section [14.1.2.6], Section [14.1.2.7], Section [14.2.2.4], and Section [14.2.2.5].

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.8 PHYSICS TESTS Exceptions Systems—MODE 1

BASES

BACKGROUND

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

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

a. 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, low power operations, and power ascension; at high powers; and after each fueling. 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 (Ref. 4).

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

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BASES

BACKGROUND (continued)

execution of testing required to ensure the design intent is met. PHYSICS TESTS are performed in accordance with these procedures, and test results are approved prior to continued power escalation and long term power operation. Examples of PHYSICS TESTS include determination of critical boron concentration, CONTROL ROD group worths, reactivity coefficients, flux symmetry, and core power distribution.

APPLICABLE SAFETY ANALYSES

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

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

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

LCO 3.1.4, "CONTROL ROD Group Alignment Limits";
LCO 3.1.5, "Safety Rod Insertion Limits";
LCO 3.1.6, "AXIAL POWER SHAPING ROD (APSR) Alignment Limits";
LCO 3.2.1, "Regulating Rod Insertion Limits," for the restricted operation region only;
LCO 3.2.3, "AXIAL POWER IMBALANCE Operating Limits"; or
LCO 3.2.4, "QUADRANT POWER TILT (QPT)"

are suspended for PHYSICS TESTS, the fuel design criteria are preserved by maintaining the nuclear hot channel factors (in MODE 1 PHYSICS TESTS) within their limits, maintaining ejected rod worth within limits by restricting regulating rod insertion to within the acceptable operating region or the restricted operating region, by limiting maximum THERMAL POWER and by maintaining SDM $\geq 1.0\% \Delta k/k$. Therefore,

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

surveillance of the $F_Q(Z)$, the $F^N_{\Delta H}$, and SDM is required to verify that their limits are not exceeded. The limits for the nuclear hot channel factors are specified in the COLR. Refer to the Bases for LCO 3.2.5 for a complete discussion of $F_Q(Z)$ and $F^N_{\Delta H}$. During PHYSICS TESTS, one or more of the LCOs that normally preserve the $F_Q(Z)$ and $F^N_{\Delta H}$ limits may be suspended. However, the results of the safety analysis are not adversely impacted if verification that $F_Q(Z)$ and $F^N_{\Delta H}$ are within their limits is obtained, while one or more of the LCOs is suspended. Therefore, SRs are placed on $F_Q(Z)$ and $F^N_{\Delta H}$ during MODE 1 PHYSICS TESTS to verify that these factors remain within their limits. Periodic verification of these factors allows PHYSICS TESTS to be conducted while continuing to maintain the design criteria.

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

PHYSICS TESTS satisfy Criteria 1, 2, and 3 of the NRC Policy Statement.

LCO

This LCO permits individual CONTROL RODS to be positioned outside of their specified group alignment and withdrawal limits and to be assigned to other than specified CONTROL ROD groups, and permits AXIAL POWER IMBALANCE and QPT limits to be exceeded during the performance of PHYSICS TESTS. In addition, this LCO permits verification of the fundamental core characteristics and nuclear instrumentation operation.

The requirements of LCO 3.1.4, LCO 3.1.5, LCO 3.1.6, LCO 3.2.1 (for the restricted operation region only), LCO 3.2.3, and LCO 3.2.4 may be suspended during the performance of PHYSICS TESTS provided:

a. THERMAL POWER is maintained ≤ 85% RTP;

(continued)

BASES

LCO (continued)

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

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

d. SDM is maintained $\geq$ 1.0% $\Delta k/k$.

Operation with THERMAL POWER $\leq$ 85% RTP during PHYSICS TESTS provides an acceptable thermal margin when one or more of the applicable LCOs is out of specification. Eighty-five percent RTP is consistent with the maximum power level for conducting the intermediate core power distribution test specified in Reference 4. The nuclear overpower trip setpoint is reduced so that a similar margin exists between the steady state condition and trip setpoint as exists during normal operation at RTP.

APPLICABILITY

This LCO is applicable in MODE 1, when the reactor has completed low power testing and is in power ascension, or during power operation with THERMAL POWER > 5% RTP but $\leq$ 85% RTP. This LCO is applicable for power ascension testing, as defined by Regulatory Guide 1.68 (Ref. 3). In MODE 2, Applicability of this LCO is not required because LCO 3.1.9, "PHYSICS TESTS Exceptions—MODE 2," addresses PHYSICS TESTS exceptions in MODE 2. In MODES 3, 4, 5, and 6, Applicability is not required because PHYSICS TESTS are not performed in these MODES.

ACTIONS

A.1 and A.2

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. The operator should begin boration with the best source available for the plant conditions. Boration will be continued until SDM is within limit. In the determination of the required combination of boration flow rate and boron concentration, there is no unique requirement that must be satisfied.

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

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

B.1

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

If the nuclear overpower trip setpoint is not within the specified limits, then 1 hour is allowed for the operator to restore the nuclear overpower trip setpoint within limits or to complete an orderly suspension of PHYSICS TESTS exceptions. Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable individual LCOs to within specification. This required Completion Time is consistent with, or more conservative than, those specified for the individual LCO, addressed by these PHYSICS TESTS exceptions.

If the results of the incore flux map indicate that either $F_Q(Z)$ or $F^N_{\Delta H}$ has exceeded its limit, then PHYSICS TESTS are suspended. This action is required because of direct indication that the core peaking factors, which are fundamental initial conditions for the safety analysis, are excessive. Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable LCOs to within specification.

SURVEILLANCE REQUIREMENTS

SR 3.1.8.1

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

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.8.1 (continued)

determine any degradation of the established thermal margin during PHYSICS TESTS.

SR 3.1.8.2

Verification that $F_Q(Z)$ and $F^N_{\Delta H}$ are within their limits ensures that core local linear heat rate and departure from nucleate boiling ratio will remain within their limits, while one or more of the LCOs that normally control these design limits are out of specification. The required Frequency of 2 hours allows the operator adequate time for collecting a flux map and for performing the hot channel factor verifications, based on operating experience. If SR 3.2.5.1 is not met, PHYSICS TESTS are suspended and LCO 3.2.5 applies. This Frequency is more conservative than the Completion Time for restoration of the individual LCOs that preserve the $F_Q(Z)$ and $F^N_{\Delta H}$ limits.

SR 3.1.8.3

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

SR 3.1.8.4

The SDM is verified by performing a reactivity balance calculation, considering the following reactivity effects:

a. Reactor Coolant System (RCS) boron concentration;

b. CONTROL ROD position;

c. RCS average temperature;

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.8.4 (continued)

d. Fuel burnup based on gross thermal energy generation;

e. Xenon concentration; and

f. Isothermal temperature coefficient (ITC).

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

The Frequency of 24 hours is based on the generally slow change in required boron concentration and 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. ANSI/ANS-19.6.1-1985, December 13, 1985.

5. FSAR, Section [13.4.8].

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

B 3.1 REACTIVITY CONTROL

B 3.1.9 PHYSICS TESTS Exceptions—MODE 2

BASES

BACKGROUND

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

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

a. 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, low power operations, and power ascension; at high powers; 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 (Ref. 4).

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

(continued)

BASES

BACKGROUND (continued)

execution of testing required to ensure that the design intent is met. PHYSICS TESTS are performed in accordance with these procedures, and test results are approved prior to continued power escalation and long term power operation.

Examples of MODE 2 PHYSICS TESTS include determination of critical boron concentration, CONTROL ROD group worth, and reactivity coefficients.

APPLICABLE SAFETY ANALYSES

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

It is acceptable to suspend the following LCOs for PHYSICS TESTS because reactor protection criteria are preserved by the LCOs still maintained and by the SRs:

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

Even if an accident occurs during PHYSICS TESTS with one or more LCOs suspended, fuel damage criteria are preserved because the limits on THERMAL POWER and shutdown capability are maintained during the PHYSICS TESTS.

Shutdown capability is preserved by limiting maximum obtainable THERMAL POWER and maintaining adequate SDM, when in MODE 2 PHYSICS TESTS. In MODE 2, the Reactor Coolant System (RCS) temperature must be within the narrow range instrumentation for plant control. The narrow range temperature instrumentation goes on scale at 520°F. Therefore, it is considered safe to allow the minimum RCS

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

temperature to decrease to 520°F during MODE 2 PHYSICS TESTS, based on the low probability of an accident occurring and on prior operating experience.

PHYSICS TESTS include measurement of core nuclear parameters or exercise of control components that affect process variables.

PHYSICS TESTS satisfy Criteria 1, 2, and 3 of the NRC Policy Statement.

LCO

This LCO permits individual CONTROL RODS to be positioned outside of their specified group alignment and withdrawal limits and to be assigned to other than specified CONTROL ROD groups during the performance of PHYSICS TESTS. In addition, this LCO permits verification of the fundamental core characteristics.

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

a. THERMAL POWER is ≤ 5% RTP;

b. Nuclear overpower trip setpoints on the OPERABLE nuclear power range channels are set to ≤ 25% RTP;

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

d. SDM is maintained ≥ [1.0]% $\Delta k/k$.

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

APPLICABILITY

This LCO is applicable in MODE 2 when the reactor is either not critical or when THERMAL POWER is ≤ 5% RTP. This LCO is applicable for initial criticality or low power testing, as defined by Regulatory Guide 1.68 (Ref. 3). In MODE 1,

(continued)

BASES

APPLICABILITY (continued)

Applicability of this LCO is not required because LCO 3.1.8, "PHYSICS TESTS Exceptions," addresses PHYSICS TESTS exceptions in MODE 1. In MODES 3, 4, 5, and 6, Applicability is not required because physics testing is not performed in these MODES.

ACTIONS

A.1

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

B.1 and B.2

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. The operator should begin boration with the best source available for the plant conditions. Boration will be continued until SDM is within limit. In the determination of the required combination of boration flow rate and boron concentration, there is no unique requirement that must be satisfied.

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

C.1

If the nuclear overpower trip setpoint is > 25% RTP, then 1 hour is allowed for the operator to restore the nuclear overpower trip setpoint within limits or to complete an orderly suspension of PHYSICS TESTS exceptions. Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable individual LCOs to within specification, in order to ensure that continuity of reactor operation is within initial condition limits. This required Completion

(continued)

BASES

ACTIONS

C.1 (continued)

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

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

SURVEILLANCE REQUIREMENTS

SR 3.1.9.1

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

SR 3.1.9.2

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

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.9.3

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

SR 3.1.9.4

The SDM is verified by performing a reactivity balance calculation, considering the following reactivity effects:

a. RCS boron concentration;

b. CONTROL ROD position;

c. RCS average temperature;

d. Fuel burnup based on gross thermal energy generation;

e. Xenon concentration; and

f. Isothermal temperature coefficient (ITC).

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

The Frequency of 24 hours is based on the generally slow change in required boron concentration and 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.

(continued)

BASES

REFERENCES (continued)

4. ANSI/ANS-19.6.1-1985, December 13, 1985.

5. FSAR, Section [13.4.8].

6. FSAR, Section [13.4.8], [Table 13-3 and Table 13-4].

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1 Regulating Rod Insertion Limits

BASES

BACKGROUND

The insertion limits of the regulating rods are initial condition assumptions used in all safety analyses that assume rod insertion upon reactor trip. The insertion limits directly affect the core power distributions, the worth of a potential ejected rod, the assumptions of available SDM, and the initial reactivity insertion rate.

The applicable criteria for these reactivity and power distribution design requirements are described in 10 CFR 50, Appendix A, GDC 10, "Reactor Design," GDC-26, "Reactivity Control System Redundancy-and Capability," GDC 28, "Reactivity Limits" (Ref. 1), and in 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plants" (Ref. 2).

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

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

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

The power density at any point in the core must be limited to maintain specified acceptable fuel design limits, including limits that ensure that the criteria specified in 10 CFR 50.46 (Ref. 2) are not violated. Together,

(continued)

BASES

BACKGROUND (continued)

LCO 3.2.1, "Regulating Rod Insertion Limits," LCO 3.2.2, "AXIAL POWER SHAPING ROD (APSR) Insertion Limits," LCO 3.2.3, "AXIAL POWER IMBALANCE Operating Limits," and LCO 3.2.4, "QUADRANT POWER TILT (QPT)," provide limits on control component operation and on monitored process variables to ensure that the core operates within the $F_Q(Z)$ and $F^N_{\Delta H}$ limits in the COLR. Operation within the $F_Q(Z)$ limits given in the COLR prevents power peaks that would exceed the loss of coolant accident (LOCA) limits derived from the analysis of the Emergency Core Cooling Systems (ECCS). Operation within the $F^N_{\Delta H}$ limits given in the COLR prevents departure from nucleate boiling (DNB) during a loss of forced reactor coolant flow accident. In addition to the $F_Q(Z)$ and $F^N_{\Delta H}$ limits, certain reactivity limits are met by regulating rod insertion limits. The regulating rod insertion limits also restrict the ejected CONTROL ROD worth to the values assumed in the safety analysis and maintain the minimum required SDM in MODES 1 and 2.

This LCO is required to minimize fuel cladding failures that breach the primary fission product barrier and release fission products into the reactor coolant in the event of a LOCA, loss of flow accident, ejected rod accident, or other postulated accidents requiring termination by a Reactor Protection System trip function.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation (Condition 1) or anticipated operational occurrences (Condition 2). The LCOs governing regulating rod insertion, APSR position, AXIAL POWER IMBALANCE, and QPT preclude core power distributions that violate the following fuel design criteria:

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

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

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

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

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

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

The SDM requirement is met by limiting the regulating and safety rod insertion limits such that sufficient inserted reactivity is available in the rods to shut down the reactor to hot zero power with a reactivity margin that assumes that the maximum worth rod remains fully withdrawn upon trip (Ref. 4). Operation at the SDM based regulating rod insertion limit may also indicate that the maximum ejected rod worth could be equal to the limiting value.

Operation at the regulating rod insertion limits may cause the local core power to approach the maximum linear heat generation rate or peaking factor with the allowed QPT present.

The regulating rod and safety rod insertion limits ensure that the safety analysis assumptions for SDM, ejected rod worth, and power distribution peaking factors remain valid (Refs. 3, 5, and 6).

The regulating rod insertion limits LCO satisfies Criterion 2 of the NRC Policy Statement.

LCO

The limits on CONTROL ROD sequence, including group overlap, and insertion positions as defined in the COLR, must be maintained because they ensure that the resulting power distribution is within the range of analyzed power distributions and that the SDM and ejected rod worth are maintained.

(continued)

BASES

LCO (continued)

The overlap between regulating groups provides more uniform rates of reactivity insertion and withdrawal and is imposed to maintain acceptable power peaking during regulating rod motion.

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

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

APPLICABILITY

The regulating rod sequence, overlap, and physical insertion limits shall be maintained with the reactor in MODES 1 and 2. These limits maintain the validity of the assumed power distribution, ejected rod worth, SDM, and reactivity rate insertion assumptions used in the safety analyses. Applicability in MODES 3, 4, and 5 is not required, because neither the power distribution nor ejected rod worth assumptions are exceeded in these MODES. SDM in MODES 3, 4, and 5 is governed by LCO 3.1.1, "SHUTDOWN MARGIN (SDM)."

LCO 3.1.1 has been modified by a Note that suspends the LCO requirement during the performance of SR 3.1.4.2, which verifies the freedom of the rods to move. This SR requires the regulating rods to move below the LCO limit, which normally violates the LCO.

ACTIONS

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

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BASES

ACTIONS (continued)

violated. The area between the boundaries of acceptable operation and unacceptable operation, illustrated on the regulating rod insertion limit figures in the COLR, is the restricted region. The actions required when operation occurs in the restricted region are described under Condition A. The actions required when operation occurs in the unacceptable region are described under Condition C.

A.1

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

For verification that $F_Q(Z)$ and $F^N_{\Delta H}$ are within their limits, SR 3.2.5.1 is performed using the Incore Detector System to obtain a three dimensional power distribution map. Verification that $F_Q(Z)$ and $F^N_{\Delta H}$ are within their limits ensures that operation with the regulating rods inserted into the restricted region does not violate the ECCS or DNB criteria (Ref. 7). The required Completion Time of 2 hours is acceptable in that it allows the operator sufficient time for obtaining a power distribution map and for verifying the power peaking factors. Repeating SR 3.2.5.1 every 2 hours is acceptable because it ensures that continued verification of the power peaking factors is performed as core conditions (primarily regulating rod insertion and induced xenon redistribution) change.

Monitoring the power peaking factors $F_Q(Z)$ and $F^N_{\Delta H}$ does not provide verification that the reactivity insertion rate on the rod trip or the ejected rod worth limit is maintained, because worth is a reactivity parameter rather than a power peaking parameter. However, if the COLR figures do not show

(continued)

BASES

ACTIONS

A.1 (continued)

that a rod insertion limit is ejected rod worth limited, then the ejected rod worth is no more limiting than the SDM based rod insertion limit in the core design (Ref. 8). Ejected rod worth limits are independently maintained by the Required Actions of Conditions A and C.

A.2

Indefinite operation with the regulating rods inserted in the restricted region, or in violation of the group sequence or overlap limits, is not prudent. Even if power peaking monitoring per Required Action A.1 is continued, reactivity limits may not be met and the abnormal regulating rod insertion or group configuration may cause an adverse xenon redistribution, may cause the limits on AXIAL POWER IMBALANCE to be exceeded, or may adversely affect the long term fuel depletion pattern. Therefore, power peaking monitoring is allowed for up to 24 hours after discovery of failure to meet the requirements of this LCO. This required Completion Time is reasonable based on the low probability of an event occurring simultaneously with the limit out of specification in this relatively short time period. In addition, it precludes long term depletion with abnormal group insertions or configurations, thereby limiting the potential for an adverse xenon redistribution.

B.1

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

(continued)

BASES

ACTIONS

B.1 (continued)

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

C.1

Operation in the unacceptable region shown on the figures in the COLR corresponds to power operation with an SDM less than the minimum required value or with the ejected rod worth greater than the allowable value. The regulating rods may be inserted too far to provide sufficient negative reactivity insertion following a reactor trip and the ejected rod worth may exceed its initial condition limit. Therefore, the RCS boron concentration must be increased to restore the regulating rod insertion to a value that preserves the SDM and ejected rod worth limits. The RCS boration must occur as described in Section B 3.1.1. The required Completion Time of 15 minutes to initiate boration is reasonable, based on limiting the potential xenon redistribution, the low probability of an accident occurring in this relatively short time period, and the number of steps required to complete this Action. This period allows the operator sufficient time for aligning the required valves and for starting the boric acid pumps. Boration continues until the regulating rod group positions are restored to at least within the restricted operational region, which restores the minimum SDM capability and reduces the potential ejected rod worth to within its limit.

C.2.1

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

(continued)

BASES

ACTIONS (continued)

C.2.2

The SDM and ejected rod worth limit can also be restored by reducing the THERMAL POWER to a value allowed by the regulating rod insertion limits in the COLR. The required Completion Time of 2 hours is sufficient to allow the operator to complete the power reduction in an orderly manner and without challenging the plant systems. Operation for up to 2 hours more in the restricted region shown in the COLR is acceptable, based on the low probability of an event occurring simultaneously with the limit out of specification in this relatively short time period. In addition, it precludes long term depletion with abnormal group insertions or configurations and limits the potential for an adverse xenon redistribution.

D.1

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

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1

This Surveillance ensures that the sequence and overlap limits are not violated. A Surveillance Frequency of 12 hours or 4 hours, depending on whether the CONTROL ROD drive sequence alarm is OPERABLE or not, is acceptable because little rod motion occurs in 4 hours due to fuel burnup, and the probability of a deviation occurring simultaneously with an inoperable sequence monitor in this relatively short time frame is low. Also, the Frequency

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1 (continued)

takes into account other information available in the control room for monitoring the status of the regulating rods.

SR 3.2.1.2

With an OPERABLE regulating rod insertion limit alarm, verification of the regulating rod insertion limits as specified in the COLR at a Frequency of 12 hours is sufficient to ensure the OPERABILITY of the regulating rod insertion limit alarm and to detect regulating rod banks that may be approaching the group insertion limits, because little rod motion due to fuel burnup occurs in 12 hours. If the insertion limit alarm becomes inoperable, verification of the regulating rod group position at a Frequency of 4 hours is sufficient to detect whether the regulating rod groups may be approaching or exceeding their group insertion limits, although more frequent surveillance is prudent if the regulating rod insertion limit alarm is not OPERABLE. Also, the Frequency takes into account other information available in the control room for monitoring the status of the regulating rods.

SR 3.2.1.3

Prior to achieving criticality, an estimated critical position for the CONTROL RODS is determined. Verification that SDM meets the minimum requirements ensures that sufficient SDM capability exists with the CONTROL RODS at the estimated critical position if it is necessary to shut down or trip the reactor after criticality. The Frequency of 4 hours prior to criticality provides sufficient time to verify SDM capability and establish the estimated critical position.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10 and GDC 26.
2. 10 CFR 50.46.

(continued)

BASES

REFERENCES (continued)

3. FSAR, Section [ ].
4. FSAR, Section [ ].
5. FSAR, Section [ ].
6. FSAR, Section [ ].
7. FSAR, Section [ ].
8. FSAR, Section [ ].

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.2 AXIAL POWER SHAPING ROD (APSR) Insertion Limits

BASES

BACKGROUND

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

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

The power density at any point in the core must be limited to maintain specified acceptable fuel design limits, including limits that meet the criteria specified in Reference 2. Together, LCO 3.2.1, "Regulating Rod Insertion Limits," LCO 3.2.2, "AXIAL POWER SHAPING ROD (APSR) Insertion Limits," LCO 3.2.3, "AXIAL POWER IMBALANCE Operating Limits," and LCO 3.2.4, "QUADRANT POWER TILT (QPT)," provide limits on control component operation and on monitored process variables to ensure that the core operates within the $F_Q(Z)$ and $F^N_{\Delta H}$ limits in the COLR. Operation within the $F_Q(Z)$ limits given in the COLR prevents power peaks that exceed the loss of coolant accident (LOCA) limits derived from the analysis of the Emergency Core Cooling Systems (ECCS). Operation within the $F^N_{\Delta H}$ limits given in the COLR prevents departure from nucleate boiling (DNB) during a loss of forced reactor coolant flow accident. The APSRs are not required for reactivity insertion rate on trip or SDM and, therefore, they do not trip upon a reactor trip.

This LCO is required to minimize fuel cladding failures that would breach the primary fission product barrier and release fission products to the reactor coolant in the event of a LOCA, loss of flow accident, ejected rod accident, or other postulated accident requiring termination by a Reactor Protection System trip function.

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BASES (continued)

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation (Condition 1) or anticipated operational occurrences (Condition 2). Acceptance criteria for the safety and regulating rod insertion, APSR position, AXIAL POWER IMBALANCE, and QPT LCOs preclude core power distributions that violate the following fuel design criteria:

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

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

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

d. CONTROL RODS must be capable of shutting down the reactor with a minimum required SDM with the highest worth CONTROL ROD stuck fully withdrawn (GDC 26, Ref. 1).

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

Operation at the APSR insertion limits may approach the maximum allowable linear heat generation rate or peaking factor with the allowed QPT present.

The APSR insertion limits satisfy Criterion 2 of the NRC Policy Statement.

LCO

The limits on APSR physical insertion as defined in the COLR must be maintained because they serve the function of

(continued)

BASES

LCO (continued)

controlling the power distribution within an acceptable range.

The fuel cycle design assumes APSR withdrawal at the effective full power day (EFPD) burnup window specified in the COLR. Prior to this window, the APSRs cannot be maintained fully withdrawn in steady state operation. After this window, the APSRs are not allowed to be reinserted for the remainder of the fuel cycle.

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

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

APPLICABILITY

The APSR physical insertion limits shall be maintained with the reactor in MODES 1 and 2. These limits maintain the power distribution within the range assumed in the accident analyses. In MODE 1, the limits on APSR insertion specified by this LCO maintain the axial fuel burnup design conditions assumed in the reload safety evaluation analysis. In MODE 2, applicability is required because $k_{eff} \geq 0.99$. Applicability in MODES 3, 4, and 5 is not required, because the power distribution assumptions in the accident analyses would not be exceeded in these MODES.

ACTIONS

For steady state power operation, a normal position for APSR insertion is specified in the station operating procedures. The APSRs may be positioned as necessary for transient AXIAL POWER IMBALANCE control until the fuel cycle design requires them to be fully withdrawn. (Not all fuel cycles may incorporate APSR withdrawal.) APSR position limits are not imposed for gray APSRs, with two exceptions. If the fuel cycle design incorporates an APSR withdrawal (usually near end of cycle (EOC)), the APSRs may not be maintained in the fully withdrawn position prior to the fuel cycle burnup for

(continued)

BASES

ACTIONS (continued)

the APSR withdrawal. If this occurs, the APSRs must be restored to their normal inserted position. Conversely, after the fuel cycle burnup for the APSR withdrawal occurs, the APSRs may not be reinserted for the remainder of the fuel cycle. These restrictions apply to ensure the axial burnup distribution that accumulates in the fuel will be consistent with the expected (as designed) distribution.

A.1

For verification that the core parameters $F_Q(Z)$ and $F^N_{\Delta H}$ are within their limits, SR 3.2.5.1 is performed using the Incore Detector System to obtain a three dimensional power distribution map. Successful verification that $F_Q(Z)$ and $F^N_{\Delta H}$ are within their limits ensures that operation with the APSRs inserted or withdrawn in violation of the times specified in the COLR do not violate either the ECCS or DNB criteria (Ref. 4). The required Completion Time of 2 hours is reasonable to allow the operator to obtain a power distribution map and to verify the power peaking factors. Repeating SR 3.2.5.1 every 2 hours is reasonable to ensure that continued verification of the power peaking factors is obtained as core conditions (primarily the regulating rod insertion and induced xenon redistribution) change.

A.2

Indefinite operation with the APSRs inserted or withdrawn in violation of the times specified in the COLR is not prudent. Even if power peaking monitoring per Required Action A.1 is continued, the abnormal APSR insertion or withdrawal may cause an adverse xenon redistribution, may cause the limits on AXIAL POWER IMBALANCE to be exceeded, or may affect the long term fuel depletion pattern. Therefore, power peaking monitoring is allowed for up to 24 hours. This required Completion Time is reasonable based on the low probability of an event occurring simultaneously with the APSR limit out of specification. In addition, it precludes long term depletion with the APSRs in positions that have not been analyzed, thereby limiting the potential for an adverse xenon redistribution. This time limit also ensures that the intended burnup distribution is maintained, and allows the operator sufficient time to reposition the APSRs to correct their positions.

(continued)

BASES

ACTIONS

A.2 (continued)

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

B.1

If the APSRs cannot be restored to their intended positions within the required Completion Time of 24 hours, the reactor must be placed in MODE 3, in which this LCO does not apply. This action ensures that the fuel does not continue to be depleted in an unintended burnup distribution. The required Completion Time of 6 hours is reasonable, based on operating experience regarding the time required to reach MODE 3 from RTP in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1

Fuel cycle designs that allow APSR withdrawal near EOC do not permit reinsertion of APSRs after the time of withdrawal. When the plant computer is OPERABLE, the operator will receive a computer alarm if the APSRs insert after that time in core life when the APSR withdrawal occurs. Verification that the APSRs are within their insertion limits at a 12 hour Frequency is sufficient to ensure that the APSR insertion limits are preserved and the computer alarm remains OPERABLE. The 12 hour Frequency required for performing this verification is sufficient because APSRs are positioned by manual control and are normally moved infrequently. The probability of a deviation occurring simultaneously with an inoperable computer alarm is low in this relatively short time frame. Also, the Frequency takes into account other information available in the control room for monitoring the axial power distribution in the reactor core.

(continued)

BASES (continued)

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10 and GDC 26.
2. 10 CFR 50.46.
3. FSAR, Chapter [ ].
4. FSAR, Chapter [ ].

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3 AXIAL POWER IMBALANCE Operating Limits

BASES

BACKGROUND

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

The power density at any point in the core must be limited to maintain specified acceptable fuel design limits, including limits that satisfy the criteria specified in 10 CFR 50.46 (Ref. 1). This LCO provides limits on AXIAL POWER IMBALANCE to ensure that the core operates within the $F_Q(Z)$ and $F^N_{\Delta H}$ limits given in the COLR. Operation within the $F_Q(Z)$ limits given in the COLR prevents power peaks that exceed the loss of coolant accident (LOCA) limits derived from the analysis of the Emergency Core Cooling Systems (ECCS). Operation within the $F^N_{\Delta H}$ limits given in the COLR prevents departure from nucleate boiling (DNB) during a loss of forced reactor coolant flow accident.

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

Fuel cladding failure during a postulated LOCA is limited by restricting the maximum linear heat rate (LHR) so that the peak cladding temperature does not exceed 2200°F (Ref. 2). Peak cladding temperatures > 2200°F cause severe cladding failure by oxidation due to a Zircaloy water reaction. Other criteria must also be met (e.g., maximum cladding oxidation, maximum hydrogen generation, coolable geometry, and long term cooling). However, peak cladding temperature is usually most limiting.

Proximity to the DNB condition is expressed by the departure from nucleate boiling ratio (DNBR), defined as the ratio of the cladding surface heat flux required to cause DNB to the actual cladding surface heat flux. The minimum DNBR value

(continued)

BASES

BACKGROUND (continued)

during both normal operation and anticipated transients is limited to the DNBR correlation limit for the particular fuel design in use and is accepted as an appropriate margin to DNB. The DNB correlation limit ensures that there is at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB.

The measurement system independent limits on AXIAL POWER IMBALANCE are determined directly by the reload safety evaluation analysis without adjustment for measurement system error and uncertainty. Operation beyond these limits could invalidate the assumptions used in the accident analyses regarding the core power distribution.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation (Condition 1) and anticipated operational occurrences (Condition 2). The LCOs based on power distribution, LCO 3.2.1, "Regulating Rod Insertion Limits," LCO 3.2.2, "AXIAL POWER SHAPING ROD (APSR) Insertion Limits," LCO 3.2.3, "AXIAL POWER IMBALANCE Operating Limits," and LCO 3.2.4, "QUADRANT POWER TILT (QPT)," preclude core power distributions that would violate the following fuel design criteria:

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

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

The regulating rod positions, the APSR positions, the AXIAL POWER IMBALANCE, and the QPT are process variables that characterize and control the three dimensional power distribution of the reactor core.

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

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

potential for fuel cladding damage exists because changes in the power distribution can cause increased power peaking and corresponding increased local LHRs.

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

The dependence of the core power distribution on burnup, regulating rod insertion, APSR position, and spatial xenon distribution is taken into account when the reload safety evaluation analysis is performed.

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

AXIAL POWER IMBALANCE satisfies Criterion 2 of the NRC Policy Statement.

LCO

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

Operation beyond the power distribution based LCO limits for the corresponding ALLOWABLE THERMAL POWER and simultaneous occurrence of either the LOCA or loss of forced reactor coolant flow accident has an acceptably low probability.

(continued)

BASES

LCO (continued)

Therefore, if the LCO limits are violated, a short time is allowed for corrective action before a significant power reduction is required.

The AXIAL POWER IMBALANCE maximum allowable setpoints (measurement system dependent limits) applicable for the full Incore Detector System, the Minimum Incore Detector System, and the Excore Detector System are provided in the COLR.

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

APPLICABILITY

In MODE 1, the limits on AXIAL POWER IMBALANCE must be maintained when THERMAL POWER is > 40% RTP to prevent the core power distribution from exceeding the LOCA and loss of flow assumptions used in the accident analyses. Applicability of these limits at < 40% RTP in MODE 1 is not required. This operation is acceptable because the combination of AXIAL POWER IMBALANCE with the maximum allowable THERMAL POWER level will not result in LHRs sufficiently large to violate the fuel design limits. In MODES 2, 3, 4, 5, and 6, this LCO is not applicable because the reactor is not generating sufficient THERMAL POWER to produce fuel damage.

In MODE 1, it may be necessary to suspend the AXIAL POWER IMBALANCE limits during PHYSICS TESTS per LCO 3.1.8, "PHYSICS TESTS Exceptions—MODE 1." Suspension of these limits is permissible because the reactor protection criteria are maintained by the remaining LCOs governing the three dimensional power distribution and by the Surveillances required by LCO 3.1.8.

ACTIONS

A.1

The AXIAL POWER IMBALANCE operating limits that maintain the validity of the assumptions regarding the power distributions in the accident analyses of the LOCA and the loss of flow accident are provided in the COLR. Operation

(continued)

BASES

ACTIONS

A.1 (continued)

within the AXIAL POWER IMBALANCE limits given in the COLR is the acceptable region of operation. Operation in violation of the AXIAL POWER IMBALANCE limits given in the COLR is the restricted region of operation.

Operation with AXIAL POWER IMBALANCE in the restricted region shown on the AXIAL POWER IMBALANCE figures in the COLR potentially violates the LOCA LHR limits ($F_Q(Z)$ limits) or the loss of flow accident DNB peaking limits ($F^N_{\Delta H}$ limits) or both. For verification that $F_Q(Z)$ and $F^N_{\Delta H}$ are within their specified limits, SR 3.2.5.1 is performed using the Incore Detector System to obtain a three dimensional power distribution map. Verification that $F_Q(Z)$ and $F^N_{\Delta H}$ are within their specified limits ensures that operation with the AXIAL POWER IMBALANCE in the restricted region does not violate the ECCS or 95/95 DNB criteria. The required Completion Time of 2 hours provides reasonable time for the operator to obtain a power distribution map and to determine and verify that the power peaking factors are within their specified limits. The 2 hour Frequency provides reasonable time to ensure that continued verification of the power peaking factors is obtained as core conditions (primarily regulating rod insertion and induced xenon redistribution) change, because little rod motion occurs in 2 hours due to fuel burnup, the potential for xenon redistribution is limited, and the probability of an event occurring in this short time frame is low.

A.2

Indefinite operation with the AXIAL POWER IMBALANCE in the restricted region is not prudent. Even if power peaking monitoring per Required Action A.1 is continued, excessive AXIAL POWER IMBALANCE over an extended period of time may cause a potentially adverse xenon redistribution to occur. Therefore, power peaking monitoring is only allowed for a maximum of 24 hours. This required Completion Time is reasonable based on the low probability of a limiting event occurring simultaneously with the AXIAL POWER IMBALANCE outside the limits of this LCO. In addition, this limited Completion Time precludes long term depletion of the reactor fuel with excessive AXIAL POWER IMBALANCE and gives the operator sufficient time to reposition the APSRs or

(continued)

BASES

ACTIONS

A.2 (continued)

regulating rods to reduce the AXIAL POWER IMBALANCE because adverse effects of xenon redistribution and fuel depletion are limited.

B.1

If the Required Actions and the associated Completion Times of Condition A cannot be met, the AXIAL POWER IMBALANCE may exceed its specified limits and the reactor may be operating with a global axial power distribution mismatch. Continued operation in this configuration may induce an axial xenon oscillation and may result in an increased linear heat generation rate when the xenon redistributes. Reducing THERMAL POWER to $\leq$ 40% RTP reduces the maximum LHR to a value that does not exceed the $F_Q(Z)$ and $F^N_{\Delta H}$ initial condition limits assumed in the accident analyses. The required Completion Time of 2 hours is reasonable based on limiting a potentially adverse xenon redistribution, the low probability of an accident occurring in this relatively short time period, and the number of steps required to complete this Action.

SURVEILLANCE REQUIREMENTS

The AXIAL POWER IMBALANCE can be monitored by both the Incore and Excore Detector Systems. The AXIAL POWER IMBALANCE maximum allowable setpoints are derived from their corresponding measurement system independent limits by adjusting for both the system observability errors and instrumentation errors. Although they may be based on the same measurement system independent limits, the setpoints for the different systems are not identical because of differences in the errors applicable for each of these systems. The uncertainty analysis that defines the required error adjustment to convert the measurement system independent limits to alarm setpoints assumes that 75% of the detectors in each quadrant are OPERABLE. Detectors located on the core major axes are assumed to contribute one half of their output to each quadrant; detectors in the center assembly are assumed to contribute one quarter of their output to each quadrant. For AXIAL POWER IMBALANCE measurements using the Incore Detector System, the Minimum

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

Incore Detector System consists of OPERABLE detectors configured as follows:

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

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

c. The detector strings shall not have radial symmetry.

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

SR 3.2.3.1

If the plant computer becomes inoperable, then the Excore System or Minimum Incore Detector System may be used to monitor the AXIAL POWER IMBALANCE. Although these systems do not provide a direct calculation and display of the AXIAL POWER IMBALANCE, a 1 hour Frequency provides reasonable time between calculations for detecting any trends in the AXIAL POWER IMBALANCE that may exceed its alarm setpoint and for undertaking corrective action.

When the Full Incore Detector System is OPERABLE, the operator receives an alarm if the AXIAL POWER IMBALANCE increases to its alarm setpoint. When the AXIAL POWER IMBALANCE is less then the alarm setpoint, verification of the AXIAL POWER IMBALANCE indication every 12 hours ensures

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.3.1 (continued)

that the AXIAL POWER IMBALANCE limits are not violated and verifies that the alarm system is OPERABLE. This Surveillance Frequency is acceptable because the mechanisms that can cause AXIAL POWER IMBALANCE, such as xenon redistribution or CONTROL ROD drive mechanism malfunctions that cause slow AXIAL POWER IMBALANCE increases, can be discovered by the operator before the specified limits are violated.

REFERENCES

1. 10 CFR 50.46.

2. FSAR, Chapter [15].

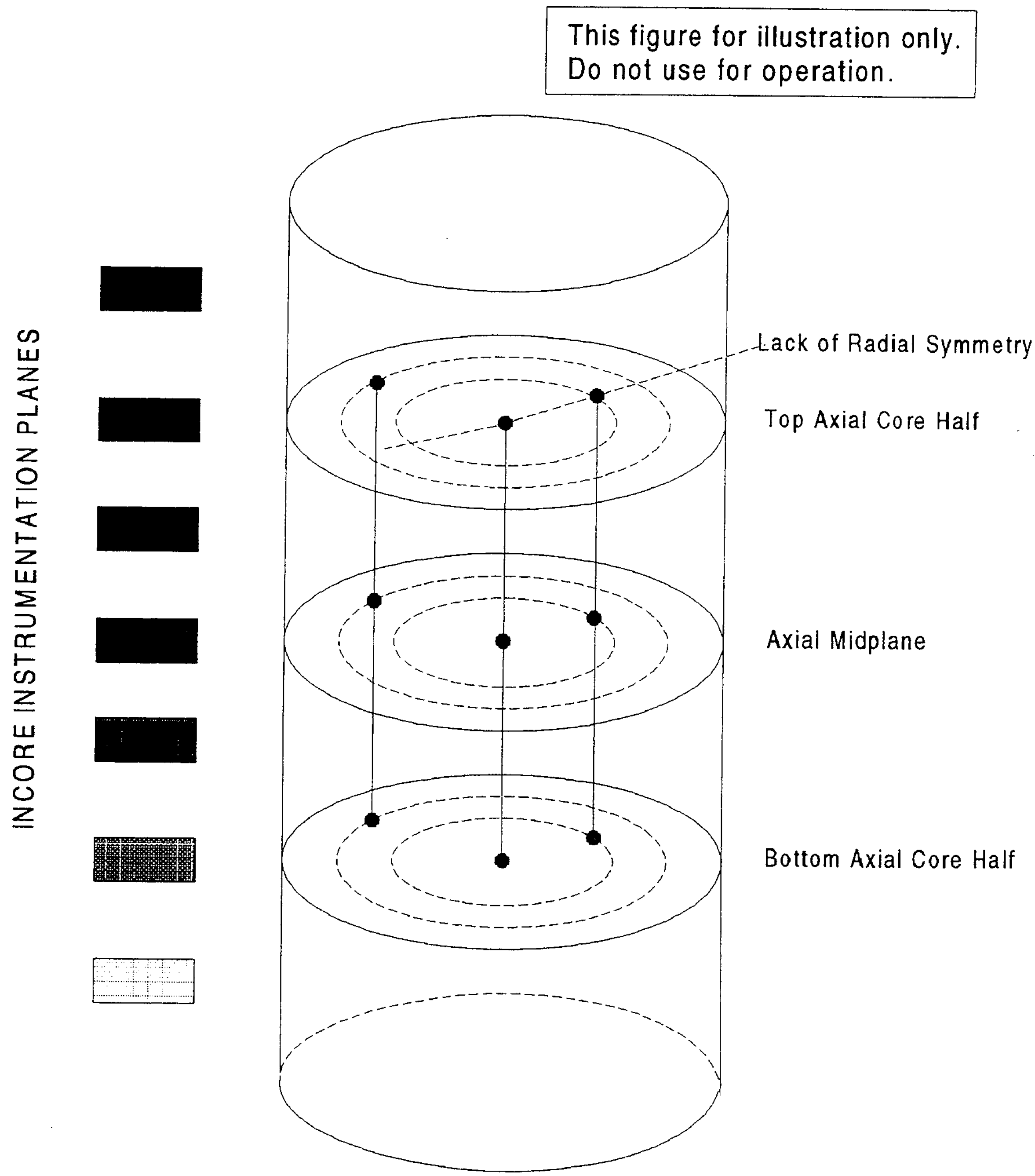

Figure B 3.2.3-1 (page 1 of 1)
Minimum Incore System for AXIAL POWER IMBALANCE Measurement

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 QUADRANT POWER TILT (QPT)

BASES

BACKGROUND

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

The power density at any point in the core must be limited to maintain specified acceptable fuel design limits, including limits that preserve the criteria specified in 10 CFR 50.46 (Ref. 1). Together, LCO 3.2.1, "Regulating Rod Insertion Limits," LCO 3.2.2, "AXIAL POWER SHAPING ROD (APSR) Insertion Limits," LCO 3.2.3, "AXIAL POWER IMBALANCE Operating Limits," and LCO 3.2.4, "QUADRANT POWER TILT (QPT)," provide limits on control component operation and on monitored process variables to ensure that the core operates within the $F_Q(Z)$ and $F^N_{\Delta H}$ limits given in the COLR. Operation within the $F_Q(Z)$ limits given in the COLR prevents power peaks that exceed the loss of coolant accident (LOCA) limits derived by Emergency Core Cooling Systems (ECCS) analysis. Operation within the $F^N_{\Delta H}$ limits given in the COLR prevents departure from nucleate boiling (DNB) during a loss of forced reactor coolant flow accident.

This LCO is required to limit fuel cladding failures that breach the primary fission product barrier and release fission products to the reactor coolant in the event of a LOCA, loss of forced reactor coolant flow, or other accident requiring termination by a Reactor Protection System trip function. This LCO limits the amount of damage to the fuel cladding during an accident by maintaining the validity of the assumptions used in the safety analysis related to the initial power distribution and reactivity.

Fuel cladding failure during a postulated LOCA is limited by restricting the maximum linear heat rate (LHR) so that the peak cladding temperature does not exceed 2200°F (Ref. 2). Peak cladding temperatures > 2200°F cause severe cladding failure by oxidation due to a Zircaloy water reaction. Other criteria must also be met (e.g., maximum cladding oxidation, maximum hydrogen generation, coolable geometry, and long term cooling). However, peak cladding temperature is usually most limiting.

(continued)

BASES

BACKGROUND (continued)

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

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

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation (Condition 1) and anticipated operational occurrences (Condition 2). The LCOs based on power distribution (LCO 3.2.1, LCO 3.2.2, LCO 3.2.3, and LCO 3.2.4) preclude core power distributions that violate the following fuel design criteria:

a. During a large break LOCA, the peak cladding temperature must not exceed 2200°F (Ref. 3).

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

QPT is one of the process variables that characterize and control the three dimensional power distribution of the reactor core.

Fuel cladding damage does not occur when the core is operated outside this LCO during normal operation. However, fuel cladding damage could result if an accident occurs with

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

simultaneous violation of one or more of the LCOs governing the core power distribution. Changes in the power distribution can cause increased power peaking and correspondingly increased local LHRs.

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

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

QPT satisfies Criterion 2 of the NRC Policy Statement.

LCO

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

Operation beyond the power distribution based LCO limits for the corresponding allowable THERMAL POWER and simultaneous occurrence of one of a LOCA, loss of forced reactor coolant

(continued)

BASES

LCO (continued)

flow accident, or ejected rod accident has an acceptably low probability. Therefore, if these LCO limits are violated, a short time is allowed for corrective action before a significant power reduction is required.

The maximum allowable setpoints for steady state, transient, and maximum limits for QPT applicable for the full symmetrical Incore Detector System, Minimum Incore Detector System, and Excore Detector System are provided; the setpoints are given in the COLR. The setpoints for the three systems are derived by adjustment of the measurement system independent QPT limits given in the COLR to allow for system observability and instrumentation errors.

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

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

APPLICABILITY

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

In MODE 2, the combination of QPT with maximum ALLOWABLE THERMAL POWER level does not result in LHRs sufficiently large to violate the fuel design limits, and therefore, applicability in this MODE is not required. Although not

(continued)

BASES

APPLICABILITY (continued)

specifically addressed in the LCO, QPTs > 20% in MODE 1 with THERMAL POWER < 20% RTP are allowed for the same reason.

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

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

ACTIONS

A.1.1

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

Operation with QPT greater than the steady state limit specified in the COLR potentially violates the LOCA LHR limits ($F_Q(Z)$ limits), or loss of flow accident DNB peaking limits ($F^N_{\Delta H}$ limits), or both. For verification that $F_Q(Z)$ and $F^N_{\Delta H}$ are within their specified limits, SR 3.1.5.2 is performed using the Incore Detector System to obtain a three dimensional power distribution map. Verification that $F_Q(Z)$ and $F^N_{\Delta H}$ are within their limits ensures that operation with QPT greater than the steady state limit does not violate the ECCS or 95/95 DNB criteria. The required Completion Time of once per 2 hours is a reasonable amount of time to allow the operator to obtain a power distribution map and to verify the power peaking factors. Repeating SR 3.2.5.1 every 2 hours is a reasonable Frequency at which to ensure that continued verification of the power peaking factors is obtained as core conditions that influence QPT change.

(continued)

BASES

ACTIONS (continued)

A.1.2.1

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

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

The required Completion Time of 2 hours after the last performance of SR 3.5.2.1 allows reduction of THERMAL POWER in the event the operators cannot or choose not to continue to perform SR 3.5.2.1 as required by Required Action A.1.1.

A.1.2.2

Power operation is allowed to continue if THERMAL POWER is reduced in accordance with Required Action A.1.2.1. The same reduction (i.e., 2% RTP or more) is also applicable to the nuclear overpower trip setpoint and the nuclear overpower based on Reactor Coolant System (RCS) flow and AXIAL POWER IMBALANCE trip setpoint, for each 1% of QPT in excess of the steady state limit. This reduction maintains both core protection and an OPERABILITY margin at the reduced THERMAL POWER level similar to that at RTP. The required Completion Time of 10 hours is reasonable based on the need to limit the potentially adverse xenon redistribution, the low probability of an accident occurring while operating out of specification, and the number of steps required to complete the Required Action.

(continued)

BASES

ACTIONS (continued)

A.2

Although the actions directed by Required Action A.1.2.1 restore margins, if the source of the QPT is not established and corrected, it is prudent to establish increased margins. A required Completion Time of 24 hours to reduce QPT to less than the steady state limit is a reasonable time for investigation and corrective measures.

B.1

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

B.2

When a misaligned CONTROL ROD or APSR occurs, a local xenon redistribution may occur. The required Completion Time of 2 hours allows the operator sufficient time to relatch or realign a CONTROL ROD or APSR, but is short enough to limit xenon redistribution so that large increases in the local LHR do not occur due to xenon redistribution resulting from the QPT.

C.1

If the Required Action and associated Completion Time of Condition A or B are not met, a further power reduction is required. Power reduction to < 60% RTP provides conservative protection from increased peaking due to xenon redistribution. The required Completion Time of 2 hours is reasonable to allow the operator to reduce THERMAL POWER to < 60% of ALLOWABLE THERMAL POWER without challenging plant systems.

(continued)

BASES

ACTIONS (continued)

C.2

Reduction of the nuclear overpower trip setpoint to ≤ 65.5% of ALLOWABLE THERMAL POWER after THERMAL POWER has been reduced to < 60% of ALLOWABLE THERMAL POWER maintains both core protection and OPERABILITY margin at reduced power similar to that at full power. The required Completion Time of 10 hours allows the operator sufficient time to reset the trip setpoint and is reasonable based on operating experience.

D.1

Power reduction to 60% of the ALLOWABLE THERMAL POWER is a conservative method of limiting the maximum core LHR for QPTs up to 20%. Although the power reduction is based on the correlation used in Required Actions A.1.2.1 and B.1, the database for a power peaking increase as a function of QPT is less extensive for tilt mechanisms other than misaligned CONTROL RODS and APSRs. Because greater uncertainty in the potential power peaking increase exists with the less extensive database, a more conservative action is taken when the tilt is caused by a mechanism other than a misaligned CONTROL ROD or APSR. The required Completion Time of 2 hours allows the operator to reduce THERMAL POWER to < 60% of the ALLOWABLE THERMAL POWER without challenging plant systems.

D.2

Reduction of the nuclear overpower trip setpoint to ≤ 65.5% of the ALLOWABLE THERMAL POWER after THERMAL POWER has been reduced to < 60% of the ALLOWABLE THERMAL POWER maintains both core protection and an operating margin at reduced power similar to that at full power. The required Completion Time of 10 hours allows the operator sufficient time to reset the trip setpoint and is reasonable based on operating experience.

E.1

If the Required Actions for Condition C or D cannot be met within the required Completion Time, then the reactor will

(continued)

BASES

ACTIONS

E.1 (continued)

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

F.1

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

The maximum limit of 20% QPT is consistent with allowing power operation up to 60% of ALLOWABLE THERMAL POWER when QPT setpoints are exceeded. QPT in excess of the maximum limit can be an indication of a severe power distribution anomaly, and a power reduction to at most 20% RTP ensures local LHRs do not exceed allowable limits while the cause is being determined and corrected.

The required Completion Time of 2 hours is reasonable to allow the operator to reduce THERMAL POWER to ≤ 20% RTP without challenging plant systems.

SURVEILLANCE REQUIREMENTS

QPT can be monitored by both the incore and excore detector systems. The QPT setpoints are derived from their corresponding measurement system independent limits by adjustment for system observability errors and instrumentation errors. Although they may be based on the same measurement system independent limit, the setpoints for the different systems are not identical because of

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

differences in the errors applicable for these systems. For QPT measurements using the Incore Detector System, the Minimum Incore Detector System consists of OPERABLE detectors configured as follows:

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

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

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

SR 3.2.4.1

Should the plant computer become inoperable, then the Excore System or Minimum Incore Detector System may be used to monitor the QPT. Because these systems do not provide a direct calculation and display of the QPT, performing the calculations at a 12 hour Frequency is sufficient to follow any changes in the QPT that may approach the setpoint because with the exception of CONTROL ROD related effects detected by other systems, QPT changes are slow. This Frequency also provides operators sufficient time to undertake corrective actions if QPT approaches the setpoints.

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

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1 (continued)

experience has confirmed the acceptability of a Surveillance Frequency of 7 days.

Following restoration of the QPT to within the steady state limit, operation at $\geq$ 95% RTP may proceed provided the QPT is determined to remain within the steady state limit at the increased THERMAL POWER level. In case QPT exceeds the steady state limit for more than 24 hours or exceeds the transient limit (Condition A, B, or D), the potential for xenon redistribution is greater. Therefore, the QPT is monitored for 12 consecutive hourly intervals to determine whether the period of any oscillation due to xenon redistribution causes the QPT to exceed the steady state limit again.

REFERENCES

1. 10 CFR 50.46.
2. FSAR, Section [ ].
3. ANSI N18.2-1973, American National Standards Institute, August 6, 1973.
4. BAW 10122A, Rev. 1, May 1984.

This figure for illustration only.
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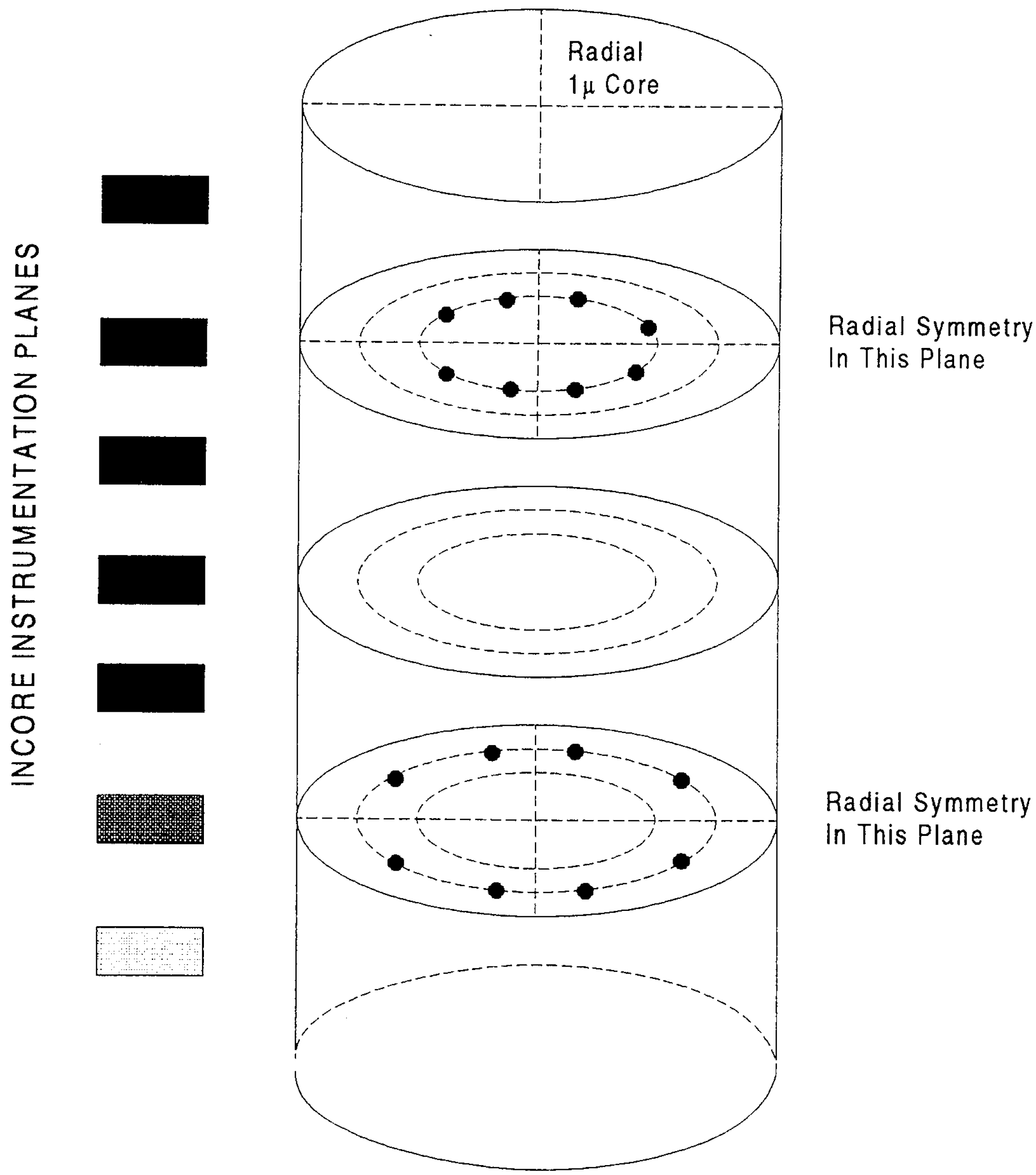

Figure B 3.2.4-1 (page 1 of 1)
Minimum Incore System for QUADRANT POWER TILT Measurement

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.5 Power Peaking Factors

BASES

BACKGROUND

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

$F_Q(Z)$ is a specified acceptable fuel design limit that preserves the initial conditions for the Emergency Core Cooling Systems (ECCS) analysis. $F_Q(Z)$ is defined as the maximum local fuel rod linear power density divided by the average fuel rod linear power density, assuming nominal fuel pellet and rod dimensions. Because $F_Q(Z)$ is a ratio of local power densities, it is related to the maximum local (pellet) power density in a fuel rod. Operation within the $F_Q(Z)$ limits given in the COLR prevents power peaking that would exceed the loss of coolant accident (LOCA) linear heat rate (LHR) limits derived from the analysis of the ECCS.

The $F^N_{\Delta H}$ limit is a specified acceptable fuel design limit that preserves the initial conditions for the limiting loss of flow transient. $F^N_{\Delta H}$ is defined as the ratio of the integral of linear power along the fuel rod on which the minimum departure from nucleate boiling ratio (DNBR) occurs to the average integrated rod power. Because $F^N_{\Delta H}$ is a ratio of integrated powers, it is related to the maximum total power produced in a fuel rod. Operation within the $F^N_{\Delta H}$ limits given in the COLR prevents departure from nucleate boiling (DNB) during a postulated loss of forced reactor coolant flow accident.

Measurement of the core power peaking factors using the Incore Detector System to obtain a three dimensional power distribution map provides direct confirmation that $F_Q(Z)$ and $F^N_{\Delta H}$ are within their limits, and may be used to verify that the power peaking factors remain bounded when one or more normal operating parameters exceed their limits.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The limits on $F_Q(Z)$ are determined by the ECCS analysis in order to limit peak cladding temperatures to 2200°F during a LOCA. The maximum acceptable cladding temperature is specified by 10 CFR 50.46 (Ref. 1). Higher cladding temperatures could cause severe cladding failure by oxidation due to a Zircaloy water reaction. Other criteria must also be met (e.g., maximum cladding oxidation, maximum hydrogen generation, coolable geometry, and long term cooling). However, peak cladding temperature is usually most limiting.

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

This LCO precludes core power distributions that violate the following fuel design criteria:

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

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

The reload safety evaluation analysis determines limits on global core parameters that characterize the core power distribution. The primary parameters used to monitor and control the core power distribution are the regulating rod position, the APSR position, the AXIAL POWER IMBALANCE, and the QPT. These parameters are normally used to monitor and control the core power distribution because their measurements are continuously observable. Limits are placed on these parameters to ensure that the core power peaking factors remain bounded during operation in MODE 1. Nuclear

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

design model calculational uncertainty, manufacturing tolerances (e.g., the engineering hot channel factor), effects of fuel densification and rod bow, and modeling simplifications (such as treatment of the spacer grid effects) are accommodated through use of peaking augmentation factors in the reload safety evaluation analysis.

$F_Q(Z)$ and $F^N_{\Delta H}$ satisfy Criterion 2 of the NRC Policy Statement.

LCO

This LCO for the power peaking factors $F_Q(Z)$ and $F^N_{\Delta H}$ ensures that the core operates within the bounds assumed for the ECCS and thermal hydraulic analyses. Verification that $F_Q(Z)$ and $F^N_{\Delta H}$ are within the limits of this LCO as specified in the COLR allows continued operation at THERMAL POWER when the Required Actions of LCO 3.1.4, "CONTROL ROD Group Alignment Limits," LCO 3.2.1, "Regulating Rod Insertion Limits," LCO 3.2.2, "AXIAL POWER SHAPING ROD Insertion Limits," LCO 3.2.3, "AXIAL POWER IMBALANCE Operating Limits," and LCO 3.2.4, "QUADRANT POWER TILT," are entered. Conservative THERMAL POWER reductions are required if the limits on $F_Q(Z)$ and $F^N_{\Delta H}$ are exceeded. Verification that $F_Q(Z)$ and $F^N_{\Delta H}$ are within limits is also required during MODE 1 PHYSICS TESTS per LCO 3.1.8, "PHYSICS TESTS Exceptions—MODE 1."

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

APPLICABILITY

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

(continued)

BASES (continued)

ACTIONS

The operator must take care in interpreting the relationship of the power peaking factors $F_Q(Z)$ and $F^N_{\Delta H}$ to their limits. Limit values of $F_Q(Z)$ and $F^N_{\Delta H}$ in the COLR may be expressed in either LHR or in peaking units. Because $F_Q(Z)$ and $F^N_{\Delta H}$ are power peaking factors, constant LHR is maintained as THERMAL POWER is reduced, thereby allowing power peaking to be increased in inverse proportion to THERMAL POWER.

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

A.1

When $F_Q(Z)$ is determined not to be within its specified limit as determined by a three dimensional power distribution map, a THERMAL POWER reduction is taken to reduce the maximum LHR in the core. Design calculations have verified that a conservative THERMAL POWER reduction is 1% RTP or more for each 1% by which $F_Q(Z)$ exceeds its limit (Ref. [ ]). The Completion Time of 15 minutes provides an acceptable time to reduce power in an orderly manner and without allowing the plant to remain in an unacceptable condition for an extended period of time.

A.2

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

(continued)

BASES

ACTIONS (continued)

A.3

Continued operation with $F_Q(Z)$ exceeding its limit is not permitted, because the initial conditions assumed in the accident analyses are no longer valid. The required Completion Time of 24 hours to restore $F_Q(Z)$ within its limits at the reduced THERMAL POWER level is reasonable based on the low probability of a limiting event occurring simultaneously with $F_Q(Z)$ exceeding its limit. In addition, it precludes long term depletion with local LHRs higher than the limiting values, and limits the potential for inducing an adverse perturbation in the axial xenon distribution.

B.1

When $F^N_{\Delta H}$ is determined not to be within its acceptable limit as determined by a three dimensional power distribution map, a THERMAL POWER reduction is taken to reduce the maximum LHR in the core. The parameter RH by which THERMAL POWER is decreased per 1% increase in $F^N_{\Delta H}$ above the limit has been verified to be conservative by design calculations, and is defined in the COLR. The parameter RH is the inverse of the increase in $F^N_{\Delta H}$ allowed as THERMAL POWER decreases by 1% RTP, and is based on an analysis of the DNBR during the limiting loss of forced reactor coolant flow transient from various initial THERMAL POWER levels. The required Completion Time of 15 minutes is reasonable for the operator to take the actions necessary to reduce the unit power.

B.2

When a decrease in THERMAL POWER is required because $F^N_{\Delta H}$ has exceeded its limit, Required Action B.2 requires reduction of the high flux trip setpoint and the nuclear overpower based on RCS flow and AXIAL POWER IMBALANCE trip setpoint. The amount of reduction of these trip setpoints is governed by the same factor (RH(%) for each 1% that $F^N_{\Delta H}$ exceeds its limit) that determines the THERMAL POWER reduction. This process maintains core protection by providing margin to the trip setpoints at the reduced THERMAL POWER similar to that at RTP. The parameter RH is specified in the COLR. The required Completion Time of 8 hours is reasonable based on the low probability of an accident occurring in this short

(continued)

BASES

ACTIONS

B.2 (continued)

time period and the number of steps required to complete this Action.

B.3

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

C.1

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

SURVEILLANCE REQUIREMENTS

SR 3.2.5.1

Core monitoring is performed using the Incore Detector System to obtain a three dimensional power distribution map. Maximum values of $F_Q(Z)$ and $F^N_{\Delta H}$ obtained from this map may then be compared with the $F_Q(Z)$ and limits in the COLR to verify that the limits have not been exceeded. Measurement of the core power peaking factors in this manner may be used to verify that the measured values of $F_Q(Z)$ and $F^N_{\Delta H}$ remain within their specified limits when one or more of the limits specified by LCO 3.1.4, LCO 3.2.1, LCO 3.2.2, LCO 3.2.3, or

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.5.1 (continued)

LCO 3.2.4 is exceeded, or when LCO 3.1.8 is applicable. If $F_Q(Z)$ and $F^N_{\Delta H}$ remain within their limits when one or more of these parameters exceed their limits, operation at THERMAL POWER may continue because the true initial conditions (the power peaking factors) remain within their specified limits.

Because the limits on $F_Q(Z)$ and $F^N_{\Delta H}$ are preserved when the parameters specified by LCO 3.1.4, LCO 3.2.1, LCO 3.2.2, LCO 3.2.3, and LCO 3.2.4 are within their limits, a Note is provided in the SR to indicate that monitoring of the power peaking factors is required only when complying with the Required Actions of these LCOs and when LCO 3.1.8 is applicable.

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

REFERENCES

1. 10 CFR 50.46.

B 3.3 INSTRUMENTATION

B 3.3.1 Reactor Protection System (RPS) Instrumentation

BASES

BACKGROUND

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

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

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

During AOOs, which are those events expected to occur one or more times during the unit's life, the acceptable limit is:

a. The departure from nucleate boiling ratio (DNBR) shall be maintained above the Safety Limit (SL) value;

b. Fuel centerline melt shall not occur; and

c. The RCS pressure SL of 2750 psia shall not be exceeded.

Maintaining the parameters within the above values ensures that the offsite dose will be within the 10 CFR 20 and 10 CFR 100 criteria during AOOs.

Accidents are events that are analyzed even though they are not expected to occur during the unit's life. The acceptable limit during accidents is that the offsite dose shall be maintained within 10 CFR 100 limits. Meeting the acceptable dose limit for an accident category is considered having acceptable consequences for that event.

(continued)

BASES

BACKGROUND (continued)

RPS Overview

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

Figure [ ], FSAR, Chapter [7] (Ref. 1), shows the arrangement of a typical RPS protection channel. A protection channel is composed of measurement channels, a manual trip channel, a reactor trip module (RTM), and CONTROL ROD drive (CRD) trip devices. LCO 3.3.1 provides requirements for the individual measurement channels. These channels encompass all equipment and electronics from the point at which the measured parameter is sensed through the bistable relay contacts in the trip string. LCO 3.3.2, "Reactor Protection System (RPS) Manual Reactor Trip," LCO 3.3.3, "Reactor Protection System (RPS)—Reactor Trip Module (RTM)," and LCO 3.3.4, "CONTROL ROD Drive (CRD) Trip Devices," discuss the remaining RPS elements.

The RPS instrumentation measures critical unit parameters and compares these to predetermined setpoints. If the setpoint is exceeded, a channel trip signal is generated. The generation of any two trip signals in any of the four RPS channels will result in the trip of the reactor.

The Reactor Trip System (RTS) contains multiple CRD trip devices, two AC trip breakers, and two DC trip breaker pairs that provide a path for power to the CRD System. Additionally, the power for most of the CRDs passes through electronic trip assembly (ETA) relays. The system has two separate paths (or channels), with each path having either two breakers or a breaker and an ETA relay in series. Each path provides independent power to the CRDs. Either path can provide sufficient power to operate all CRDs. Two separate power paths to the CRDs ensure that a single failure that opens one path will not cause an unwanted reactor trip.

The RPS consists of four independent protection channels, each containing an RTM. The RTM receives signals from its own measurement channels that indicate a protection channel trip is required. The RTM transmits this signal to its own two-out-of-four trip logic and to the two-out-of-four logic

(continued)

BASES

BACKGROUND

RPS Overview (continued)

of the RTMs in the other three RPS channels. Whenever any two RPS channels transmit channel trip signals, the RTM logic in each channel actuates to remove 120 VAC power from its associated CRD trip breaker.

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

The RPS has two bypasses: a shutdown bypass and a channel bypass. Shutdown bypass allows the withdrawal of safety rods for SDM availability and rapid negative reactivity insertion during unit cooldowns or heatups. Channel bypass is used for maintenance and testing. Test circuits in the trip strings allow complete testing of all RPS trip Functions.

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

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

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

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

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

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BASES

BACKGROUND (continued)

Power Range Nuclear Instrumentation

Power Range Nuclear Instrumentation channels provide inputs to the following trip Functions:

1. Nuclear Overpower

   a. Nuclear Overpower—High Setpoint;

   b. Nuclear Overpower—Low Setpoint;

7. Reactor Coolant Pump to Power;

8. Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE (Power Imbalance Flow);

9. Main Turbine Trip (Control Oil Pressure); and

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

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

Reactor Coolant System Outlet Temperature

The Reactor Coolant System Outlet Temperature provides input to the following Functions:

2. RCS High Outlet Temperature; and

5. RCS Variable Low Pressure.

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

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BASES

BACKGROUND (continued)

Reactor Coolant System Pressure

The Reactor Coolant System Pressure provides input to the following Functions:

3. RCS High Pressure;

4. RCS Low Pressure;

5. RCS Variable Low Pressure; and

11. Shutdown Bypass RCS High Pressure.

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

Reactor Building Pressure

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

Reactor Coolant Pump Power Monitoring

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

Reactor Coolant System Flow

The Reactor Coolant System Flow measurements are an input to the Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE trip, Function 8. The reactor coolant flow inputs to the RPS are provided by eight high accuracy differential pressure transmitters, four on each loop, which measure flow

(continued)

BASES

BACKGROUND

Reactor Coolant System Flow (continued)

through calibrated flow tubes. One flow input in each loop is associated with each protection channel.

Main Turbine Automatic Stop Oil Pressure

Main Turbine Automatic Stop Oil Pressure is an input to the Main Turbine Trip (Control Oil Pressure) reactor trip, Function 9. Each of the four protection channels receives turbine status information from the same four pressure switches monitoring main turbine automatic stop oil pressure. An open indication will be provided to the RPS on a turbine trip. Contact buffers in each protection channel continuously monitor the status of the contact inputs and initiate an RPS trip when a turbine trip is indicated.

Feedwater Pump Control Oil Pressure

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

RPS Bypasses

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

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

Channel Bypass

A channel bypass provision is provided to allow for maintenance and testing of the RPS. The use of channel bypass keeps the protection channel trip relay energized regardless of the status of the instrumentation channel of

(continued)

BASES

BACKGROUND

Channel Bypass (continued)

the bistable relay contacts. To place a protection channel in channel bypass, the other three channels must not be in channel bypass. This is ensured by contacts from the other channels being in series with the channel bypass relay. If any contact is open, the second channel cannot be bypassed. The second condition is the closing of the key switch. When the bypass relay is energized, the bypass contact closes, maintaining the channel trip relay in an energized condition. All RPS trips are reduced to a two-out-of-three logic in channel bypass.

Shutdown Bypass

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

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

In shutdown bypass, a normally closed contact opens and the operator closes the shutdown bypass key switch. This action bypasses the RCS Low Pressure trip, Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE trip, Reactor Coolant Pump to Power trip, and the RCS Variable Low Pressure trip, and inserts a new RCS High Pressure, 1850 psig trip. The operator can now withdraw the safety rods for additional SDM.

The insertion of the new high pressure trip performs two functions. First, with a trip setpoint of 1720 psig, the bistable prevents operation at normal system pressure, 2155 psig, with a portion of the RPS bypassed. The second

(continued)

BASES

BACKGROUND

Shutdown Bypass (continued)

function is to ensure that the bypass is removed prior to normal operation. When the RCS pressure is increased during a unit heatup, the safety rods are inserted prior to reaching 1720 psig. The shutdown bypass is removed, which returns the RPS to normal, and system pressure is increased to greater than 1800 psig. The safety rods are then withdrawn and remain at the full out condition for the rest of the heatup.

In addition to the Shutdown Bypass RCS High Pressure trip, the high flux trip setpoint is administratively reduced to 5% RTP while the RPS is in shutdown bypass. This provides a backup to the Shutdown Bypass RCS High Pressure trip and allows low temperature physics testing while preventing the generation of any significant amount of power.

Module Interlock and Test Trip Relay

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

Trip Setpoints/Allowable Value

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

The trip setpoints used in the bistables are based on the analytical limits stated in FSAR, Chapter [14] (Ref. 2). The selection of these trip setpoints is such that adequate protection is provided when all sensor and processing time delays are taken into account. To allow for calibration tolerances, instrumentation uncertainties, instrument drift, and severe environment errors for those RPS channels that must function in harsh environments as defined by 10 CFR 50.49 (Ref. 3), the Allowable Values specified in Table 3.3.1-1 in the accompanying LCO are conservatively

(continued)

BASES

BACKGROUND

Trip Setpoints/Allowable Value (continued)

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 "[Unit Specific Setpoint Methodology]" (Ref. 4). The actual nominal trip setpoint entered into the bistable is more conservative than that specified by the Allowable Value to account for changes in random measurement errors detectable by a CHANNEL FUNCTIONAL TEST. One example of such a change in measurement error is drift during the Surveillance Frequency. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value.

Setpoints in accordance with the Allowable Value ensure that the limits of Chapter 2.0, "Safety Limits," in the Technical Specifications 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 LCO 3.3.1 the Allowable Values listed in Table 3.3.1-1 are the LSSS.

Each channel can be tested online to verify that the signal and setpoint accuracy are within the specified allowance requirements of Reference 4. Once a designated channel is taken out of service for testing, a simulated signal is injected in place of the field instrument signal. The process equipment for the channel in test is then tested, verified, and calibrated. Surveillances for the channels are specified in the SR section.

The Allowable Values listed in Table 3.3.1-1 are based on the methodology described in "[Unit Specific Setpoint Methodology]" (Ref. 4), which incorporates all of the known uncertainties applicable for each channel. The magnitudes of those uncertainties are factored into the determination of each trip setpoint. All field sensors and signal processing equipment for these channels are assumed to operate within the allowances of these uncertainty magnitudes.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

Each of the analyzed accidents and transients can be detected by one or more RPS Functions. The accident analysis contained in (Ref. [ ]) takes credit for most RPS trip Functions. Functions not specifically credited in the accident analysis were qualitatively credited in the safety analysis and the NRC staff approved licensing basis for the unit. These Functions are high RB pressure, high temperature, turbine trip, and loss of main feedwater. These Functions may provide protection for conditions that do not require dynamic transient analysis to demonstrate Function performance. These Functions also serve as backups to Functions that were credited in the safety analysis.

The LCO requires all instrumentation performing an RPS Function to be OPERABLE. Failure of any instrument renders the affected channel(s) inoperable and reduces the reliability of the affected Functions. The four channels of each Function in Table 3.3.1-1 of the RPS instrumentation shall be OPERABLE at all times the reactor is critical to ensure that a reactor trip will be actuated if needed. Additionally, during shutdown bypass with any CRD trip breaker closed, the applicable RPS Functions must also be available. This ensures the capability to trip the withdrawn CONTROL RODS exists at all times that rod motion is possible. The trip Function channels specified in Table 3.3.1-1 are considered OPERABLE when all channel components necessary to provide a reactor trip are functional and in service for the required MODE or Other Specified Condition listed in Table 3.3.1-1.

Required Actions allow maintenance (protection channel) bypass of individual channels, but the bypass activates interlocks that prevent operation with a second channel bypass. Bypass effectively places the unit in a two-out-of-three logic configuration that can still initiate a reactor trip, even with a single failure within the system.

Only the Allowable Values are specified for each RPS trip Function in the LCO. Nominal trip setpoints are specified in the unit specific setpoint calculations. The nominal setpoints are selected to ensure that the setpoint measured by CHANNEL FUNCTIONAL TESTS does not exceed the Allowable Value if the bistable is performing as required. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

provided that operation and testing are consistent with the assumptions of the unit specific setpoint calculations. Each Allowable Value specified is more conservative than instrument uncertainties appropriate to the trip Function. These uncertainties are defined in the "[Unit Specific Setpoint Methodology]" (Ref. 4).

For most RPS Functions, the trip setpoint Allowable Value is to ensure that the departure from nucleate boiling (DNB) or RCS pressure SLs are not challenged. Cycle specific figures for use during operation are contained in the COLR.

Certain RPS trips function to indirectly protect the SLs by detecting specific conditions that do not immediately challenge SLs but will eventually lead to challenge if no action is taken. These trips function to minimize the unit transients caused by the specific conditions. The Allowable Value for these Functions is selected at the minimum deviation from normal values that will indicate the condition, without risking spurious trips due to normal fluctuations in the measured parameter.

The Allowable Values for bypass removal Functions are stated in the Applicable MODE or Other Specified Condition column of Table 3.3.1-1.

The safety analyses applicable to each RPS Function are discussed next.

1. <u>Nuclear Overpower</u>

   a. <u>Nuclear Overpower—High Setpoint</u>

      The Nuclear Overpower—High Setpoint trip provides protection for the design thermal overpower condition based on the measured out of core fast neutron leakage flux.

      The Nuclear Overpower—High Setpoint trip initiates a reactor trip when the neutron power reaches a predefined setpoint at the design overpower limit. Because THERMAL POWER lags the neutron power, tripping when the neutron power reaches the design overpower will limit THERMAL POWER to a maximum value of the design overpower.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

a. Nuclear Overpower—High Setpoint (continued)

Thus, the Nuclear Overpower—High Setpoint trip protects against violation of the DNBR and fuel centerline melt SLs. However, the RCS Variable Low Pressure, and Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE, provide more direct protection. The role of the Nuclear Overpower—High Setpoint trip is to limit reactor THERMAL POWER below the highest power at which the other two trips are known to provide protection.

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

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

The specified Allowable Value is selected to ensure that a trip occurs before reactor power exceeds the highest point at which the RCS Variable Low Pressure and the Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE trips are analyzed to provide protection against DNB and fuel centerline melt. The Allowable Value does not account for harsh environment induced errors, because the trip will actuate prior to degraded environmental conditions being reached.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

b. Nuclear Overpower—Low Setpoint

While in shutdown bypass, with the Shutdown Bypass RCS High Pressure trip OPERABLE, the Nuclear Overpower—Low Setpoint trip must be reduced to ≤ 5% RTP. The low power setpoint, in conjunction with the lower Shutdown Bypass RCS High Pressure setpoint, ensure that the unit is protected from excessive power conditions when other RPS trips are bypassed.

The setpoint Allowable Value was chosen to be as low as practical and still lie within the range of the out of core instrumentation.

2. RCS High Outlet Temperature

The RCS High Outlet Temperature trip, in conjunction with the RCS Low Pressure and RCS Variable Low Pressure trips, provides protection for the DNBR SL. A trip is initiated whenever the reactor vessel outlet temperature approaches the conditions necessary for DNB. Portions of each RCS High Outlet Temperature trip channel are common with the RCS Variable Low Pressure trip. The RCS High Outlet Temperature trip provides steady state protection for the DNBR SL.

The RCS High Outlet Temperature trip limits the maximum RCS temperature to below the highest value for which DNB protection by the Variable Low Pressure trip is ensured. The trip setpoint Allowable Value is selected to ensure that a trip occurs before hot leg temperatures reach the point beyond which the RCS Low Pressure and Variable Low Pressure trips are analyzed. Above the high temperature trip, the variable low pressure trip need not provide protection, because the unit would have tripped already. The setpoint Allowable Value does not reflect errors induced by harsh environmental conditions that the equipment is expected to experience because the trip is not required to mitigate accidents that create harsh conditions in the RB.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

3. RCS High Pressure

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

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

The setpoint Allowable Value is selected to ensure that the RCS High Pressure SL is not challenged during steady state operation or slow power increasing transients. The setpoint Allowable Value does not reflect errors induced by harsh environmental conditions because the equipment is not required to mitigate accidents that create harsh conditions in the RB.

4. RCS Low Pressure

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

The RCS Low Pressure setpoint Allowable Value is selected to ensure that a reactor trip occurs before RCS pressure is reduced below the lowest point at which the RCS Variable Low Pressure trip is analyzed. The RCS Low Pressure trip provides protection for primary system depressurization events and has been

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

4. RCS Low Pressure (continued)

credited in the accident analysis calculations for small break loss of coolant accidents (LOCAs). Consequently, harsh RB conditions created by small break LOCAs can affect performance of the RCS pressure sensors and transmitters. Therefore, degraded environmental conditions are considered in the Allowable Value determination.

5. RCS Variable Low Pressure

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

The RCS Variable Low Pressure setpoint Allowable Value is selected to ensure that a trip occurs when temperature and pressure approach the conditions necessary for DNB while operating in a temperature pressure region constrained by the low pressure and high temperature trips. The RCS Variable Low Pressure trip is not assumed for transient protection in the unit safety analysis; therefore, determination of the setpoint Allowable Value does not account for errors induced by a harsh RB environment.

6. Reactor Building High Pressure

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

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

6. Reactor Building High Pressure (continued)

The Allowable Value for RB High Pressure trip is set at the lowest value consistent with avoiding spurious trips during normal operation. The electronic components of the RB High Pressure trip are located in an area that is not exposed to high temperature steam environments during HELB transients. The components are exposed to high radiation conditions. Therefore, the determination of the setpoint Allowable Value accounts for errors induced by the high radiation.

7. Reactor Coolant Pump to Power

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

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

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

The Allowable Value for the Reactor Coolant Pump to Power trip setpoint is selected to prevent normal power operation unless at least three RCPs are operating. RCP status is monitored by power transducers on each pump. These relays indicate a loss of an RCP on overpower with an Allowable Value of ≥ 14,400 kW and on underpower with an Allowable Value of ≤ 1752 kW. The overpower Allowable Value is selected low enough to detect locked rotor conditions

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

7. Reactor Coolant Pump to Power (continued)

(although credit is not allowed for this capability) but high enough to avoid a spurious trip on the in rush current when the pumps start. The underpower Allowable Value is selected to reliably trip on loss of voltage to the RCPs. Neither the reactor power nor the pump power Allowable Value account for instrumentation errors caused by harsh environments because the trip Function is not required to respond to events that could create harsh environments around the equipment.

8. Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE

The Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE trip provides steady state protection for the power imbalance SLs. A reactor trip is initiated when the core power, AXIAL POWER IMBALANCE, and reactor coolant flow conditions indicate an approach to DNB or fuel centerline melt limits.

This trip supplements the protection provided by the Reactor Coolant Pump to Power trip, through the power to flow ratio, for loss of reactor coolant flow events. The power to flow ratio provides direct protection for the DNBR SL for the loss of a single RCP and for locked RCP rotor accidents. The imbalance portion of the trip is credited for steady state protection only.

The power to flow ratio of the Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE trip also provides steady state protection to prevent reactor power from exceeding the allowable power when the primary system flow rate is less than full four pump flow. Thus, the power to flow ratio prevents overpower conditions similar to the Nuclear Overpower trip. This protection ensures that during reduced flow conditions the core power is maintained below that required to begin DNB.

The Allowable Value is selected to ensure that a trip occurs when the core power, axial power peaking, and

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

8. Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE (continued)

reactor coolant flow conditions indicate an approach to DNB or fuel centerline melt limits. By measuring reactor coolant flow and by tripping only when conditions approach an SL, the unit can operate with the loss of one pump from a four pump initial condition. The Allowable Value for this Function is given in the unit COLR because the cycle specific core peaking changes affect the Allowable Value.

9. Main Turbine Trip (Control Oil Pressure)

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

Each of the four turbine oil pressure switches feeds all four protection channels through buffers that continuously monitor the status of the contacts. Therefore, failure of any pressure switch affects all protection channels.

For the Main Turbine Trip (Control Oil Pressure) bistable, the Allowable Value of 45 psig is selected to provide a trip whenever feedwater pump control oil pressure drops below the normal operating range. To ensure that the trip is enabled as required by the LCO, the reactor power bypass is set with an Allowable Value of 45% RTP. The turbine trip is not required to protect against events that can create a harsh environment in the turbine building. Therefore, errors induced by harsh environments are not included in the determination of the setpoint Allowable Value.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

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

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

For the feedwater pump control oil pressure bistable, the Allowable Value of 55 psig is selected to provide a trip whenever feedwater pump control oil pressure drops below the normal operating range. To ensure that the trip is enabled as required by the LCO, the reactor power bypass is set with an Allowable Value of 15% RTP. The Loss of Main Feedwater Pumps (Control Oil Pressure) trip is not required to protect against events that can create a harsh environment in the turbine building. Therefore, errors caused by harsh environments are not included in the determination of the setpoint Allowable Value.

11. Shutdown Bypass RCS High Pressure

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

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

11. Shutdown Bypass RCS High Pressure (continued)

Overpower—High Power trip setpoint, and again withdraw the safety rod groups before proceeding with startup.

Accidents analyzed in the FSAR, Chapter [14] (Ref. 2), do not describe events that occur during shutdown bypass operation, because the consequences of these events are enveloped by the events presented in the FSAR.

During shutdown bypass operation with the Shutdown Bypass RCS High Pressure trip active with a setpoint of ≤ [1720] psig and the Nuclear Overpower—Low Setpoint set at or below 5% RTP, the trips listed below can be bypassed. Under these conditions, the Shutdown Bypass RCS High Pressure trip and the Nuclear Overpower—Low Setpoint trip act to prevent unit conditions from reaching a point where actuation of these Functions is necessary.

1.a Nuclear Overpower—High Setpoint;

4. RCS Low Pressure;

5. RCS Variable Low Pressure;

7. Reactor Coolant Pump to Power; and

8. Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE.

The Shutdown Bypass RCS High Pressure Function's Allowable Value is selected to ensure a trip occurs before producing THERMAL POWER.

The RPS satisfies Criterion 3 of the NRC Policy Statement.

In MODES 1 and 2, the following trips shall be OPERABLE because the reactor is critical in these MODES. These trips are designed to take the reactor subcritical to maintain the SLs during AOOs and to assist the ESFAS in providing acceptable consequences during accidents.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

11. Shutdown Bypass RCS High Pressure (continued)

1.a Nuclear Overpower—High Setpoint;

2. RCS High Outlet Temperature;

3. RCS High Pressure;

4. RCS Low Pressure;

5. RCS Variable Low Pressure;

6. Reactor Building High Pressure;

7. Reactor Coolant Pump to Power; and

8. Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE.

Functions 1, 4, 5, 7, and 8 just listed may be bypassed in MODE 2 when RCS pressure is below [1720] psig, provided the Shutdown Bypass RCS High Pressure and the Nuclear Overpower—Low setpoint trip are placed in operation. Under these conditions, the Shutdown Bypass RCS High Pressure trip and the Nuclear Overpower—Low setpoint trip act to prevent unit conditions from reaching a point where actuation of these Functions is necessary.

Two other Functions are required to be OPERABLE during portions of MODE 1. These are the Main Turbine Trip (Control Oil Pressure) and the Loss of Main Feedwater Pumps (Control Oil Pressure) trip. These Functions are required to be OPERABLE above [45]% RTP and [15]% RTP, respectively. Analyses presented in BAW-1893 (Ref. 6) have shown that for operation below these power levels, these trips are not necessary to minimize challenges to the PORVs as required by NUREG-0737 (Ref. 5).

Because the only safety function of the RPS is to trip the CONTROL RODS, the RPS is not required to be OPERABLE in MODE 3, 4, or 5 if the reactor trip breakers are open, or the CRD System is incapable of rod withdrawal. Similarly, the RPS is not required to be OPERABLE in MODE 6 when the CONTROL RODS are decoupled from the CRDs.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

However, in MODE 2, 3, 4, or 5, the Shutdown Bypass RCS High Pressure and Nuclear Overpower—Low setpoint trips are required to be OPERABLE if the CRD trip breakers are closed and the CRD System is capable of rod withdrawal. Under these conditions, the Shutdown Bypass RCS High Pressure and Nuclear Overpower—Low setpoint trips are sufficient to prevent an approach to conditions that could challenge SLs.

ACTIONS

Conditions A, B, and C are applicable to all RPS protection Functions. If a channel's trip setpoint is found nonconservative with respect to the required Allowable Value in Table 3.3.1-1, or the transmitter, instrument loop, signal processing electronics or bistable is found inoperable, the channel must be declared inoperable and Condition A or Conditions A and B entered immediately.

When the number of inoperable channels in a trip Function exceed those specified in the 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.

[Reviewer's Note: If a unit is to take credit for topical reports as the basis for justifying Completion Times, the reports must be supported by an NRC Staff Safety Evaluation Report (SER) that establishes the acceptability of each topical report for that unit.]

A.1

If one or more Functions in one protection channel become inoperable, the affected protection channel must be placed in bypass or trip. If the channel is bypassed, all RPS Functions are placed in a two-out-of-three logic configuration and the bypass of any other channel is prevented. In this configuration, the RPS can still perform its safety function in the presence of a random failure of any single channel. Alternatively, the inoperable channel can be placed in trip. Tripping the affected protection channel places all RPS Functions in a one-out-of-three configuration.

(continued)

BASES

ACTIONS

A.1 (continued)

Operation in the two-out-of-three configuration or in the one-out-of-three configuration may continue indefinitely based on the NRC SER for BAW-10167, Supplement 2 (Ref. 7). In this configuration, the RPS is capable of performing its trip Function in the presence of any single random failure. The 1 hour Completion Time is sufficient to perform Required Action A.1.

B.1 and B.2

For Required Action B.1 and Required Action B.2, if one or more Functions in two protection channels become inoperable, one of two inoperable protection channels must be placed in trip and the other in bypass. These Required Actions place all RPS Functions in a one-out-of-two logic configuration and prevent bypass of a second channel. In this configuration, the RPS can still perform its safety functions in the presence of a random failure of any single channel. The 1 hour Completion Time is sufficient time to perform Required Action B.1 and Required Action B.2.

C.1

Required Action C.1 directs entry into the appropriate Condition referenced in Table 3.3.1-1. The applicable Condition referenced in the table is Function dependent. Each time an inoperable channel has not met any Required Action of Condition A or B, as applicable, and the associated Completion Time has expired, Condition C is entered for that channel and provides for transfer to the appropriate subsequent Condition.

D.1 and D.2

If the Required Action and associated Completion Time of Condition A or B are not met and Table 3.3.1-1 directs entry into Condition D, the unit must be brought to a MODE in which the specified RPS trip Functions are not required to be OPERABLE. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3

(continued)

BASES

ACTIONS

D.1 and D.2 (continued)

from full power conditions in an orderly manner and to open all CRD trip breakers without challenging plant systems.

E.1

If the Required Action and associated Completion Time of Condition A or B are not met and Table 3.3.1-1 directs entry into Condition E, the unit must be brought to a MODE in which the specified RPS trip Functions are not required to be OPERABLE. To achieve this status, all CRD trip breakers must be opened. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to open CRD trip breakers without challenging plant systems.

F.1

If the Required Action and associated Completion Time of Condition A or B are not met and Table 3.3.1-1 directs entry into Condition F, the unit must be brought to a MODE in which the specified RPS trip Function is not required to be OPERABLE. To achieve this status, THERMAL POWER must be reduced < [45]% RTP. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach [45]% RTP from full power conditions in an orderly manner without challenging plant systems.

G.1

If the Required Action and associated Completion Time of Condition A or B are not met and Table 3.3.1-1 directs entry into Condition G, the unit must be brought to a MODE in which the specified RPS trip Function is not required to be OPERABLE. To achieve this status, THERMAL POWER must be reduced < [15]% RTP. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach [15]% RTP from full power conditions in an orderly manner without challenging plant systems.

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BASES (continued)

SURVEILLANCE REQUIREMENTS

The SRs for each RPS Function are identified by the SRs column of Table 3.3.1-1 for that Function. Most Functions are subject to CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, CHANNEL CALIBRATION, and RPS RESPONSE TIME testing.

The SRs are modified by a Note. The [first] Note directs the reader to Table 3.3.1-1 to determine the correct SRs to perform for each RPS Function.

[Reviewer's Note: The CHANNEL FUNCTIONAL TEST Frequencies are based on approved topical reports. For a licensee to use these times, the licensee must justify the Frequencies as required by the NRC Staff SER for the topical report.]

SR 3.3.1.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; therefore, it is key in verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the unit staff based on a combination of the channel instrument uncertainties, including isolation, indication, and readability. If a channel is outside the criteria, it may be an indication that the transmitter or the signal processing equipment has drifted outside its limit. If the channels are within the criteria, it is an indication that the channels are OPERABLE. If the channels are normally off scale during times when surveillance is required, the CHANNEL CHECK will only verify that they are off scale in the same direction. Off scale low current loop channels are verified to be reading at the bottom of the range and not failed downscale.

The Frequency, about once every shift, is based on operating experience that demonstrates channel failure is rare. Since

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1 (continued)

the probability of two random failures in redundant channels in any 12 hour period is extremely low, the CHANNEL CHECK minimizes the chance of loss of protective function due to failure of redundant channels. The CHANNEL CHECK supplements less formal but more frequent checks of channel OPERABILITY during normal operational use of the displays associated with the LCO's required channels.

For Functions that trip on a combination of several measurements, such as the Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE Function, the CHANNEL CHECK must be performed on each input.

SR 3.3.1.2

This SR is the performance of a heat balance calibration for the power range channels every 24 hours when reactor power is > 15% RTP. The heat balance calibration consists of a comparison of the results of the calorimetric with the power range channel output. The outputs of the power range channels are normalized to the calorimetric. If the calorimetric exceeds the Nuclear Instrumentation System (NIS) channel output by ≥ [2]% RTP, the NIS is not declared inoperable but must be adjusted. If the NIS channel cannot be properly adjusted, the channel is declared inoperable. A Note clarifies that this Surveillance is required only if reactor power is ≥ 15% RTP and that 24 hours is allowed for performing the first Surveillance after reaching 15% RTP. At lower power levels, calorimetric data are inaccurate.

The power range channel's output shall be adjusted consistent with the calorimetric results if the calorimetric exceeds the power range channel's output by ≥ [2]% RTP. The value of [2]% is adequate because this value is assumed in the safety analyses of FSAR, Chapter [14] (Ref. 2). These checks and, if necessary, the adjustment of the power range channels ensure that channel accuracy is maintained within the analyzed error margins. The 24 hour Frequency is adequate, based on unit operating experience, which demonstrates the change in the difference between the power range indication and the calorimetric results rarely exceeds

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SURVEILLANCE REQUIREMENTS

SR 3.3.1.2 (continued)

a small fraction of [2]% in any 24 hour period. Furthermore, the control room operators monitor redundant indications and alarms to detect deviations in channel outputs.

SR 3.3.1.3

A comparison of power range nuclear instrumentation channels against incore detectors shall be performed at a 31 day Frequency when reactor power is > 15% RTP. A Note clarifies that 24 hours is allowed for performing the first Surveillance after reaching 15% RTP. If the absolute difference between the power range and incore measurements is ≥ [2]% RTP, the power range channel is not inoperable, but a CHANNEL CALIBRATION that adjusts the measured imbalance to agree with the incore measurements is necessary. If the power range channel cannot be properly recalibrated, the channel is declared inoperable. The calculation of the Allowable Value envelope assumes a difference in out of core to incore measurements of 2.5%. Additional inaccuracies beyond those that are measured are also included in the setpoint envelope calculation. The 31 day Frequency is adequate, considering that long term drift of the excore linear amplifiers is small and burnup of the detectors is slow. Also, the excore readings are a strong function of the power produced in the peripheral fuel bundles, and do not represent an integrated reading across the core. The slow changes in neutron flux during the fuel cycle can also be detected at this interval.

SR 3.3.1.4

A CHANNEL FUNCTIONAL TEST is performed on each required RPS channel to ensure that the entire channel will perform the intended function. Setpoints must be found within the Allowable Values specified in Table 3.3.1-1. Any setpoint adjustment shall be consistent with the assumptions of the current unit specific setpoint analysis.

The as found and as left values must also be recorded and reviewed for consistency with the assumptions of the

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.4 (continued)

surveillance interval extension analysis. The requirements for this review are outlined in BAW-10167 (Ref. 8).

The Frequency of [45] days on a STAGGERED TEST BASIS is consistent with the calculations of Reference 7 that indicate the RPS retains a high level of reliability for this test interval.

SR 3.3.1.5

This SR is the performance of a CHANNEL CALIBRATION every [92] days. This CHANNEL CALIBRATION normalizes the power range channel output to the calorimetric coincident with the imbalance output being normalized to the imbalance condition predicted by the incore neutron detector system.

The calibration for both imbalance and total power is integrated in the power imbalance detector calibration procedure. The [92] day Frequency specified for the Nuclear Overpower trip string is consistent with the drift assumptions made in the "[Unit Specific Setpoint Methodology]" (Ref. 4). Furthermore, operating experience shows the reliability of the trip string is acceptable when calibrated on this interval. A Note clarifies that the neutron detectors are not required to be tested as part of the CHANNEL CALIBRATION. There is no adjustment that can be made to the detectors. Furthermore, adjustment of the detectors is unnecessary because they are passive devices with minimal drift. Slow changes in detector sensitivity are compensated for by performing the daily calorimetric calibration and the monthly axial channel calibration.

SR 3.3.1.6

A Note to the Surveillance indicates that neutron detectors are excluded from CHANNEL CALIBRATION. This Note is necessary because of the difficulty in generating an appropriate detector input signal. Excluding the detectors is acceptable because the principles of detector operation ensure a virtually instantaneous response.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.6 (continued)

A CHANNEL CALIBRATION is a complete check of the instrument channel, including the sensor. The test verifies that the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift to ensure that the instrument channel remains operational between successive tests. CHANNEL CALIBRATION shall find that measurement errors and bistable setpoint errors are within the assumptions of the unit specific setpoint analysis. CHANNEL CALIBRATIONS must be performed consistent with the assumptions of the unit specific setpoint analysis.

The Frequency is justified by the assumption of an [18] month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

SR 3.3.1.7

This SR verifies individual channel actuation response times are less than or equal to the maximum values assumed in the accident analysis. Individual component response times are not modeled in the analyses. The analyses model the overall, or total, elapsed time from the point at which the parameter exceeds the analytical limit at the sensor to the point of rod insertion. Response time testing acceptance criteria for this unit are included in Reference 1.

A Note to the Surveillance indicates that neutron detectors are excluded from RPS RESPONSE TIME testing. This Note is necessary because of the difficulty in generating an appropriate detector input signal. Excluding the detectors is acceptable because the principles of detector operation ensure a virtually instantaneous response.

Response time tests are conducted on an [18] month STAGGERED TEST BASIS. Testing of the final actuation devices, which make up the bulk of the response time, is included in the testing of each channel. Therefore, staggered testing results in response time verification of these devices every [18] months. The [18] month Frequency is based on unit operating experience, which shows that random failures of

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.7 (continued)

instrumentation components causing serious response time degradation, but not channel failure, are infrequent occurrences.

REFERENCES

1. FSAR, Chapter [7].
2. FSAR, Chapter [14].
3. 10 CFR 50.49.
4. "[Unit Specific Setpoint Methodology]."
5. NUREG-0737, November 1979.
6. BAW-1893.
7. NRC SER for BAW-10167, Supplement 2, July 8, 1992.
8. BAW-10167, May 1986.

B 3.3 INSTRUMENTATION

B 3.3.2 Reactor Protection System (RPS) Manual Reactor Trip

BASES

BACKGROUND

The RPS Manual Reactor Trip provides the operator with the capability to trip the reactor from the control room in the absence of any other trip condition. Manual trip is provided by a trip push button on the main control board. This push button operates four electrically independent switches, one for each train. This trip is independent of the automatic trip system. As shown in Figure [ ], FSAR, Chapter [7] (Ref. 1), power for the CONTROL ROD drive (CRD) breaker undervoltage coils and contactor coils comes from the reactor trip modules (RTMs). The manual trip switches are located between the RTM output and the breaker undervoltage coils. Opening of the switches opens the lines to the breakers, tripping them. The switches also energize the breaker shunt trip mechanisms. There is a separate switch in series, with the output of each of the four RTMs. All switches are actuated through a mechanical linkage from a single push button.

APPLICABLE SAFETY ANALYSES

The Manual Reactor Trip ensures that the control room operator can initiate a reactor trip at any time. The Manual Reactor Trip Function is required as a backup to the automatic trip functions and allows operators to shut down the reactor whenever any parameter is rapidly trending toward its trip setpoint.

The Manual Reactor Trip Function satisfies Criterion 3 of the NRC Policy Statement.

LCO

The LCO on the RPS Manual Reactor Trip requires that the trip shall be OPERABLE whenever the reactor is critical or any time any control rod breaker is closed and rods are capable of being withdrawn, including shutdown bypass. This enables the operator to terminate any reactivity excursion that in the operator's judgment requires protective action, even if no automatic trip condition exists.

(continued)

BASES

LCO (continued)

The Manual Reactor Trip Function is composed of four electrically independent trip switches sharing a common mechanical push button.

APPLICABILITY

The Manual Reactor Trip Function is required to be OPERABLE in MODES 1 and 2. It is also required to be OPERABLE in MODES 3, 4, and 5 if any CRD trip breaker is in the closed position and if the CRD System is capable of rod withdrawal. The only safety function of the RPS is to trip the CONTROL RODS; therefore, the Manual Reactor Trip Function is not needed in MODE 3, 4, or 5 if the reactor trip breakers are open or if the CRD System is incapable of rod withdrawal. Similarly, the RPS Manual Reactor Trip is not needed in MODE 6 when the CONTROL RODS are decoupled from the CRDs.

ACTIONS

A.1

Condition A applies when the Manual Reactor Trip Function is found inoperable. One hour is allowed to restore Function to OPERABLE status. The automatic functions and various alternative manual trip methods, such as removing power to the RTMs, are still available. The 1 hour Completion Time is sufficient time to correct minor problems.

B.1 and B.2

With the Manual Reactor Trip Function inoperable and unable to be returned to OPERABLE status within 1 hour in MODE 1, 2, or 3, the unit must be placed in a MODE in which manual trip is not required. Required Action B.1 and Required Action B.2 place the unit in at least MODE 3 with all CRD trip breakers open within 6 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging unit systems.

C.1

With the Manual Reactor Trip Function inoperable and unable to be returned to OPERABLE status within 1 hour in MODE 4

(continued)

BASES

ACTIONS	C.1 (continued) or 5, the unit must be placed in a MODE in which manual trip is not required. To achieve this status, all CRD trip breakers must be opened. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to open all CRD trip breakers without challenging unit systems.
SURVEILLANCE REQUIREMENTS	SR 3.3.2.1 This SR requires the performance of a CHANNEL FUNCTIONAL TEST of the Manual Reactor Trip Function. This test verifies the OPERABILITY of the Manual Reactor Trip by actuation of the CRD trip breakers. The Frequency shall be once prior to each reactor startup if not performed within the preceding 7 days to ensure the OPERABILITY of the Manual Reactor Trip Function prior to achieving criticality. The Frequency was developed in consideration that these Surveillances are only performed during a unit outage.
REFERENCES	1. FSAR, Chapter [7].

B 3.3 INSTRUMENTATION

B 3.3.3 Reactor Protection System (RPS)—Reactor Trip Module (RTM)

BASES

BACKGROUND

The RPS consists of four independent protection channels, each containing an RTM. Figure [ ], FSAR, Chapter [7] (Ref. 1), shows a typical RPS protection channel and the relationship of the RTM to the RPS instrumentation, manual trip, and CONTROL ROD drive (CRD) trip devices. The RTM receives bistable trip signals from the functions in its own channel and channel trip signals from the other three RPS—RTMs. The RTM provides these signals to its own two-out-of-four trip logic and transmits its own channel trip signal to the two-out-of-four logic of the RTMs in the other three RPS channels. Whenever any two RPS channels transmit channel trip signals, the RTM logic in each channel actuates to remove 120 VAC power from its associated CRD trip device.

The RPS trip scheme consists of series contacts that are operated by bistables. During normal unit operations, all contacts are closed and the RTM channel trip relay remains energized. However, if any trip parameter exceeds its setpoint, its associated contact opens, which de-energizes the channel trip relay.

When an RTM channel trip relay de-energizes, several things occur:

a. Each of the four (4) output logic relays "informs" its associated RPS channel that a reactor trip signal has occurred in the tripped RPS channel;

b. The contacts in the trip device circuitry, powered by the tripped channel, open, but the trip device remains energized through the closed contacts from the other RTMs. (This condition exists in each RPS—RTM. Each RPS—RTM controls power to a trip device.); and

c. The contact in parallel with the channel reset switch opens and the trip is sealed in. To re-energize the channel trip relay, the channel reset switch must be depressed after the trip condition has cleared.

(continued)

BASES

BACKGROUND (continued)

When the second RPS channel senses a reactor trip condition, the output logic relays for the second channel de-energize and open contacts that supply power to the trip devices. With contacts opened by two separate RPS channels, power to the trip devices is interrupted and the CONTROL RODS fall into the core.

A minimum of two out of four RTMs must sense a trip condition to cause a reactor trip. Also, because the bistable relay contacts for each function are in series with the channel trip relays, two channel trips caused by different trip functions can result in a reactor trip.

APPLICABLE SAFETY ANALYSES

Accident analyses rely on a reactor trip for protection of reactor core integrity, reactor coolant pressure boundary integrity, and reactor building OPERABILITY. A reactor trip must occur when needed to prevent accident conditions from exceeding those calculated in the accident analyses. More detailed descriptions of the applicable accident analyses are found in the bases for each of the RPS trip Functions in LCO 3.3.1, "Reactor Protection System (RPS) Instrumentation."

RTM response time is included in the overall required response time for each RPS trip and is not specified separately.

The RTMs satisfy Criterion 3 of the NRC Policy Statement.

LCO

The RTM LCO requires all four RTMs to be OPERABLE. Failure of any RTM renders a portion of the RPS inoperable and reduces the reliability of the affected Functions.

Four RTMs must be OPERABLE to ensure that a reactor trip would occur if needed any time the reactor is critical. OPERABILITY is defined as the RTM being able to receive and interpret trip signals from its own and other RPS channels and to open its associated trip device.

The requirement of four channels to be OPERABLE ensures that a minimum of two RPS channels will remain OPERABLE if a single failure has occurred in one channel and if a second

(continued)

BASES

LCO (continued)

channel has been bypassed for surveillance or maintenance. This two-out-of-four trip logic also ensures that a single RPS channel failure will not cause an unwanted reactor trip. Violation of this LCO could result in a trip signal not causing a reactor trip when needed.

APPLICABILITY

The RTMs are required to be OPERABLE in MODES 1 and 2. They are also required to be OPERABLE in MODES 3, 4, and 5 if any CRD trip breakers are in the closed position and the CRD System is capable of rod withdrawal. The RTMs are designed to ensure a reactor trip would occur, if needed, any time the reactor is critical. This condition can exist in all of these MODES; therefore, the RTMs must be OPERABLE.

ACTIONS

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

When an RTM is inoperable, the associated CRD trip breaker must then be placed in a condition that is equivalent to a tripped condition for the RTM. Required Action A.1.1 or Required Action A.1.2 requires this either by tripping the CRD trip breaker or by removing power to the CRD trip device. Tripping one RTM or removing power opens one set of CRD trip devices. Power to hold up CONTROL RODS is still provided via the parallel CRD trip device(s). Therefore, a reactor trip will not occur until a second protection channel trips.

To ensure the trip signal is registered in the other channels, Required Action A.2 requires that the inoperable RTM be removed from the cabinet. This action causes the electrical interlocks to indicate a tripped channel in the remaining three RTMs. Operation in this condition is allowed indefinitely because the actions put the RPS into a one-out-of-three configuration. The 1 hour Completion Time is sufficient time to perform the Required Actions.

B.1, B.2.1, and B.2.2

Condition B applies if the Required Actions of Condition A are not met within the required Completion Time in MODE 1, 2, or 3. In this case, the unit must be placed in a MODE in

(continued)

BASES

ACTIONS

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

which the LCO does not apply. This is done by placing the unit in at least MODE 3 with all CRD trip breakers open or with all power to the CRD System removed within 6 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging unit systems.

C.1 and C.2

Condition C applies if the Required Actions of Condition A are not met within the required Completion Time in MODE 4 or 5. In this case, the unit must be placed in a MODE in which the LCO does not apply. This is done by opening all CRD trip breakers or removing all power to the CRD System. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to open all CRD trip breakers or remove all power to the CRD System without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.3.3.1

The Note defines a channel as being OPERABLE for up to 8 hours while bypassed for Surveillance testing. The Note allows channel bypass for testing without defining it as inoperable although during this time period it cannot actuate a reactor trip. This allowance is based on the assumption of the RPS reliability analysis in BAW-10167 (Ref. 2) that 8 hours is the average time required to perform channel Surveillance. The analysis demonstrated that the 8 hour testing allowance does not significantly reduce the probability that the RPS will trip when necessary. It is not acceptable to routinely remove channels from service for more than 8 hours to perform required Surveillance testing. Such a practice would be contrary to the assumptions of the reliability analysis that justified the LCO's Completion Times.

[Reviewer's Note: The CHANNEL FUNCTIONAL TEST Frequency is based on an approved topical report. For a licensee to use

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.3.1 (continued)

[this Frequency, the licensee must justify the Frequency as required by the NRC Staff SER for the topical report.]

The SRs include performance of a CHANNEL FUNCTIONAL TEST every [45] days on a STAGGERED TEST BASIS. This test shall verify the OPERABILITY of the RTM and its ability to receive and properly respond to channel trip and reactor trip signals. Calculations have shown that the Frequency (45 days) maintains a high level of reliability of the Reactor Trip System in BAW-10167 (Ref. 2).

REFERENCES

1. FSAR, Chapter [7].

2. BAW-10167, May 1986.

B 3.3 INSTRUMENTATION

B 3.3.4 CONTROL ROD Drive (CRD) Trip Devices

BASES

BACKGROUND

The Reactor Protection System (RPS) contains multiple CRD trip devices: two AC trip breakers, two DC trip breaker pairs, and eight electronic trip assembly (ETA) relays. The system has two separate paths (or channels), with each path having one AC breaker in series with either a pair of DC breakers or four ETA relays in parallel. Each path provides independent power to the CRDs. Either path can provide sufficient power to operate the entire CRD System.

Figure [ ], FSAR, Chapter [7] (Ref. 1), illustrates the configuration of CRD trip devices. To trip the reactor, power to the CRDs must be removed. Loss of power causes the CRD's mechanisms to release the CONTROL RODS, which then fall by gravity into the core.

Power to CRDs is supplied from two separate unit sources through the AC trip circuit breakers. These breakers are designated A and B, and their undervoltage and shunt trip coils are powered by RPS channels A and B, respectively. From the circuit breakers, the CRD power travels through voltage regulators and stepdown transformers. These devices in turn supply redundant buses that feed the DC power supplies and the regulating rod power supplies.

The DC power supplies rectify the AC input and supply power to hold the safety rods in their fully withdrawn position. One of the redundant power sources supplies phase A; the other, phase C. Either phase being energized is sufficient to hold the rod. Two breakers are located on the output of each power supply. Each breaker controls power to one of the four safety rod groups. The undervoltage and shunt trip coils on the two circuit breakers on the output of one of the power supplies is controlled by RPS channel C. The other two breakers are controlled by RPS channel D.

In addition to the DC power supplies, the redundant buses also supply power to the regulating and auxiliary power supplies. These power supplies consist of ETAs that are gated on by programming lamps. Programming lamp power is controlled by contactors (E and F), which are controlled by

(continued)

BASES

BACKGROUND (continued)

RPS power. One of the redundant programming lamp supplies is controlled by RPS channel C; the other, by RPS channel D.

The AC breaker and DC breakers are in series in one of the power supplies; whereas, the redundant AC breaker and DC breakers are in series in the other power supply to the CONTROL RODS. The logic required to cause a reactor trip is the opening of a circuit breaker in each of the redundant power supplies. (The pair of DC circuit breakers on the output of the power supply are treated as one breaker.) This is known as a one-out-of-two taken twice logic. The following examples illustrate the operation of the reactor trip circuit breakers.

a. If the A AC circuit breaker opens:

    1. the input power to associated DC power supply is lost, and

    2. the SCR supply from the associated power source is lost.

b. If the D DC circuit breaker(s) and F contactors open:

    1. the output of the redundant DC power supply is lost and the safety rods de-energize, and

    2. when the F contactor opens, programming lamp power is lost and the regulating rods will be de-energized.

c. The combination of (a) and (b) causes a reactor trip.

Any other combination of at least one circuit breaker opening in each power supply will cause a reactor trip.

In summary, two tripped RPS channels will cause a reactor trip. For example, a reactor trip occurs if RPS channel B senses a low Reactor Coolant System (RCS) pressure condition and if RPS channel C senses a variable low RCS pressure condition. When the channel B bistable relay de-energizes, the channel trip relay de-energizes and opens its associated contacts. The same thing occurs in channel C, except the variable lower pressure bistable relay de-energizes the channel C trip relay. When the output logic relays de-energize, the B and C contacts in the undervoltage and E

(continued)

BASES

BACKGROUND (continued)

and F contacts de-energize, all circuit breakers open, and programming lamp power is removed. All rods fall into the core, resulting in a reactor trip.

APPLICABLE SAFETY ANALYSES

Accident analyses rely on a reactor trip for protection of reactor core integrity, reactor coolant pressure boundary integrity, and reactor building OPERABILITY. A reactor trip must occur when needed to prevent accident consequences from exceeding those calculated in the accident analyses. The control rod insertion limits ensure that adequate rod worth is available upon reactor trip to shut down the reactor to the required SDM. Further, OPERABILITY of the CRD trip devices ensures that all CONTROL RODS (except Group 8) will trip when required. More detailed descriptions of the applicable accident analyses are found in the Bases for each of the individual RPS trip Functions in LCO 3.3.1, "Reactor Protection System (RPS) Instrumentation."

The CRD trip devices satisfy Criterion 3 of the NRC Policy Statement.

LCO

The LCO requires all of the CRD trip devices to be OPERABLE. Failure of any CRD trip device renders a portion of the RPS inoperable and reduces the reliability of the affected Functions. Without reliable CRD reactor trip circuit breakers and associated support circuitry, a reactor trip cannot occur when initiated either automatically or manually.

All CRD trip devices shall be OPERABLE to ensure that the reactor remains capable of being tripped any time it is critical. OPERABILITY is defined as the CRD trip device being able to receive a reactor trip signal and to respond to this trip signal by interrupting AC power to the CRDs. Both of the AC breaker's trip devices and the breaker itself must be functioning properly for the AC breaker to be OPERABLE.

Requiring all breakers and ETA relays to be OPERABLE ensures that at least one device in each of the two power paths to the CRDs will remain OPERABLE even with a single failure.

(continued)

BASES

LCO (continued)

Requiring all devices OPERABLE also ensures that a single failure will not cause an unwanted reactor trip.

APPLICABILITY

The CRD trip devices shall be OPERABLE in MODES 1 and 2, and in MODES 3, 4, and 5 when any CRD trip breaker is in the closed position and the CRD System is capable of rod withdrawal.

The CRD trip devices are designed to ensure that a reactor trip would occur if needed any time the reactor is critical. Since this condition can exist in all of these MODES, the CRD trip devices shall be OPERABLE.

ACTIONS

A Note has been added to the ACTIONS indicating separate Condition entry is allowed for each CRD trip device.

Condition A

Condition A represents reduced redundancy in the CRD trip Function. Condition A applies when:

- One diverse trip Function (undervoltage or shunt trip device) is inoperable in one or more CRD trip breaker(s) [or breaker pair]; or
- One diverse trip Function is inoperable in both DC trip breakers associated with one protection channel. In this case, the inoperable trip Function does not need to be the same for both breakers.

A.1 and A.2

If one of the diverse trip Functions on a CRD trip breaker [or breaker pair] becomes inoperable, actions must be taken to preclude the inoperable CRD trip device from preventing a reactor trip when needed. This is done by manually tripping the inoperable CRD trip breaker or by removing power from the channel containing the inoperable CRD trip breaker. Either of these actions places the affected CRDs in a one-out-of-two trip configuration, which precludes a single

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

failure, which in turn could prevent tripping of the reactor. The 48 hour Completion Time has been shown to be acceptable through operating experience.

Condition B

Condition B represents a loss of redundancy for the CRD trip Function. Condition B applies when:

- One or more CRD trip breaker(s) [or breaker pair] will not function on either undervoltage or shunt trip Functions; or
- Both diverse trip Functions are inoperable in one or both DC trip breakers associated with one protection channel.

B.1 and B.2

Required Action B.1 and Required Action B.2 are the same as Required Action A.1 and Required Action A.2, but the Completion Time is shortened. The 1 hour Completion Time allowed to trip or remove power from the CRD trip breaker allows the operator to take all the appropriate actions for the inoperable breaker and still ensures that the risk involved is acceptable.

C.1 and C.2

Condition C represents a loss of redundancy for the CRD trip Function. Condition C applies when one or more ETA relays are inoperable. The preferred action is to restore the ETA relay to OPERABLE status. If this cannot be done, the operator can perform one of two actions to eliminate reliance on the failed ETA relay. This first option is to switch the affected control rod group to an alternate power supply. This removes the failed ETA relay from the trip sequence, and the unit can operate indefinitely. The second option is to trip the corresponding AC CRD trip breaker. This results in the safety function being performed, thereby eliminating the failed ETA relay from the trip sequence.

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

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

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

If the Required Actions of Condition A, B, or C are not met within the required Completion Time in MODE 1, 2, or 3, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3, with all CRD trip breakers open or with all power to the CRD System removed within 6 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging unit systems.

E.1 and E.2

If the Required Actions of Condition A, B, or C are not met within the required Completion Time in MODE 4 or 5, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, all CRD trip breakers must be opened or all power to the CRD System removed within 6 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to open all CRD trip breakers or remove all power to the CRD System without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.3.4.1

SR 3.3.4.1 is to perform a CHANNEL FUNCTIONAL TEST every 31 days. This test verifies the OPERABILITY of the trip devices by actuation of the end devices. Also, this test independently verifies the undervoltage and shunt trip mechanisms of the AC breakers. The Frequency of 31 days is based on operating experience, which has demonstrated that failure of more than one channel of a given function in any 31 day interval is a rare event.

REFERENCES

1. FSAR, Chapter [7].

B 3.3 INSTRUMENTATION

B 3.3.5 Engineered Safety Feature Actuation System (ESFAS) Instrumentation

BASES

BACKGROUND

The ESFAS initiates necessary safety systems, based on the values of selected unit Parameters, to protect against violating core design limits and reactor coolant pressure boundary and to mitigate accidents.

ESFAS actuates the following systems:

- High pressure injection (HPI) Actuation;
- Low pressure injection (LPI) Actuation;
- Reactor building (RB) Cooling;
- RB Spray;
- RB Isolation; and
- Emergency diesel generator (EDG) Start.

ESFAS also provides a signal to the Emergency Feedwater Isolation and Control (EFIC) System. This signal initiates emergency feedwater (EFW) when HPI is initiated.

The ESFAS operates in a distributed manner to initiate the appropriate systems. The ESFAS does this by determining the need for actuation in each of three channels monitoring each actuation Parameter. Once the need for actuation is determined, the condition is transmitted to automatic actuation logics, which perform the two-out-of-three logic to determine the actuation of each end device. Each end device has its own automatic actuation logic, although all automatic actuation logics take their signals from the same point in each channel for each Parameter.

Four Parameters are used for actuation:

- Low Reactor Coolant System (RCS) Pressure;
- Low Low RCS Pressure;
- High RB Pressure; and

(continued)

BASES

BACKGROUND (continued)

- High High RB Pressure.

LCO 3.3.5 covers only the instrumentation channels that measure these Parameters. These channels include all intervening equipment necessary to produce actuation before the measured process Parameter exceeds the limits assumed by the accident analysis. This includes sensors, bistable devices, operational bypass circuitry, block timers, and output relays. LCO 3.3.6, "Engineered Safety Feature Actuation System (ESFAS) Manual Initiation," and LCO 3.3.7, "Engineered Safety Feature Actuation System (ESFAS) Automatic Actuation Logic," provide requirements on the manual initiation and automatic actuation logic Functions.

The ESFAS consists of three protection channels. Each channel provides input to logics that initiate equipment with a two-out-of-three logic on each component. Each protection channel includes bistable inputs from one instrumentation channel of Low RB Pressure, Low Low RCS Pressure, High RB Pressure, and High High RB Pressure. Automatic actuation logics combine the three protection channel trips in each train to actuate the individual Engineered Safety Feature (ESF) components needed to initiate each ESF System. Figure [ ], FSAR, Chapter [7] (Ref. 1), illustrates how instrumentation channel trips combine to cause protection channel trips.

The RCS pressure sensors are common to both trains. Isolation is provided via separate bistables for each train. Separate RB pressure sensors are used for the high and high high pressure Functions in each train, and separate sensors are used for each train.

The following matrix identifies the measurement channels and the Function actuated by each.

(continued)

BASES

BACKGROUND (continued)

PARAMETER	LOW RCS PRESSURE	LOW LOW RCS PRESSURE	HIGH RB PRESSURE	HIGH HIGH RB PRESSURE
HPI	X	X	X	
LPI		X		X
RB Cooling	X	X	X	(b)
RB Spray	(b)			
RB Isolation[(a)]	X	X	X	
EDG Start	X	X	X	
Control Room Isolation			X	

(a) Only isolates systems not required for RB or RCS heat removal.

(b) Actuates on High High RB Pressure coincident with HPI actuation.

Engineered safeguards bus undervoltage will also sequence on the HPI loads started by the HPI block timers. However, HPI will not occur unless the ESFAS HPI signal is also present. LCO 3.3.8, "Emergency Diesel Generator (EDG) Loss of Power Start (LOPS)," contains the requirements for the undervoltage channels.

The ESF equipment is divided between the two redundant actuation trains A and B. The division of the equipment between the two actuation trains is based on the equipment redundancy and function and is accomplished in such a manner that the failure of one of the actuation channels and the related safeguards equipment will not inhibit the overall ESF Functions. Where a motor operated or a solenoid operated valve is driven by either of two matrices, one is from actuation channel A and one from actuation channel B. Redundant ESF pumps are controlled from separate and independent actuation channels.

The actuation of ESF equipment is also available by manual actuation switches located on the control room console.

(continued)

BASES

BACKGROUND (continued)

The ESFAS, in conjunction with the actuated equipment, provides protective functions necessary to mitigate Design Basis Accidents (DBAs), specifically the loss of coolant accident (LOCA) and steam line break (SLB) events. The ESFAS relies on the OPERABILITY of the automatic actuation logic for each component to perform the actuation of the selected systems of LCO 3.3.7.

Engineered Safety Feature Actuation System Bypasses

No provisions are made for maintenance bypass of ESFAS instrumentation channels. Operational bypass of certain channels is necessary to allow accident recovery actions to continue and, for some channels, to allow reactor shutdown without spurious ESFAS actuation.

The ESFAS RCS pressure instrumentation channels include permissive bistables that allow manual bypass when reactor pressure is below the point at which the low and low low pressure trips are required to be OPERABLE. Once permissive conditions are sensed, the RCS pressure trips may be manually bypassed. Bypasses are automatically removed when bypass permissive conditions are exceeded.

Each High RB Pressure channel may be manually bypassed after the other two channels in the Parameter have tripped. The manual bypass allows operators to take manual control of ESF Functions after initiation to allow recovery actions. The bypass may be manually removed and is automatically removed when RB pressure returns to below the trip setpoint.

Reactor Coolant System Pressure

The RCS pressure is monitored by three independent pressure transmitters located in the RB. These transmitters are separate from the transmitters that feed the Reactor Protection System (RPS). Each of the pressure signals generated by these transmitters is monitored by four bistables to provide two trip signals, at 1500 psig and 500 psig, and two bypass permissive signals, at 1700 psig and 900 psig.

The outputs of the three bistables, associated with the low RCS pressure, 1500 psig, trip drive relays in two sets

(continued)

BASES

BACKGROUND

Reactor Coolant System Pressure (continued)

(actuation channels A and B) of identical and independent channels. These two sets of HPI channels each use three logic channels used in two-out-of-three coincidence networks for HPI Actuation. The outputs of the three bistables associated with the Low Low RCS Pressure [500 psig] trip drive relays in two sets (actuation channels A and B) of identical and independent channels. These two sets of LPI channels each use three logic channels used in two-out-of-three coincidence networks for LPI Actuation. The outputs of the three Low Low RCS Pressure bistables also trip the drive relays in the corresponding HPI Actuation channel as previously described.

Reactor Building Pressure

RB pressure inputs to the ESFAS are provided by 12 pressure switches. Six pressure switches are used for the High RB Pressure Parameter, and six pressure switches are used for the High High Pressure Parameter.

The output contacts of six High RB Pressure switches are used in two sets of identical and independent actuation trains. These two trains each use three logic channels. The outputs of these channels are used in two-out-of-three coincidence networks. The output contacts of the six RB pressure switches also trip the drive relays in the corresponding HPI and LPI Actuation channels as previously described.

The output contacts of six High High RB Pressure switches are used in two sets of identical and independent actuation trains. These two trains each use three logic channels (RB4, RB5, and RB6). The outputs of these channels are used in two-out-of-three coincident networks for RB Spray Actuation. Each high high pressure train actuates one RB Spray train when the High High RB signal and the HPI signal are coincident in that train.

Trip Setpoints and Allowable Values

Trip setpoints are the nominal value at which the bistables are set. Any bistable is considered to be properly adjusted

(continued)

BASES

BACKGROUND

Trip Setpoints and Allowable Values (continued)

when the "as left" value is within the band for CHANNEL CALIBRATION accuracy (i.e., ± [rack calibration + comparator setting accuracy]).

The trip setpoints used in the bistables are based on the analytical limits stated in Figure [ ], FSAR, Chapter [7] (Ref. 1). The selection of these trip setpoints is such that adequate protection is provided when all sensor and processing time delays are taken into account. To allow for calibration tolerances, instrumentation uncertainties, instrument drift, and severe environment induced errors for those ESFAS channels that must function in harsh environments as defined by 10 CFR 50.49 (Ref. 2), the Allowable Values specified in Table 3.3.5-1 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 "Unit Specific Setpoint Methodology" (Ref. 3). The actual nominal trip setpoint entered into the bistable is more conservative than that specified by the Allowable Value to account for changes in random measurement errors detectable by a CHANNEL FUNCTIONAL TEST. One example of such a change in measurement error is drift during the surveillance interval. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value.

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

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

The Allowable Values listed in Table 3.3.5-1 are based on the methodology described in FSAR, Chapter [14] (Ref. 4), which incorporates all of the known uncertainties applicable for each channel. The magnitudes of these uncertainties are

(continued)

BASES

BACKGROUND

Trip Setpoints and Allowable Values (continued)

factored into the determination of each trip setpoint. All field sensors and signal processing equipment for these channels are assumed to operate within the allowances of these uncertainty magnitudes.

[Reviewer's Note: The ESFAS LCOs in the BWOG Standard Technical Specifications are based on a system representative of the Crystal River Unit 3 design.] As discussed earlier, this arrangement involves measurement channels shared among all actuation functions, with separate actuation logic channels for each actuated component. In this arrangement, multiple components are affected by each instrumentation channel failure, but a single automatic actuation logic failure affects only one component. The organization of BWOG STS ESFAS LCOs reflects the described logic arrangement by identifying instrumentation requirements on an instrumentation channel rather than on a protective function basis. This greatly simplifies delineation of ESFAS LCOs. Furthermore, the LCO requirements on instrumentation channels, automatic actuation logics, and manual initiation are specified separately to reflect the different impact each has on ESFAS OPERABILITY.

APPLICABLE SAFETY ANALYSES

The following ESFAS Functions have been assumed within the accident analyses.

High Pressure Injection

The ESFAS actuation of HPI has been assumed for core cooling in the LOCA analysis and is credited with boron addition in the SLB analysis.

Low Pressure Injection

The ESFAS actuation of LPI has been assumed for large break LOCAs.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Reactor Building Spray, Reactor Building Cooling, and Reactor Building Isolation

The ESFAS actuation of the RB coolers and RB Spray have been credited in RB analysis for LOCAs, both for RB performance and equipment environmental qualification pressure and temperature envelope definition. Accident dose calculations have credited RB Isolation and RB Spray.

Emergency Diesel Generator Start

The ESFAS initiated EDG Start has been assumed in the LOCA analysis to ensure that emergency power is available throughout the limiting LOCA scenarios.

The small and large break LOCA analyses assume a conservative 35 second delay time for the actuation of HPI and LPI in FSAR, Chapter [14] (Ref. 4). This delay time includes allowances for EDG starting, EDG loading, Emergency Core Cooling Systems (ECCS) pump starts, and valve openings. Similarly, the RB Cooling, RB Isolation, and RB Spray have been analyzed with delays appropriate for the entire system analyzed. Typical values used in the analysis are 35 seconds for RB Cooling, 60 seconds for RB Isolation, and 56 seconds for RB Spray.

Accident analyses rely on automatic ESFAS actuation for protection of the core temperature and containment pressure limits and for limiting off site dose levels following an accident. These include LOCA, SLB, and feedwater line break events that result in RCS inventory reduction or severe loss of RCS cooling.

The ESFAS channels satisfy Criterion 3 of the NRC Policy Statement.

LCO

The LCO requires three channels of ESFAS instrumentation for each Parameter in Table 3.3.5-1 to be OPERABLE in each ESFAS train. Failure of any instrument renders the affected channel(s) inoperable and reduces the reliability of the affected Functions.

(continued)

BASES

LCO (continued)

Only the Allowable Value is specified for each ESFAS Function in the LCO. Nominal trip setpoints are specified in the unit specific setpoint calculations. The nominal trip setpoints are selected to ensure the setpoints measured by CHANNEL FUNCTIONAL TESTS do not exceed the Allowable Value if the bistable is performing as required. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable provided that operation and testing are consistent with the assumptions of the unit specific setpoint calculations. Each Allowable Value specified is more conservative than the analytical limit assumed in the safety analysis to account for instrument uncertainties appropriate to the trip Parameter. These uncertainties are defined in the "Unit Specific Setpoint Methodology" (Ref. 3).

The Allowable Values for bypass removal functions are stated in the Applicable MODES or Other Specified Condition column of Table 3.3.5-1.

Three ESFAS instrumentation channels shall be OPERABLE in each ESFAS train to ensure that a single failure in one channel will not result in loss of the ability to automatically actuate the required safety systems.

The bases for the LCO on ESFAS Parameters include the following.

<u>Reactor Coolant System Pressure</u>

Three channels each of RCS Pressure—Low and RCS Pressure—Low Low are required OPERABLE in each train. Each channel includes a sensor, trip bistable, bypass bistable, bypass relays, output relays, and block timers. The analog portion of each pressure channel is common to both trains of both RCS Pressure Parameters. Therefore, failure of one analog channel renders one channel of the low pressure and low low pressure Functions in each train inoperable. The bistable portions of the channels are Function and train specific. Therefore, a bistable failure renders only one Function in one train inoperable. Failure of a bypass bistable or bypass circuitry, such that a trip channel cannot be bypassed, does not render the channel inoperable. Output relays and block timer relays are train specific but may be shared among Parameters. Therefore, output or block

(continued)

BASES

LCO

Reactor Coolant System Pressure (continued)

timer relay failure renders all affected Functions in one train inoperable.

1. Reactor Coolant System Pressure—Low Setpoint

The RCS Pressure—Low Setpoint is based on HPI actuation for small break LOCAs. The setpoint ensures that the HPI will be actuated at a pressure greater than or equal to the value assumed in accident analyses plus the instrument uncertainties. The maximum value assumed for the setpoint of the RCS Pressure—Low trip of HPI in safety analyses is 1480 psig. The setpoint for the low RCS and Allowable Value of ≥ [1600] psig for the low pressure Parameter is selected to ensure actuation occurs when actual RCS pressure is above 1480 psig. The RCS Pressure instrumentation must function while subject to the severe environment created by a LOCA. Therefore, the trip setpoint Allowable Value accounts for severe environment induced errors.

To ensure the RCS Pressure—Low trip is not bypassed when required to be OPERABLE by the safety analysis, each channel's bypass removal bistable must be set with an Allowable Value of ≤ [1800] psig. The bypass removal does not need to function for accidents initiated from RCS Pressures below the bypass removal setpoint. Therefore, the bypass removal setpoint Allowable Value need not account for severe environment induced errors.

2. Reactor Coolant System Pressure—Low Low Setpoint

The RCS Pressure—Low Low Setpoint LPI actuation occurs in sufficient time to ensure LPI flow prior to the emptying of the core flood tanks during a large break LOCA. The Allowable Value of ≥ [400] psig ensures sufficient overlap of the core flood tank flow and the LPI flow to keep the reactor vessel downcomer full during a large break LOCA. The RCS Pressure instrumentation must function while subject to the severe environment created by a LOCA. Therefore, the trip setpoint Allowable Value accounts for severe environment induced errors.

(continued)

BASES

LCO

2. Reactor Coolant System Pressure—Low Low Setpoint (continued)

To ensure the RCS Pressure—Low Low trip is not bypassed when assumed OPERABLE by the safety analysis, each channel's bypass removal bistable must be set with an Allowable Value of ≤ [900] psig. The bypass removal does not need to function for accidents initiated by RCS Pressure below the bypass removal setpoint. Therefore, the bypass removal setpoint Allowable Value need not account for severe environment induced errors.

Reactor Building Pressure

Three channels each of RCS Pressure—Low and RB Pressure—High are required to be OPERABLE in each train. Each channel includes a pressure switch, bypass relays, and output relays. The high pressure channels also include block timers. Each pressure switch is Function and train specific, so there are 12 pressure switches total. Therefore, a pressure switch renders only one Function in one train inoperable. Output relays and block timer relays are train specific but may be shared among Parameters. Therefore, output or block timer relay failure renders all affected Functions in one train inoperable.

The RB Pressure switches may be subjected to high radiatior conditions during the accidents that they are intended to mitigate. The sensor portion of the switches is also exposed to the steam environment present in the RB following a LOCA or high energy line break. Therefore, the trip setpoint Allowable Value accounts for measurement errors induced by these environments.

1. Reactor Building Pressure—High Setpoint

The RB Pressure—High Setpoint Allowable Value ≤ [5] psig was selected to be low enough to detect a rise in RB Pressure that would occur due to a small break LOCA, thus ensuring that the RB high pressure actuation of the safety systems will occur for a wide spectrum of break sizes. The trip setpoint also causes the RB coolers to shift to emergency mode to prevent damage to the cooler fans due to the increase

(continued)

BASES

LCO

1. Reactor Building Pressure—High Setpoint (continued)

in the density of the air steam mixture present in the containment following a LOCA.

2. Reactor Building Pressure—High High Setpoint

The RB Pressure—High High Setpoint Allowable Value ≤ [30] psig was chosen to be high enough to avoid actuation during an SLB, but also low enough to ensure a timely actuation during a large break LOCA.

APPLICABILITY

Three channels of ESFAS instrumentation for each Parameter listed next shall be OPERABLE in each ESFAS train.

1. Reactor Coolant System Pressure—Low Setpoint

The RCS Pressure—Low Setpoint actuation Parameter shall be OPERABLE during operation above 1800 psig. This requirement ensures the capability to automatically actuate safety systems and components during conditions indicative of a LOCA or secondary unit overcooling. Below 1800 psig, the low RCS Pressure actuation Parameter can be bypassed to avoid actuation during normal unit cooldowns when safety systems actuations are not required.

The allowance for the bypass is consistent with the transition of the unit to a lower energy state, providing greater margins to safety limits. The unit response to any event, given that the reactor is already tripped, will be less severe and allows sufficient time for operator action to provide manual safety system actuations. This is even more appropriate during unit heatups when the primary system and core energy content is low, prior to power operation.

In MODES 5 and 6, there is adequate time for the operator to evaluate unit conditions and respond by manually starting individual systems, pumps, and other equipment to mitigate the consequences of an abnormal condition or accident. Plant pressure and temperature

(continued)

BASES

APPLICABILITY

1. Reactor Coolant System Pressure—Low Setpoint (continued)

are very low, and many ESF components are administratively locked out or otherwise prevented from actuating to prevent inadvertent overpressurization of unit systems.

2. Reactor Coolant System Pressure—Low Low Setpoint

The RCS Pressure—Low Low Setpoint actuation Parameter shall be OPERABLE during operation above [900] psig. This requirement ensures the capability to automatically actuate safety systems and components during conditions indicative of a LOCA or secondary unit overcooling. Below [900] psig, the low low RCS Pressure actuation Parameter can be bypassed to avoid actuation during normal unit cooldowns when safety system actuations are not required.

The allowance for the bypass is consistent with the transition of the unit to a lower energy state, providing greater margins to safety limits. The unit response to any event, given that the reactor is already tripped, will be less severe and allows sufficient time for operator action to provide manual safety system actuations. This is even more appropriate during unit heatups when the primary system and core energy content is low, prior to power operation.

In MODES 5 and 6, there is adequate time for the operator to evaluate unit conditions and respond by manually starting individual systems, pumps, and other equipment to mitigate the consequences of an abnormal condition or accident. Plant pressure and temperature are very low, and many ESF components are administratively locked out or otherwise prevented from actuating to prevent inadvertent overpressurization of unit systems.

(continued)

BASES

APPLICABILITY (continued)

3, 4. <u>Reactor Building Pressure—High and Reactor Building Pressure—High High Setpoints</u>

The RB Pressure—High and RB Pressure—High High actuation Functions of ESFAS shall be OPERABLE in MODES 1, 2, 3, and 4 when the potential for a HELB exists. In MODES 5 and 6, the unit conditions are such that there is insufficient energy in the primary and secondary systems to raise the containment pressure to either the RB Pressure—High or RB Pressure—High High Setpoints. Furthermore, in MODES 5 and 6, there is adequate time for the operator to evaluate unit conditions and respond by manually starting individual systems, pumps, and other equipment to mitigate the consequences of an abnormal condition or accident. Plant pressure and temperature are very low and many ESF components are administratively locked out or otherwise prevented from actuating to prevent inadvertent overpressurization of unit systems.

ACTIONS

Required Actions A and B apply to all ESFAS instrumentation Parameters listed in Table 3.3.5-1.

A Note has been added to the ACTIONS indicating separate Condition entry is allowed for each Parameter.

If a channel's trip setpoint is found nonconservative with respect to the Allowable Value, or the transmitter, instrument loop, signal processing electronics, or ESFAS bistable is found inoperable, then all affected functions provided by that channel should be declared inoperable and the unit must enter the Conditions for the particular protection Parameter affected.

When the number of inoperable channels in a trip Parameter exceeds those specified, then the unit is outside the safety analysis. Therefore, LCO 3.0.3 shall be immediately entered if applicable in the current MODE of operation.

(continued)

BASES

ACTIONS (continued)

A.1

Condition A applies when one channel becomes inoperable in one or more Parameters. If one ESFAS channel is inoperable, placing it in a tripped condition leaves the system in a one-out-of-two condition for actuation. Thus, if another channel were to fail, the ESFAS instrumentation could still perform its actuation functions. This action is completed when all of the affected output relays and block timers are tripped. This can normally be accomplished by tripping the affected bistables or tripping the individual output relays and block timers. [At this unit, the specific output relays associated with each ESFAS instrumentation channel are listed in the following document:]

The 1 hour Completion Time is sufficient time to perform the Required Action.

B.1, B.2.1, B.2.2, and B.2.3

Condition B applies when Required Action A.1 is not met within the required Completion Time. If Required Action A.1 cannot be met within the required Completion Time, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and, for the RCS Pressure—Low Setpoint, to < [1800] psig, for the RCS Pressure—Low Low Setpoint, to < [900] psig, and for the RB Pressure High Setpoint and High High Setpoint, to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

All ESFAS Parameters listed in Table 3.3.5-1 are subject to CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, CHANNEL CALIBRATION, and response time testing. The operational bypasses associated with each ESFAS instrumentation channel are also subject to these SRs to ensure OPERABILITY of the ESFAS instrumentation channel.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.5.1

Performance of the CHANNEL CHECK every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; therefore, it is key in verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the unit staff, based on a combination of the channel instrument uncertainties, including isolation, indication, and readability. If a channel is outside the criteria, it may be an indication that the transmitter or the signal processing equipment has drifted outside its limit. If the channels are normally off scale during times when surveillance is required, the CHANNEL CHECK will only verify that they are off scale in the same direction. Off scale low current loop channels are verified to be reading at the bottom of the range and not failed downscale.

The Frequency, about once every shift, is based on operating experience that demonstrates channel failure is rare. Since the probability of two random failures in redundant channels in any 12 hour period is extremely low, the CHANNEL CHECK minimizes the chance of loss of protective function due to failure of redundant channels. The CHANNEL CHECK supplements less formal, but more frequent, checks of channel operability during normal operational use of the displays associated with the LCO's required channels.

SR 3.3.5.2

A Note defines a channel as being OPERABLE for up to 8 hours while bypassed for Surveillance testing provided the remaining two ESFAS channels are OPERABLE or tripped. The Note allows channel bypass for testing without defining it

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.5.2 (continued)

as inoperable, although during this time period it cannot initiate ESFAS. This allowance is based on the inability to perform the Surveillance in the time permitted by the Required Actions. Eight hours is the average time required to perform the Surveillance. It is not acceptable to routinely remove channels from service for more than 8 hours to perform required Surveillance testing.

A CHANNEL FUNCTIONAL TEST is performed on each required ESFAS channel to ensure the entire channel will perform the intended functions. Any setpoint adjustment shall be consistent with the assumptions of the current unit specific setpoint analysis.

The Frequency of 31 days is based on unit operating experience, with regard to channel OPERABILITY and drift, which demonstrates that failure of more than one channel of a given function in any 31 day interval is a rare event.

SR 3.3.5.3

CHANNEL CALIBRATION is a complete check of the instrument channel, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift to ensure that the instrument channel remains operational between successive tests. CHANNEL CALIBRATION shall find that measurement errors and bistable setpoint errors are within the assumptions of the unit specific setpoint analysis. CHANNEL CALIBRATIONS must be performed consistent with the assumptions of the unit specific setpoint analysis.

This Frequency is justified by the assumption of an [18] month calibration interval to determine the magnitude of equipment drift in the setpoint analysis.

SR 3.3.5.4

SR 3.3.5.4 ensures that the ESFAS actuation channel response times are less than or equal to the maximum times assumed in the accident analysis. The response time values are the

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.5.4 (continued)

maximum values assumed in the safety analyses. Individual component response times are not modeled in the analyses. Response time testing acceptance criteria for this unit are included in Reference 1. The analyses model the overall or total elapsed time from the point at which the parameter exceeds the actuation setpoint value at the sensor to the point at which the end device is actuated. Thus, this SR encompasses the automatic actuation logic components covered by LCO 3.3.7 and the operation of the mechanical ESF components.

Response time tests are conducted on an [18] month STAGGERED TEST BASIS. Testing of the final actuation devices, which make up the bulk of the response time, is included in the testing of each channel. Therefore, staggered testing results in response time verification of these devices every [18] months. The 18 month test Frequency is based on unit operating experience, which shows that random failures of instrumentation components causing serious response time degradation but not channel failure are infrequent occurrences.

REFERENCES

1. FSAR, Chapter [7].
2. 10 CFR 50.49.
3. "Unit Specific Setpoint Methodology."
4. FSAR, Chapter [14].

B 3.3 INSTRUMENTATION

B 3.3.6 Engineered Safety Feature Actuation System (ESFAS) Manual Initiation

BASES

BACKGROUND

The ESFAS manual initiation capability allows the operator to actuate ESFAS Functions from the main control room in the absence of any other initiation condition. Manually actuated Functions include High Pressure Injection, Low Pressure Injection, Reactor Building (RB) Cooling, RB Spray, RB Isolation, and Control Room Isolation. This ESFAS manual initiation capability is provided in the event the operator determines that an ESFAS Function is needed and has not been automatically actuated. Furthermore, the ESFAS manual initiation capability allows operators to rapidly initiate Engineered Safety Feature (ESF) Functions if the trend of unit parameters indicates that ESF actuation will be needed.

LCO 3.3.6 covers only the system level manual initiation of these Functions. LCO 3.3.5, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," and LCO 3.3.7, "Engineered Safety Feature Actuation System (ESFAS) Automatic Actuation Logic," provide requirements on the portions of the ESFAS that automatically initiate the Functions described earlier.

The ESFAS manual initiation Function relies on the OPERABILITY of the automatic actuation logic (LCO 3.3.7) for each component to perform the actuation of the systems. A manual trip push button is provided on the ESF panel of the control room console for each of the levels of protection for each actuation. Operation of the push button energizes relays whose contacts perform a logical "OR" function with the matrices of the automatic actuation, except for the matrices which are part of the ESF buses loading sequence. Manual actuation of the ESF buses loading sequence is made by de-energizing the timed output relays. The power supply for the manual trip relays is taken from the station batteries. Different batteries are used for the two actuations.

The ESFAS manual initiation channel is defined as the instrumentation between the console switch and the automatic actuation logic, which actuates the end devices. Other means of manual initiation, such as controls for individual ESF devices, may be available in the control room and other

(continued)

BASES

BACKGROUND (continued)	unit locations. These alternative means are not required by this LCO, nor may they be credited to fulfill the requirements of this LCO.
APPLICABLE SAFETY ANALYSES	The ESFAS, in conjunction with the actuated equipment, provides protective functions necessary to mitigate Design Basis Accidents, specifically, the loss of coolant accident and steam line break events. The ESFAS manual initiation ensures that the control room operator can rapidly initiate ESF Functions at any time. The manual initiation trip Function is required as a backup to automatic trip functions and allows operators to initiate ESFAS whenever any parameter is rapidly trending toward its trip setpoint. Furthermore, the ESFAS manual initiation may be specified in operating procedures for verification that ESF systems are running. The ESFAS manual initiation functions satisfy Criterion 3 of the NRC Policy Statement.
LCO	Two ESFAS manual initiation channels of each ESFAS Function shall be OPERABLE whenever conditions exist that could require ESF protection of the reactor or RB. Two OPERABLE channels ensure that no single random failure will prevent system level manual initiation of any ESFAS Function. The ESFAS manual initiation Function allows the operator to initiate protective action prior to automatic initiation or in the event the automatic initiation does not occur.
APPLICABILITY	The ESFAS manual initiation Functions shall be OPERABLE in MODES 1, 2, and 3, and in MODE 4 when the associated engineered safeguard equipment is required to be OPERABLE. The manual initiation channels are required because ESF Functions are designed to provide protection in these MODES. In MODES 5 and 6, ESFAS initiates systems that are either reconfigured or disabled for shutdown cooling operation. Accidents in these MODES are slow to develop and would be mitigated by manual operation of individual components. Adequate time is available to evaluate unit conditions and

(continued)

BASES

APPLICABILITY (continued)

to respond by manually operating the ESF components, if required.

ACTIONS

A Note has been added to the ACTIONS indicating separate Condition entry is allowed for each ESFAS manual initiation Function.

A.1

Condition A applies when one manual initiation channel of one or more ESFAS Functions becomes inoperable. Required Action A.1 must be taken to restore the channel to OPERABLE status within the next 72 hours. The Completion Time of 72 hours is based on unit operating experience and administrative controls, which provide alternative means of ESFAS Function initiation via individual component controls. The 72 hour Completion Time is consistent with the allowed outage time for the safety systems actuated by ESFAS.

B.1 and B.2

Required Action B.1 and Required Action B.2 apply if Required Action A.1 cannot be met within the required Completion Time. If Required Action A.1 cannot be met within the required Completion Time, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required MODES from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.3.6.1

This SR requires the performance of a CHANNEL FUNCTIONAL TEST of the ESFAS manual initiation. This test verifies that the initiating circuitry is OPERABLE and will actuate the end device (i.e., pump, valves, etc.). The [18] month Frequency is based on the need to perform this Surveillance

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.6.1 (continued)

under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. This Frequency is demonstrated to be sufficient, based on operating experience, which shows these components usually pass the Surveillance when performed on the [18] month Frequency.

REFERENCES

None.

B 3.3 INSTRUMENTATION

B 3.3.7 Engineered Safety Feature Actuation System (ESFAS) Automatic Actuation Logic

BASES

BACKGROUND

The automatic actuation logic channels of ESFAS are defined as the logic between the buffers of the sensing channels and the controllers that actuate ESFAS equipment. Each of the components actuated by the ESFAS Functions has an associated automatic actuation logic matrix. If two-out-of-three ESFAS instrumentation channels indicate a trip, or system level manual initiation occurs, the automatic actuation logic is activated and the associated component is actuated. The purpose of requiring OPERABILITY of the ESFAS automatic actuation logic is to ensure that the Functions of the ESFAS can be automatically initiated in the event of an accident. Automatic actuation of some Functions is necessary to prevent the unit from exceeding the Emergency Core Cooling Systems (ECCS) limits in 10 CFR 50.46 (Ref. 1). It should be noted that OPERABLE automatic actuation logic channels alone will not ensure that each Function can be activated; the instrumentation channels and actuated equipment associated with each Function must also be OPERABLE to ensure that the Functions can be automatically initiated during an accident.

LCO 3.3.7 covers only the automatic actuation logic that initiates these Functions. LCO 3.3.5, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," and LCO 3.3.6, "Engineered Safety Feature Actuation System (ESFAS) Manual Initiation," provide requirements on the instrumentation and manual initiation channels that input to the automatic actuation logic.

The ESFAS, in conjunction with the actuated equipment, provides protective functions necessary to mitigate Design Basis Accidents (DBAs), specifically, the loss of coolant accident (LOCA) and steam line break (SLB) events. The ESFAS relies on the OPERABILITY of the automatic actuation logic for each component to perform the actuation of the selected systems.

The small and large break LOCA analyses assume a conservative 35 second delay time for the actuation of high pressure injection (HPI) and low pressure injection (LPI) in

(continued)

BASES

BACKGROUND (continued)

BAW-10103A, Rev. 3 (Ref. 2). This delay time includes allowances for emergency diesel generator (EDG) starts, EDG loading, ECCS pump starts, and valve openings. Similarly, the reactor building (RB) Cooling, RB Isolation, and RB Spray have been analyzed with delays appropriate for the entire system.

Typical values used in the analyses are 35 seconds for RB Cooling, 60 seconds for RB Isolation, and 58 seconds for RB Spray.

The ESFAS automatic initiation of Engineered Safety Feature (ESF) Functions to mitigate accident conditions is assumed in the DBA analysis and is required to ensure that consequences of analyzed events do not exceed the accident analysis predictions. Automatically actuated features include HPI, LPI, RB Cooling, RB Spray, and RB Isolation.

The ESFAS LCOs in the BWOG Standard Technical Specifications (STS) are based on a system representative of the Crystal River Unit 3 design. As discussed earlier, this arrangement involves measurement channels shared among all actuation functions, with separate actuation logic channels for each actuated component. In this arrangement, multiple ESF components are affected by a measurement channel failure, but a single automatic actuation logic failure affects only one component. The organization of BWOG STS ESFAS LCOs reflect the described logic arrangement by linking actions for automatic actuation logic failures directly to the actions for the affected ESF component. The overall philosophy is that if an automatic actuation logic fails, the affected component is put into its engineered safeguard configuration. This action eliminates the need for the automatic actuation logic. If the affected component cannot be placed in its engineered safeguard configuration, actions are taken to address the inoperability of the supported system component. This greatly simplifies delineation of ESFAS LCOs. Furthermore, the LCO requirements on instrumentation channels, automatic actuation logics, and manual initiation are specified separately to reflect the different impact each has on ESFAS OPERABILITY.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

Accident analyses rely on automatic ESFAS actuation for protection of the core and RB and for limiting off site dose levels following an accident. These include LOCA, SLB, and feedwater line break events that result in Reactor Coolant System (RCS) inventory reduction or severe loss of RCS cooling. The automatic actuation logic is an integral part of the ESFAS.

The ESFAS automatic actuation logics satisfy Criterion 3 of the NRC Policy Statement.

LCO

The automatic actuation logic matrix for each component actuated by the ESFAS is required to be OPERABLE whenever conditions exist that could require ESF protection of the reactor or the RB. This ensures automatic initiation of the ESF required to mitigate the consequences of accidents.

APPLICABILITY

The automatic actuation logic Function shall be OPERABLE in MODES 1, 2, and 3, and in MODE 4 when the associated engineered safeguard equipment is required to be OPERABLE, because ESF Functions are designed to provide protection in these MODES. Automatic actuation in MODE 5 or 6 is not required because the systems initiated by the ESFAS are either reconfigured or disabled for shutdown cooling operation. Accidents in these MODES are slow to develop and would be mitigated by manual operation of individual components. Adequate time is available to evaluate unit conditions and respond by manually operating the ESF components, if required.

ACTIONS

A Note has been added to the ACTIONS indicating separate Condition entry is allowed for each ESFAS automatic actuation logic matrix.

<u>A.1 and A.2</u>

When one or more automatic actuation logic matrices are inoperable, the associated component(s) can be placed in its engineered safeguard configuration. Required Action A.1 is

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

equivalent to the automatic actuation logic performing its safety function ahead of time. In some cases, placing the component in its engineered safeguard configuration would violate unit safety or operational considerations. In these cases, the component status should not be changed, but the supported system component must be declared inoperable. Conditions which would preclude the placing of a component in its engineered safeguard configuration include, but are not limited to, violation of system separation, activation of fluid systems that could lead to thermal shock, or isolation of fluid systems that are normally functioning. The Completion Time of 1 hour is based on operating experience and reflects the urgency associated with the inoperability of a safety system component.

Required Action A.2 requires entry into the Required Actions of the affected supported systems, since the true effect of automatic actuation logic failure is inoperability of the supported system. The Completion Time of 1 hour is based on operating experience and reflects the urgency associated with the inoperability of a safety system component.

SURVEILLANCE REQUIREMENTS

SR 3.3.7.1

SR 3.3.7.1 is the performance of a CHANNEL FUNCTIONAL TEST on a 31 day STAGGERED TEST BASIS. The test demonstrates that every automatic actuation logic associated with one of the two safety system trains successfully performs the two-out-of-three logic combinations every 31 days. All automatic actuation logics are thus retested every 62 days. The test simulates the required one-out-of-three inputs to the logic circuit and verifies the successful operation of the automatic actuation logic. The Frequency is based on operating experience that demonstrates the rarity of more than one channel failing within the same 31 day interval.

Automatic actuation logic response time testing is incorporated into the response time testing required by LCO 3.3.5.

(continued)

BASES (continued)

REFERENCES

1. 10 CFR 50.46.
2. BAW-10103A, Rev. 3, July 1977.

B 3.3 INSTRUMENTATION

B 3.3.8 Emergency Diesel Generator (EDG) Loss of Power Start (LOPS)

BASES

BACKGROUND

The EDGs provide a source of emergency power when offsite power is either unavailable or is insufficiently stable to allow safe unit operation. Undervoltage protection will generate a LOPS in the event a loss of voltage or degraded voltage condition occurs in the switchyard. There are two LOPS Functions for each 4.16 kV vital bus.

Three undervoltage relays with [inverse voltage time] characteristics are provided on each 4.16 kV Class 1E instrument bus for the purpose of detecting a sustained undervoltage condition or a loss of bus voltage. The relays are combined in a two-out-of-three logic to generate a LOPS if the voltage is below 75% for a short time or below 90% for a long time. The LOPS initiated ACTIONS are described in FSAR, Section [8.3] (Ref. 1).

Trip Setpoints and Allowable Value

The trip setpoints used in the bistables are based on the analytical limits presented in accident analysis in FSAR, Chapter [14] (Ref. 2). The selection of these trip setpoints is such that adequate protection is provided when all sensor and processing time delays are taken into account. The actual nominal trip setpoint entered into the bistable is more conservative than that required by the unit specific setpoint calculations. A channel is inoperable if its actuation trip setpoint is not within its required Allowable Value.

Setpoints in accordance with the Allowable Value will assure that limits of Chapter 2.0, "Safety Limits," in the Technical Specifications are not violated during anticipated operational occurrences (AOOs); that the consequences of accidents will be acceptable, providing the unit is operated from within the LCOs at the onset of the AOO or accident; and that the equipment functions as designed.

The undervoltage protection scheme has been designed to protect the unit from spurious trips caused by the offsite power source. This is made possible by the inverse voltage

(continued)

BASES

BACKGROUND

Trip Setpoints and Allowable Value (continued)

time characteristics of the relays used. A complete loss of offsite power will result in approximately a [1] second delay in LOPS actuation. The EDG starts and is available to accept loads within a 10 second time interval on the Engineered Safety Feature Actuation System (ESFAS) or LOPS. Emergency power is established within the maximum time delay assumed for each event analyzed in the accident analysis (Ref. 2).

With three protection channels in a two-out-of-three trip logic for each division of the 4.16 kV power supply, no single failure will cause or prevent protective system actuation. This arrangement meets IEEE-279-1971 criteria (Ref. 3).

APPLICABLE SAFETY ANALYSES

The EDG LOPS is required for the Engineered Safety Features (ESF) to function in any accident with a loss of offsite power. Its design basis is that of the ESFAS.

Accident analyses credit the loading of the EDG, based on the loss of offsite power, during a loss of coolant accident (LOCA). The actual EDG Start has historically been associated with the ESFAS actuation. The diesel loading has been included in the delay time associated with each safety system component requiring EDG supplied power following a loss of offsite power. The analysis assumes a nonmechanistic EDG loading, which does not explicitly account for each individual component of the loss of power detection and subsequent actions. The total actuation time for the limiting systems, high pressure injection, and low pressure injection is 35 seconds. This delay time includes contributions from the EDG Start, EDG loading, and safety injection system component actuation. The response of the EDG to a loss of power must be demonstrated to fall within this analysis response time when including the contributions of all portions of the delay.

The required channels of LOPS, in conjunction with the ESF systems powered from the EDGs, provide unit protection in the event of any of the analyzed accidents discussed in the accident analysis (Ref. 2), in which a loss of offsite power is assumed.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The delay times assumed in the safety analysis for the ESF equipment include the 10 second EDG Start delay and, if applicable, the appropriate sequencing delay. The response times for ESFAS actuated equipment in LCO 3.3.5, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," include the appropriate EDG loading and sequencing delay.

The EDG LOPS channels satisfy Criterion 3 of the NRC Policy Statement.

LCO

The LCO for the LOPS requires that three channels per bus of each LOPS instrumentation Function shall be OPERABLE in MODES 1, 2, 3, and 4 when the LOPS supports safety systems associated with the ESFAS. In MODES 5 and 6, the three channels must be OPERABLE whenever the associated EDG is required to be OPERABLE to ensure that the automatic start of the EDG is available when needed.

Loss of LOPS function could result in the delay of safety systems initiation when required. This could lead to unacceptable consequences during accidents. During the loss of offsite power, which is an AOO, the EDG powers the motor driven emergency feedwater pumps. Failure of these pumps to start would leave only the one turbine driven pump and an increased potential for a loss of decay heat removal through the secondary system.

Only Allowable Values are specified for each Function in the LCO. Nominal trip setpoints are specified in the unit specific setpoint calculations. The nominal setpoints are selected to ensure that the setpoint measured by CHANNEL FUNCTIONAL TESTS does not exceed the Allowable Value if the bistable is performing as required. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within the Allowable Value, is acceptable provided that operation and testing is consistent with the assumptions of the unit specific setpoint calculation. Each Allowable Value specified is more conservative than the analytical limit assumed in the transient and accident analysis to account for instrument uncertainties appropriate to the trip function. These uncertainties are defined in the "[Unit Specific Setpoint Methodology]" (Ref. 4).

(continued)

BASES

LCO (continued)

Degraded Voltage LOPS

Voltage: The minimum Allowable Value includes an allowance for relay coil calibration error and is based on maintaining at least [90%] of rated voltage on the 480 V motor control centers, with a [4.1%] V drop across the [4160/480] V stepdown transformers. The [4.1%] V drop associated with these transformers is the maximum expected due to ESF bus loading, while the MCC contactors are considered to require at least [90%] V for proper operation.

The maximum Allowable Value is not based on equipment operability concerns, but rather avoidance of unnecessary EDG starts due to spurious channel trip.

Time Delay: The response time includes [5 seconds] for undervoltage relay actuation following detection of degraded ES bus voltage, [13 seconds] for the bus trip delay timer, and [3 seconds] for the dead bus timer. Note that the acceptance criteria of [21 seconds] does not account for the setpoint tolerance of [10%] or [± 2.1 seconds].

Loss of Voltage LOPS

Voltage and Response Time: The Allowable Value for the loss of voltage channels is ≥ 0 V. This Allowable Value and the associated channel response time are based on the physical characteristics of the loss of voltage sensing relays. The loss of voltage channels respond to a complete loss of ES bus voltage, providing automatic starting and loading of the associated EDG. However, their response time is not critical to the overall ES equipment response time following an actuation, since the degraded voltage LOPS instrumentation will also respond to the complete loss of voltage, and will do so earlier than the loss of voltage instrumentation. The loss of voltage channel response includes only the time response associated with the undervoltage relays, including the nominal setpoint of [7.8 seconds] and a tolerance of [7%] or [0.55 seconds].

APPLICABILITY

The EDG LOPS actuation Function shall be OPERABLE in MODES 1, 2, 3, and 4 because ESF Functions are designed to provide protection in these MODES. Actuation is also

(continued)

BASES

APPLICABILITY (continued)

required whenever the EDG is required to be OPERABLE by LCO 3.8.2, "AC Sources—Shutdown," so that the EDG can perform its function on a loss of power or degraded power to the vital bus.

ACTIONS

If a channel's trip setpoint is found nonconservative with respect to the Allowable Value, or the channel is found inoperable, then the function that the channel provides must be declared inoperable and the LCO Condition entered for the particular protection function affected. Since the required channels are specified on a per EDG basis, the Condition may be entered separately for each EDG.

A Note has been added to the ACTIONS indicating that separate Condition entry is allowed for each Function.

A.1

If one channel per EDG in one or more Functions is inoperable, it must be tripped within 1 hour. With a channel in trip, the LOPS channels are configured to provide a one-out-of-two logic to initiate a trip of the incoming offsite power. In trip, one additional valid actuation will cause a LOPS signal on the bus. The 1 hour Completion Time is reasonable to evaluate and to take action by correcting a degraded condition in an orderly manner and takes into account the low probability of an event requiring LOPS occurring during this interval.

B.1

Condition B applies when two or more undervoltage or two or more degraded voltage channels on a single bus are inoperable.

Required Action B.1 requires all but one inoperable channel to be restored to OPERABLE status within 1 hour. With two or more channels inoperable, the logic is not capable of providing an automatic EDG LOPS signal for valid loss of voltage or degraded voltage conditions. The 1 hour Completion Time is reasonable to evaluate and to take action by correcting the degraded condition in an orderly manner

(continued)

BASES

ACTIONS

B.1 (continued)

and takes into account the low probability of an event requiring LOPS occurring during this interval.

C.1

Condition C applies if the Required Action of Condition A or Condition B and the associated Completion Time is not met.

Required Action C.1 ensures that Required Actions for affected diesel generator inoperabilities are initiated. Depending on unit MODE, the Actions specified in LCO 3.8.1, "AC Sources—Operating," or LCO 3.8.2, are required immediately.

SURVEILLANCE REQUIREMENTS

SR 3.3.8.1

SR 3.3.8.1 is the performance of the CHANNEL CHECK once every 12 hours to ensure that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; therefore, it is key in verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the unit staff based on a combination of the channel instrument uncertainties, including isolation, indication, and readability. If a channel is outside the criteria, it may be an indication that the transmitter or the signal processing equipment has drifted outside its limit. If the channels are within the criteria, it is an indication that the channels are OPERABLE.

The Frequency, about once every shift, is based on operating experience that demonstrates channel failure is rare. Since

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.8.1 (continued)

the probability of two random failures in redundant channels in any 12 hour period is low, the CHANNEL CHECK minimizes the chance of loss of protective function due to failure of redundant channels. The CHANNEL CHECK supplements less formal, but more frequent, checks of channel OPERABILITY during normal operational use of the displays associated with this LCO's required channels.

SR 3.3.8.2

The Note allows channel bypass for testing without defining it as inoperable although during this time period it cannot actuate a diesel start. This allowance is based on the assumption that 4 hours is the average time required to perform channel Surveillance. The 4 hour testing allowance does not significantly reduce the probability that the EDG will start trip when necessary. It is not acceptable to routinely remove channels from service for more than 4 hours to perform required Surveillance testing.

A CHANNEL FUNCTIONAL TEST is performed on each required EDG LOPS channel to ensure the entire channel will perform the intended function. Any setpoint adjustments shall be consistent with the assumptions of the current unit specific setpoint analysis. The Frequency of 31 days is considered reasonable based on the reliability of the components and on operating experience that demonstrates channel failure is rare.

SR 3.3.8.3

A CHANNEL CALIBRATION is a complete check of the instrument channel, including the sensor. The setpoints and the response to a loss of voltage and a degraded voltage test shall include a single point verification that the trip occurs within the required delay time, as shown in Reference 1. CHANNEL CALIBRATION shall find that measurement setpoint errors are within the assumptions of the unit specific setpoint analysis. CHANNEL CALIBRATIONS must be performed consistent with the assumptions of the unit specific setpoint analysis in Reference 4.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.8.3 (continued)

The Frequency is based on operating experience and consistency with the typical industry refueling cycle and is justified by the assumption of an 18 month calibration interval in the determination of equipment drift in the setpoint calculation.

REFERENCES

1. FSAR, Section [8.3].
2. FSAR, Chapter [14].
3. IEEE-279-1971, April 1972.
4. [Unit Name], "[Unit Specific Setpoint Methodology]."

B 3.3 INSTRUMENTATION

B 3.3.9 Source Range Neutron Flux

BASES

BACKGROUND

The source range neutron flux channels provide the operator with an indication of the approach to criticality at lower power levels than can be seen on the intermediate range neutron flux instrumentation. These channels also provide the operator with a flux indication that reveals changes in reactivity and helps to verify that SDM is being maintained.

The source range instrumentation has two redundant count rate channels originating in two high sensitivity proportional counters. Two source range detectors are externally located on opposite sides of the core 180°. These channels are used over a counting range of 0.1 cps to 1E6 cps and are displayed on the operator's control console in terms of log count rate. The channels also measure the rate of change of the neutron flux level, which is displayed for the operator in terms of startup rate from -0.5 decades to +5 decades per minute. An interlock provides a control rod withdraw "inhibit" on a high startup rate of +2 decades per minute in either channel.

The proportional counters of the source range channels are BF_3 chambers. The detector high voltage is automatically turned off when the flux level is approximately one decade above the useful operating range. Conversely, the high voltage is turned on automatically when the flux level returns to within approximately one decade of the detectors' maximum useful range. High voltage will be turned off automatically when the flux level is above 1E-10 amp in both intermediate range channels, or 10% power in power range channels.

APPLICABLE SAFETY ANALYSES

The source range neutron flux channels are necessary to monitor core reactivity changes. It is the primary means for detecting and triggering operator actions to respond to reactivity transients initiated from conditions in which the Reactor Protection System (RPS) is not required to be OPERABLE. It also triggers operator actions to anticipate RPS actuation in the event of reactivity transients starting from shutdown or low power conditions.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The source range neutron flux channels satisfy Criterion 2 of the NRC Policy Statement.

LCO

Two source range neutron flux channels shall be OPERABLE whenever the control rods are capable of being withdrawn to provide the operator with redundant source range neutron instrumentation. The source range instrumentation is the primary power indication at low power levels < 1E-10 amp on intermediate range instrumentation and must remain OPERABLE for the operator to continue increasing power.

A Note has been added allowing detector high voltage to be de-energized above 1E-10 amp on the intermediate range channels. Above this point, the source range instrumentation is no longer the primary power indicator. As such, the high voltage to the source range detectors may be de-energized.

APPLICABILITY

Two source range neutron flux channels shall be OPERABLE in MODE 2 to provide redundant indication during an approach to criticality. Neutron flux level is sufficient for monitoring on the intermediate range and on the power range instrumentation prior to entering MODE 1; therefore, source range instrumentation is not required in MODE 1.

In MODES 3, 4, and 5, source range neutron flux instrumentation shall be OPERABLE to provide the operator with a means of monitoring changes in SDM and to provide an early indication of reactivity changes.

The requirements for source range neutron flux instrumentation during MODE 6 refueling operations are addressed in LCO 3.9.2., "Nuclear Instrumentation."

ACTIONS

A.1

The Required Action for one channel of the source range neutron flux indication inoperable with THERMAL POWER ≤ 1E-10 amp on the intermediate range neutron flux

(continued)

BASES

ACTIONS

A.1 (continued)

instrumentation is to delay increasing reactor power until the channel is repaired and restored to OPERABLE status. This limits power increases in the range where the operators rely solely on the source range instrumentation for power indication. The Completion Time ensures the source range is available prior to further power increases. Furthermore, it ensures that power remains below the point where the intermediate range channels provide primary protection until both source range channels are available to support the overlap verification required by SR 3.3.9.4.

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

With both source range neutron flux channels inoperable with THERMAL POWER ≤ 1E-10 amp on the intermediate range neutron flux instrumentation, the operators must place the reactor in the next lowest condition for which source range instrumentation is not required. This is done by immediately suspending positive reactivity additions, initiating action to insert all CONTROL RODS, and opening the CONTROL ROD drive trip breakers within 1 hour. Periodic SDM verification of ≥ 1% Δk/k is then required to provide a means for detecting the slow reactivity changes that could be caused by mechanisms other than control rod withdrawal or operations involving positive reactivity changes. Since the source range instrumentation provides the only reliable direct indication of power in this condition, the operators must continue to verify the SDM every 12 hours until at least one channel of the source range instrumentation is returned to OPERABLE status. Required Action B.1, Required Action B.2, and Required Action B.3 preclude rapid positive reactivity additions. The 1 hour Completion Time for Required Action B.3 and Required Action B.4 provides sufficient time for operators to accomplish the actions. The 12 hour Frequency for performing the SDM verification ensures that the reactivity changes possible with CONTROL RODS inserted are detected before SDM limits are challenged.

C.1

With reactor power > 1E-10 amp in MODE 2, 3, 4, or 5 on the intermediate range neutron flux instrumentation, continued

(continued)

BASES

ACTIONS

C.1 (continued)

operation is allowed with one or more source range neutron flux channels inoperable. The ability to continue operation is justified because the instrumentation does not provide a safety function during high power operation. However, actions are initiated within 1 hour to restore the channel(s) to OPERABLE status for future availability. The Completion Time of 1 hour is sufficient to initiate the action. The action must continue until channels are restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

SR 3.3.9.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; therefore, it is key in verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the unit staff, based on a combination of the channel instrument uncertainties, including isolation, indication, and readability. If a channel is outside the criteria, it may be an indication that the transmitter or the signal processing equipment has drifted outside its limit. If the channels are within the criteria, it is an indication that the channels are OPERABLE. If the channels are normally off scale during times when surveillance is required, the CHANNEL CHECK will only verify that they are off scale in the same direction. Off scale low current loop channels are verified to be reading at the bottom of the range and not failed downscale.

The Frequency, about once every shift, is based on operating experience that demonstrates channel failure is rare. Since

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.9.1 (continued)

the probability of two random failures in redundant channels in any 12 hour period is extremely low, the CHANNEL CHECK minimizes the chance of loss of protective function due to failure of redundant channels. The CHANNEL CHECK supplements less formal, but more frequent, checks of channel OPERABILITY during normal operational use of the displays associated with the LCO's required channels. When operating in Required Action A.1, CHANNEL CHECK is still required. However, in this condition, a redundant source range is not available for comparison. CHANNEL CHECK may still be performed via comparison with intermediate range detectors, if available, and verification that the OPERABLE source range channel is energized and indicating a value consistent with current unit status.

SR 3.3.9.2

For source range neutron flux channels, CHANNEL CALIBRATION is a complete check and readjustment of the channels from the preamplifier input to the indicators. This test verifies the channel responds to measured parameters within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift to ensure that the instrument channel remains operational between successive tests.

The SR is modified by a Note excluding neutron detectors from CHANNEL CALIBRATION. It is not necessary to test the detectors because generating a meaningful test signal is difficult. The detectors are of simple construction, and any failures in the detectors will be apparent as change in channel output.

The Frequency of [18] months is based on demonstrated instrument CHANNEL CALIBRATION reliability over an [18] month interval, such that the instrument is not adversely affected by drift.

SR 3.3.9.3

SR 3.3.9.3 is the verification of one decade of overlap with the intermediate range neutron flux instrumentation prior to

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.9.3 (continued)

source range count rate exceeding 10^5 cps if not performed within 7 days prior to reactor startup. This ensures a continuous source of power indication during the approach to criticality. Failure to perform this Surveillance leaves the unit in a safe, subcritical condition until the verification can be made. The test may be omitted if performed within the previous 7 days based on operating experience, which shows that source range and intermediate range instrument overlap does not change appreciably within this test interval.

REFERENCES

None.

B 3.3 INSTRUMENTATION

B 3.3.10 Intermediate Range Neutron Flux

BASES

BACKGROUND

The intermediate range neutron flux channels provide the operator with an indication of reactor power at higher power levels than the source range instrumentation and lower power levels than the power range instrumentation.

The intermediate range instrumentation has two log N channels originating in two electrically identical gamma compensated ion chambers. Each channel provides eight decades of flux level information in terms of the log of ion chamber current from 1E-10 amp to 1E-2 amp. The channels also measure the rate of change of the neutron flux level, which is displayed for the operator in terms of startup rate from -0.5 decades to +5 decades per minute. A high startup rate of +3 decades per minute in either channel will initiate a control rod withdrawal inhibit.

The intermediate range compensated ion chambers are of the electrically adjustable gamma compensating type. Each detector has a separate adjustable high voltage power supply and an adjustable compensating voltage supply.

APPLICABLE SAFETY ANALYSES

Intermediate range neutron flux channels are necessary to monitor core reactivity changes and are the primary indication to trigger operator actions to anticipate Reactor Protection System actuation in the event of reactivity transients starting from low power conditions.

The intermediate range neutron flux channels satisfy Criterion 2 of the NRC Policy Statement.

LCO

Two intermediate range neutron flux instrumentation channels shall be OPERABLE to provide the operator with redundant neutron flux indication. These enable operators to control the increase in power and to detect neutron flux transients. This indication is used until the power range instrumentation is on scale. Violation of this requirement could prevent the operator from detecting and controlling

(continued)

BASES

LCO (continued)

neutron flux transients that could result in reactor trip during power escalation.

APPLICABILITY

The intermediate range neutron flux channels shall be OPERABLE in MODE 2 and when any CONTROL ROD drive (CRD) trip breaker is in the closed position and the CRD System is capable of rod withdrawal.

The intermediate range instrumentation is designed to detect power changes during initial criticality and power escalation when the power range and source range instrumentation cannot provide reliable indications. Since those conditions can exist in all of these MODES, the intermediate range instrumentation must be OPERABLE.

ACTIONS

A.1

If one intermediate range channel becomes inoperable when the channels indicate 1E-10 amp, the unit is exposed to the possibility that a single failure will disable all neutron monitoring instrumentation. To avoid this, the inoperable channel must be repaired or power must be reduced to the point where source range channels can provide neutron flux indication. Completion of Required Action A.1 places the unit in this state, and LCO 3.3.9, "Source Range Neutron Flux," requires OPERABILITY of two source range detectors once this state is reached. If the one channel failure occurs when indicated power is < 1E-10 amp, the Required Action prohibits increases in power above the source range capability.

The 2 hour Completion Time allows controlled reduction of power into the source range and is based on unit operating experience that demonstrates the improbability of the second intermediate range channel failing during the allowed interval.

B.1 and B.2

With two intermediate range neutron flux channels inoperable when THERMAL POWER is ≤ 5% RTP, the operators must place the

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

reactor in the next lowest condition for which the intermediate range instrumentation is not required. This involves providing power level indication on the source range instrumentation by immediately suspending operations involving positive reactivity changes and, within 1 hour, placing the reactor in the tripped condition with the CRD trip breakers open. The Completion Times are based on unit operating experience and allow the operators sufficient time to manually insert the CONTROL RODS prior to opening the CRD breakers.

SURVEILLANCE REQUIREMENTS

SR 3.3.10.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; therefore, it is key in verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the unit staff based on a combination of the channel instrument uncertainties, including isolation, indication, and readability. If a channel is outside the criteria, it may be an indication that the transmitter or the signal processing equipment has drifted outside its limit. If the channels are within the criteria, it is an indication that the channels are OPERABLE.

The Frequency, about once every shift, is based on operating experience that demonstrates channel failure is rare. Since the probability of two random failures in redundant channels in any 12 hour period is extremely low, the CHANNEL CHECK minimizes the chance of loss of protective function due to

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.10.1 (continued)

failure of redundant channels. The CHANNEL CHECK supplements less formal, but more frequent, checks of channel OPERABILITY during normal operational use of the displays associated with the LCO's required channels.

When operating in Required Action A.1, CHANNEL CHECK is still required. However, in this condition, a redundant intermediate range is not available for comparison. CHANNEL CHECK may still be performed via comparison with power or source range detectors, if available, and verification that the OPERABLE intermediate range channel is energized and indicates a value consistent with current unit status.

SR 3.3.10.2

For intermediate range neutron flux channels, CHANNEL CALIBRATION is a complete check and readjustment of the channels, from the preamplifier input to the indicators. This test verifies the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift to ensure that the instrument channel remains operational between successive tests.

The SR is modified by a Note excluding neutron detectors from CHANNEL CALIBRATION. It is not necessary to test the detectors because generating a meaningful test signal is difficult. In addition, the detectors are of simple construction, and any failures in the detectors will be apparent as a change in channel output. The Frequency is based on operating experience and consistency with the typical industry refueling cycle and is justified by demonstrated instrument reliability over an [18] month interval such that the instrument is not adversely affected by drift.

SR 3.3.10.3

SR 3.3.10.3 is the verification within 7 days prior to reactor startup of one decade of overlap with the power range neutron flux instrumentation prior to intermediate range indication exceeding 1E-6 amp. This ensures a

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.10.3 (continued)

continuous source of power indication during the approach to criticality. Failure to perform this Surveillance leaves the unit in a condition where the intermediate range channels provide adequate protection until the verification can be made.

The test may be omitted if performed within the previous 7 days based on operating experience, which shows that intermediate range instrument overlap does not change appreciably within this test interval.

REFERENCES

None.

B 3.3 INSTRUMENTATION

B 3.3.11 Emergency Feedwater Initiation and Control (EFIC) Instrumentation

BASES

BACKGROUND

The EFIC System instrumentation is designed to provide safety grade means of controlling the secondary system as a heat sink for core decay heat removal. To ensure the secondary system remains a heat sink, the EFIC System takes action to initiate emergency feedwater (EFW) when the primary source of feedwater is lost and to isolate functional components from hydraulic faults within the secondary system. These actions ensure that a source of cooling water is available to be fed to a once through steam generator (OTSG) that has a controlled steam pressure, thereby fixing the heat sink temperature at the saturation temperature of the secondary system. The EFIC Functions that are supported and the parameters that are needed for each of these Functions are described next.

The EFIC instrumentation contains devices and circuitry that generate the following signals when monitored variables reach levels that are indicative of conditions requiring protective actions.

a. EFW Initiation;

b. EFW Vector Valve Control;

c. Main Steam Line Isolation; and

d. Main Feedwater (MFW) Isolation.

EFW is initiated to restore a source of cooling water to the secondary system when conditions indicate that the normal source of feedwater is insufficient to continue heat removal. The two indications used for this are the loss of both MFW pumps and a low level in the steam generator (SG). Also, EFW is initiated when action is being taken to isolate the MFW from the SG during conditions of uncontrolled depressurizations. This is done by initiating EFW when steam pressure reaches the low SG pressure setpoint for isolation of main steam and MFW, and EFW vector valve control. Finally, EFW is initiated when the primary system experiences a total loss of forced circulation. This initiation, on the loss of all reactor coolant pumps (RCPs),

(continued)

BASES

BACKGROUND (continued)

ensures the EFW is available to raise SG levels to promote natural circulation cooling. Additionally, this ensures that EFW is available under the worst-case, small break loss of coolant accident (LOCA) conditions when secondary system cooling with high SG water levels is necessary.

The EFIC System also isolates main steam and MFW to an SG that has lost pressure control. With the loss of pressure control, the heat sink temperature control is lost and the heat removal rate cannot be controlled. The main steam and MFW are isolated to an SG when the steam pressure reaches a low setpoint, a condition which is beyond the normal operating point of the secondary system.

The EFIC System also performs an EFW control function to avoid delivering EFW to a depressurized SG when the other SG remains pressurized. This continues the function of isolating functional components from an SG whose pressure cannot be controlled. This function precludes the delivery of fluid to a depressurized SG, thereby avoiding an uncontrolled cooling condition as long as the other SG remains pressurized. When both of the SGs are depressurized, the EFIC logic provides EFW flow to both SGs until a significant pressure difference between the two SGs is developed, thereby ensuring that core cooling is maintained.

Trip Setpoints and Allowable Values

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

The trip setpoints used in the bistables are based on the analytical limits stated in FSAR, Section [14.1] (Ref. 1). The selection of these trip setpoints is such that adequate protection is provided when all sensor and processing time delays are taken into account. The Allowable Values specified in Table 3.3.11-1 in the accompanying LCO are conservatively adjusted with respect to the analytical limits to allow for calibration tolerances, instrumentation uncertainties, instrument drift, and severe environmental errors for those EFIC channels that must function in harsh

(continued)

BASES

BACKGROUND

Trip Setpoints and Allowable Values (continued)

environments as defined by 10 CFR 50.49 (Ref. 2). A detailed description of the methodology used to calculate the trip setpoints, including their explicit uncertainties, is provided in "[Unit Specific Setpoint Methodology]" (Ref. 3). The actual nominal trip setpoint entered into the bistable is more conservative than that specified by the Allowable Value to account for changes in random measurement errors detectable by a CHANNEL FUNCTIONAL TEST. One example of such a change in measurement error is drift during the surveillance interval. A channel is inoperable if its actuation trip setpoint is not within its required Allowable Value.

Setpoints in accordance with the Allowable Value ensure that the consequences of Design Basis Accidents (DBAs) are acceptable, providing the unit is operated from within the LCOs at the onset of the DBA, and that the equipment functions as designed.

Each channel can be tested on line to verify that the setpoint accuracy is within the specified allowance requirements of Figure [ ], FSAR, Chapter [7] (Ref. 4). Once a designated channel is taken out of service for testing, a simulated signal is injected in place of the field instrument signal. The process equipment for the channel in test is then tested, verified, and calibrated. The SRs for the channels are specified in the SRs Section.

The Allowable Values listed in Table 3.3.11-1 are based on the "[Unit Specific Setpoint Methodology]" (Ref. 3), which incorporates all of the known uncertainties applicable for each channel. The magnitudes of these uncertainties are factored into the determination of each trip setpoint. All field sensors and signal processing equipment for these channels are assumed to operate within the allowances of these uncertainty magnitudes.

Figure [ ], FSAR, Chapter [7] (Ref. 4), illustrates EFIC EFW Initiation logic operation.

Each EFIC train actuates on a one-out-of-two taken twice combination of trip signals from the instrumentation channels. Each EFIC channel can issue an initiate command, but an EFIC actuation will take place only if at least two

(continued)

BASES

BACKGROUND

Trip Setpoints and Allowable Values (continued)

channels issue initiate commands. The one-out-of-two taken twice logic combinations are transposed between trains so that failure of two channels prevents actuation of, at most, one train.

More detailed descriptions of the EFIC instrumentation are provided next.

1. EFW Initiation

Figure [ ], FSAR, Chapter [7] (Ref. 4), illustrates one channel of the EFIC EFW Initiation channel. The individual instrumentation channels that serve EFIC EFW Initiation Function are discussed next.

a. Loss of MFW Pumps (Control Oil Pressure)

Loss of both MFW Pumps is one of the four parameters within the EFIC System that automatically initiates EFW. Loss of MFW Pumps is detected by MFW Pump turbine control oil pressure. The MFW Pump status instrumentation is a part of the nuclear instrument (NI) and Reactor Protection System (RPS). Each RPS channel receives MFW Pump status information from pressure switches (four per pump). If both switches in a single channel trip, the associated RPS channel trips. Each RPS channel provides both MFW Pumps tripped signal to the associated EFIC channel. The trip Function is bypassed when THERMAL POWER ≤ 20% RTP and the RPS is in shutdown bypass. The bypass is automatically removed when THERMAL POWER is greater than 20% RTP.

Loss of both MFW Pumps was chosen as an EFW automatic initiating parameter because it is a direct and immediate indicator of loss of MFW.

b. SG Level—Low

Four EFIC dedicated low range level transmitters per SG Level—Low are used to generate the signals used for detection for low level

(continued)

BASES

BACKGROUND

b. SG Level—Low (continued)

conditions for EFW actuation. There is one transmitter for each of the four channels A, B, C, and D. The signals are also used after EFW is actuated to control SG level at the low level setpoint [30 inches] when one or more RCPs are operational.

The lower and upper taps for the low range level transmitters are located at 6 inches and 277 inches, respectively, above the upper face of the SG's lower tube sheet. The calibrated range is 0-150 inches.

SG Level—Low was chosen as an EFW automatic initiating parameter because it indicates that the primary feedwater source is insufficient to meet the heat removal requirements and, therefore, additional cooling water is necessary to ensure core decay heat removal.

c. SG Pressure—Low

Four transmitters per SG provide the EFIC System with channels A through D of SG Pressure—Low. These are the same transmitters used by the MFW and Main Steam Line Isolation Functions. When the SG pressure drops below the bistable setpoint of 600 psig on a given channel, an EFW Initiation signal is sent to the automatic actuation logic. The low pressure Function may be manually bypassed when both SGs are less than 750 psig. If either SG input channel exceeds 750 psig, the EFIC channel bypass is automatically removed. The low pressure operational bypass allows for normal cooldown without EFIC actuation.

SG Pressure—Low is a primary indication and actuation signal for steam line breaks (SLBs) or feedwater line breaks (FWLBs). For small breaks, which do not depressurize the SG or take a long time to depressurize, automatic actuation is not required. The operator has time to diagnose the problem and take the appropriate actions.

(continued)

BASES

BACKGROUND (continued)

d. RCP Status

A loss of power to all four RCPs is an indication of a pending loss of forced flow in the Reactor Coolant System. These sensing signals are input into the four channels of EFIC.

When at least two channels issue initiate commands based on loss of all RCPs, the EFIC System will automatically actuate EFW and switch the level control setpoint to approximately 50% in the SG. This higher setpoint provides a thermal center in the SG at a higher elevation than that of the reactor to ensure natural circulation of the reactor coolant.

To allow heatup and cooldown operations without actuation, a bypass permissive of 10% RTP is used. The 10% bypass permissive was chosen because it was an available, qualified Class 1E signal at the time the EFIC System was designed. When the first RCP is started, the "loss of four RCPs" initiation signal may be manually reset. If the bypass is not manually reset, it will be automatically reset when the unit reaches 10% power. During cooldown, the bypass may be inserted at any time the power has been reduced below 10%. However, for most operating conditions, it is recommended that this trip function remain active until after the Decay Heat Removal System has been initiated and the system is ready for the last RCP to be tripped. This trip function must be bypassed prior to stopping the last RCP.

2. EFW Vector Valve Control

Figure [ ], FSAR, Chapter [7] (Ref. 4), illustrates one channel of the EFIC EFW Vector Valve Control logic. The function of the EFW vector logic is to determine whether EFW should not be fed to one or the other SG. This is to preclude the continued addition of EFW to a depressurized SG and, thus, to minimize the overcooling effects of a steam leak.

(continued)

BASES

BACKGROUND

2. EFW Vector Valve Control (continued)

Each set of vector logic receives SG pressure information from bistables located in the input logic of the same EFIC channel. The pressure information received is:

a. SG A pressure less than 600 psig;

b. SG B pressure less than 600 psig;

c. SG A pressure 125 psid greater than SG B pressure; and

d. SG B pressure 125 psid greater than SG A pressure.

Each vector logic also receives a vector/control enable signal from both EFIC channel A and channel B when EFW is initiated. [Each logic also receives an SG high level signal. High level in an SG prevents opening the associated vector valves and enables closing the valves without either EFIC train vector valve enable.]

The vector logic develops signals to open or to close SG A and B EFW valves.

The vector logic outputs are in a neutral state until enabled by the control/vector enable from the channel A or B trip logics. When enabled, the vector logic can issue open or close commands to the EFW control valves and EFW isolation valves per the selected channel assignments.

Each vector logic may isolate EFW to one SG or the other, never both.

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BASES

BACKGROUND

2. EFW Vector Valve Control (continued)

The valve open or close commands are determined by the relative values of SG pressures as follows:

PRESSURE STATUS	SG VALVES "A"	SG VALVES "B"
SG A and SG B > 600 psig	Open	Open
¦ SG A - SG B ¦ < 125 psid	Open	Open
SG A or SG B ≤ 600 psig and SG A - SG B ≥ 125 psid	Open	Close
SG A or SG B ≤ 600 psig and SG B - SG A ≥ 125 psid	Close	Open

Bypass

One of the four initiation channels can be put into "maintenance bypass." Bypassing one initiation channel isolates that channel's signal to the functions fed from initiation channel but does not bypass the trip logic within the actuation channel. An interlock feature prevents bypassing more than one channel at a time. In addition, since the EFIC System receives signals from NI and RPS, the maintenance bypass from the NI and RPS is interlocked with the EFIC System. If one channel of the NIS and RPS is in maintenance bypass, only the corresponding channel of the EFIC may be bypassed (e.g., channel A, NI or RPS, and channel A, EFIC). This ensures that only the corresponding channels of the EFIC and NI and RPS are placed in maintenance bypass at the same time.

EFIC channel maintenance bypass does not bypass EFW Initiation from Engineered Safety Feature Actuation System (ESFAS) high pressure injection (HPI). The EFIC HPI Actuation Function is, however, bypassed when ESFAS is bypassed.

The operational bypass provisions were discussed as part of the individual Functions described earlier.

(continued)

BASES

BACKGROUND

Bypass (continued)

Operational bypass of the OTSG Level—High input to the vector valve logic is possible after EFIC initiation. [For this unit, bypassing the overfill function is for the following reasons:]

3, 4. Main Steam Line and MFW Isolation

Figure [ ], FSAR, Chapter [7] (Ref. 4) illustrates one channel of the EFIC Main Steam Line and MFW Isolation logic. Four pressure transmitters per SG provide EFIC with channels A through D logic of SG pressure. The channels are as described for EFW Initiation mentioned earlier.

Once isolated, manual action is required to defeat the isolation command if desired. The EFIC System is designed to perform its intended function with one channel in maintenance bypass (in effect, inoperable) with a single failure in one of the remaining channels. This is in compliance with IEEE-279-1971 (Ref. 5) due to the redundancy and independence in the EFIC design.

APPLICABLE SAFETY ANALYSES

1. EFW Initiation

Although loss of both MFW pumps is a direct and immediate indicator of loss of MFW, other scenarios such as valve closures could potentially cause loss of feedwater. The loss of MFW analysis, therefore, conservatively assures the actuation of EFW on low SG level. If the loss of feedwater is due to loss of MFW pumps, EFW will be actuated much earlier than assumed in the analysis, which will increase the SG heat transfer capability and will lessen the severity of the transient.

The DBA which forms the basis for initiation of the EFW systems is a loss of MFW transient. In the analysis of this transient, SG Level—Low is the parameter assumed to automatically initiate EFW. This assumption yields the least SG inventory available for heat removal and is, therefore, conservative for

(continued)

BASES

APPLICABLE SAFETY ANALYSES

1. EFW Initiation (continued)

evaluation of this DBA. SG Level—Low would be an indicator of all accidents involving a loss of primary to secondary heat removal.

SG Pressure—Low is a primary indication and provides the actuation signal for SLBs or FWLBs. For small breaks, which do not depressurize the SG or take a long time to depressurize, automatic actuation is not required. The operator has sufficient time to diagnose the problem and take the appropriate actions.

Loss of four RCPs is a primary indicator of the need for auxiliary feedwater (AFW) in the safety analyses for loss of electric power and loss of coolant flow. It also serves as a backup indicator for SLBs and small break LOCAs.

2. EFW Vector Valve Control

Most of the FSAR SLB analyses were performed prior to the development of the safety grade EFIC System. Therefore, the EFIC vector valve control was not credited in the original licensing basis for a main SLB analysis. Instead, operator action was credited with isolating AFW to the affected SG within the first 60 seconds. However, isolating the affected SG is a function automatically performed by the EFIC System. Therefore, the FSAR analysis remains conservative relative to the inclusion of the vector valve control.

3, 4. Main Steam Line and MFW Isolation

The FSAR analysis assumed integrated control system action for MFW and Main Steam Line Isolation. The analysis took credit for turbine stop valve closure and feedwater valve isolation on reactor trip and considered the isolation functions occurring on SG pressure < 600 psig as backup. These isolation functions are currently provided by the safety grade EFIC System. Use of the EFIC System in the original safety analysis would have been consistent with the licensing position allowing mitigative functions to be

(continued)

BASES

APPLICABLE SAFETY ANALYSES

3, 4. Main Steam Line and MFW Isolation (continued)

performed by safety grade systems in accident analysis. For these reasons, the SLB accident analysis remains conservative with the assumed integrated control system actions.

The EFIC System satisfies Criterion 3 of the NRC Policy Statement.

LCO

All instrumentation performing an EFIC System Function in Table B 3.3.11-1 shall be OPERABLE. Failure of any instrument renders the affected channel(s) inoperable and reduces the reliability of the affected Functions.

Four channels are required OPERABLE for all EFIC instrumentation channels to ensure that no single failure prevents actuation of a train. Each EFIC instrumentation channel is considered to include the sensors and measurement channels for each Function, the operational bypass switches, and permissives. Failures that disable the capability to place a channel in operational bypass, but which do not disable the trip Function, do not render the protection channel inoperable.

Only the Allowable Values are specified for each EFIC initiation and bypass removal function in the LCO. In Table 3.3.11-1, Allowable Values for the bypass removal functions are specified in terms of applicability limits on the associated trip Function. Nominal trip setpoints are specified in the unit specific setpoint calculations. The nominal setpoints are selected to ensure the setpoints measured by CHANNEL FUNCTIONAL TESTS do not exceed the Allowable Value if the bistable is performing as required. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable provided that operation and testing are consistent with the assumptions of the unit specific setpoint calculations. Each Allowable Value specified is more conservative than the analytical limit assumed in the safety analysis to account for instrument uncertainties appropriate to the trip Function. These uncertainties are defined in the "[Unit Specific Setpoint Methodology]" (Ref. 3).

(continued)

BASES

LCO (continued)

The Bases for the LCO requirements of each specific EFIC Function are discussed next.

Loss of MFW Pumps

Four EFIC channels shall be OPERABLE with MFW pump turbines A and B control oil low pressure actuation setpoints of > [55] psig. The 55 psig setpoint is about half of the normal operating control oil pressure. The 55 psig setpoint Allowable Value was arbitrarily chosen as a good indication of Loss of MFW Pumps. Analysis only assumes Loss of MFW Pumps and a specific value of MFW pump control oil pressure is not used in the analysis. The Loss of MFW Pumps Function includes a bypass enable and removal function from the NI/RPS. The bypass removal function is based on maintaining consistency with RPS LCO and design of system.

SG Level—Low

Four EFIC dedicated low range level transmitters per SG shall be OPERABLE with SG Level—Low actuation setpoints of ≥ [9] inches, to generate the signals used for detection for low level conditions for EFW Initiation. There is one transmitter for each of the four channels A, B, C, and D. The signals are also used after EFW is actuated to control at the low level setpoint of 30 inches when one or more RCPs are in operation. In the determination of the low level setpoint, it is desired to place the setpoint as low as possible, considering instrument errors, to give the maximum operability margin between the integrated control system low load control setpoint and the EFW Initiation setpoint. This will minimize spurious or unwanted initiation of EFW. Credit is only taken for low level actuation for those transients which do not involve a degraded environment. Therefore, normal environment errors only are used for determining the SG Level—Low level setpoint.

SG Pressure—Low

Four EFIC channels per SG shall be OPERABLE with SG low pressure actuation setpoints of ≥ [600] psig. The setpoint is chosen to avoid actuation under transient conditions not requiring secondary system isolation, preferring to maintain

(continued)

BASES

LCO

SG Pressure—Low (continued)

a steaming path to the condenser, if possible. Small break LOCA analyses have indicated minimum secondary system pressures of approximately 700 psig. The SG Pressure—Low Function includes a bypass enable and removal function. The bypass removal Allowable Value is chosen to allow sufficient operating margin for the operator to bypass when cooling down.

SG Differential Pressure—High

Four EFIC channels for SG differential pressure shall be OPERABLE with setpoints of ≤ [125] psid. The setpoint ensures that automatic EFW isolation to a depressurized SG occurs for the range of sizes of SLBs that require rapid actuation early in the event. The setpoint has also been chosen to avoid spurious isolation of EFW during conditions due to relatively small deviations in SG pressures that can be caused by primary system conditions. The SG Differential Pressure—High Function includes a bypass enable and removal function. The bypass removal Allowable Value is chosen to allow sufficient operating margin for the operator to bypass when cooling down.

RCP Status

Four EFIC channels for RCP status shall be OPERABLE. This ensures that upon the loss of four RCPs, EFW will be automatically initiated with the EFW control level automatically raised to approximately 50%, providing a higher SG level for establishing and maintaining natural circulation conditions when the forced reactor coolant flow is lost. No setpoint is specified since the status indication as used by EFIC is binary in nature. The RCP Status Function includes a bypass enable and removal function from the RPS. The Allowable Value for the bypass removal is set high enough to avoid spurious actuations during low power operation.

(continued)

BASES

LCO (continued)

SG Level—High

[For this unit, the basis for SG Level—High signal is as follows:]

APPLICABILITY

The EFIC System instrumentation Functions shall be OPERABLE in accordance with Table 3.3.11-1. Each Function has its own requirements that are based on the specific accidents and conditions that it is designed to protect against.

The initiation of EFW on the Loss of MFW Pumps shall only be required in MODE 1 and in MODES 2 and 3 when not in shutdown bypass, when core power production and heat removal requirements are the greatest. Below these unit conditions, the EFW Initiation on low SG level is rapid enough to avoid unnecessary primary system overheating.

EFW Initiation on low SG level shall be OPERABLE at all times the SG is required for heat removal. These conditions include MODES 1, 2, and 3. To avoid automatic actuation of the EFW pumps during normal heatup and cooldown transients, the low SG pressure Function can be bypassed at or below a secondary pressure of [750] psig. This secondary pressure can normally only be reached during MODE 3 operation.

The EFW System Initiation on loss of all RCPs Function shall be operable at ≥ 10% RTP. It is possible to bypass the Function below 10% RTP; however, for most cases, the Function is kept in service until the unit is placed on the Decay Heat Removal System. To prevent inadvertent actuation of the EFW pumps, it must be bypassed prior to stopping the last RCP.

The MFW, Main Steam Line Isolation, and EFW Vector Valve Control Functions shall be OPERABLE in MODES 1, 2, and 3 with SG pressure ≥ 750 psig because the SG inventory can be at a high energy level and contribute significantly to the peak pressure with a secondary side break. Both the normal feedwater and the EFW must be able to be isolated on each SG to limit overcooling of the primary and mass and energy releases to the reactor building. Once the SG pressures have decreased below 750 psig, the Main Steam Line and MFW Isolation Functions can be bypassed to avoid actuation during normal unit cooldowns. The EFW Vector Valve Control

(continued)

BASES

APPLICABILITY (continued)

logic will not perform any function when both SG pressures are low; thus, the logic can also be bypassed at the same point. In MODES 4, 5, and 6, the energy level is low and the secondary side feedwater flow rate is low or nonexistent. In MODES 4, 5, and 6, the primary system temperatures are too low to allow the SGs to effectively remove energy and EFIC instrumentation is not required to be OPERABLE.

ACTIONS

If a channel's trip setpoint is found nonconservative with respect to the Allowable Value, or any of the transmitter, signal processing electronics, or EFIC channel cabinet modules are found inoperable, then all affected Functions provided by that channel must be declared inoperable and the unit must enter the Conditions for the particular protection Function affected.

A Note has been added to the ACTIONS indicating that a separate Condition entry is allowed for each Function.

A.1 and A.2

Condition A applies to failures of a single EFW Initiation, Main Steam Line Isolation, or MFW Isolation instrumentation channel. This includes failure of a common instrumentation channel in any combination of the Functions.

With one channel inoperable in one or more EFW Initiation, Main Steam Line Isolation, or MFW Isolation Functions listed in Table 3.3.11-1, the channel(s) must be placed in bypass or trip within 1 hour. This Condition applies to failures that occur in a single channel, e.g., channel A, which when bypassed will remove initiate Functions within the channel from service. Since the RPS and EFIC channels are interlocked, only the corresponding channel in each system may be bypassed at any time. This feature is ensured by an electrical interlock. If testing of another channel in either the EFIC or RPS is required, the EFIC channel must be placed in trip to allow the other channel to be bypassed. With the channel in trip, the resultant logic is one-out-of-two. The Completion Time of 1 hour is adequate to perform Required Action A.1.

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BASES

ACTIONS

A.1 and A.2 (continued)

Required Action A.2 provides for placing the channel(s) in trip if the channel(s) is/are not restored to OPERABLE status within 72 hours.

A single inoperable EFIC instrumentation channel affects at most one train of EFW, Main Steam Line Isolation, and MFW Isolation. Therefore, the 72 hour Completion Time was selected to be consistent with the allowed out of service time for the EFW, Main Steam Line Isolation, and MFW Isolation Functions.

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

Condition B applies to a situation where two instrumentation channels for multiple protection functions of EFW Initiation, Main Steam Line Isolation, or MFW Isolation instrumentation are inoperable. For example, Condition B applies if channel A and B of the EFW Initiation Function are inoperable.

Condition B does not apply if one channel of different Functions is inoperable in the same protection channel. That condition is addressed by Condition A.

With two EFW Initiation, Main Steam Line Isolation, or MFW Isolation protection channels inoperable, one channel must be placed in bypass (Required Action B.1). Bypassing one of the remaining OPERABLE channels is not possible due to system interlocks. Therefore, the second channel must be tripped (Required Action B.2) to prevent a single failure from causing loss of the EFIC Function. The Completion Times of 1 hour are adequate to perform the Required Actions.

One of the channels must be returned to OPERABLE status (Required Action B.3) to minimize the time the system is permitted to operate in a configuration that is not capable of withstanding a single failure and still initiate EFW, Main Steam Line Isolation, and MFW Isolation. Restoring one channel changes system status to that of Condition A.
A single inoperable EFIC channel affects at most one train of EFW, Main Steam Line Isolation, and MFW Isolation. Therefore the 72 hour Completion Time was selected to be

(continued)

BASES

ACTIONS

B.1, B.2, and B.3 (continued)

consistent with the allowed out of service time for the EFW, Main Steam Line Isolation, and MFW Isolation Functions.

C.1

The function of the EFW Vector Valve Control is to meet the single-failure criterion while being able to provide EFW on demand and isolate an SG when required. These conflicting requirements result in the necessity for two valves in series, in parallel with two valves in series, and a four channel valve command system. Refer to LCO 3.3.14, "Emergency Feedwat r Initiation and Control (EFIC) Emergency Feedwater (EFW)—Vector Valve Logic."

With one EFW Vector Valve Control channel inoperable, the system cannot meet the single-failure criterion and still meet the dual functional criteria described earlier. This condition is analogous to having one EFW train inoperable. Therefore, when one vector valve control channel is inoperable, the channel must be restored to OPERABLE status (Required Action C.1) within 72 hours, which is consistent with the Completion Time associated with the loss of one train of EFW.

D.1, D.2.1, D.2.2, E.1, and F.1

If the Required Actions cannot be met within the required Completion Time, the unit must be placed in a MODE or condition in which the requirement does not apply. This is done by placing the unit in a nonapplicable MODE for the particular Function. The nonapplicable MODE is to open the CRD trip breakers for Function 1.a, MODE 4 for Function 1.b, less than 10% RTP for Function 1.d, and SG pressure less than 750 psig for all other Functions. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

A Note indicates that the SRs for each EFIC instrumentation Function are identified in the SRs column of Table 3.3.11-1. All Functions are subject to CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION. The SG—Low Level Function is the only Function that was modeled in transient analysis, and thus is the only EFW Initiation Function subjected to response time testing. Response time testing is also required for Main Steam Line and MFW Isolation. Individual EFIC subgroup relays must also be tested, one at a time, to verify the individual EFIC components will actuate when required. Some components cannot be tested at power since their actuation might lead to unit trip or equipment damage. These are specifically identified and must be tested when shut down. The various SRs account for individual functional differences and for test frequencies applicable specifically to the Functions listed in Table 3.3.11-1. The operational bypasses associated with each EFIC instrumentation channel are also subject to these SRs to ensure OPERABILITY of the EFIC instrumentation channel.

SR 3.3.11.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; therefore, it is key in verifying that the instrumentation continues to operate properly between each CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION.

Agreement criteria are determined by the unit staff, based on a combination of the channel instrument uncertainties, including isolation, indication, and readability. If a channel is outside the criteria, it may be an indication that the transmitter or the signal processing equipment has drifted outside its limit. If the channels are within the

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.11.1 (continued)

criteria, it is an indication that the channels are OPERABLE. If the channels are normally off scale during times when surveillance is required, the CHANNEL CHECK will only verify that they are off scale in the same direction. Off scale low current loop channels are verified to be reading at the bottom of the range and not failed downscale.

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

SR 3.3.11.2

A CHANNEL FUNCTIONAL TEST verifies the function of the required trip, interlock, and alarm functions of the channel. Setpoints for both trip and bypass removal functions must be found within the Allowable Value specified in the LCO. (Note that the Allowable Values for the bypass removal functions are specified in the Applicable MODES or Other Specified Condition column of Table 3.3.11-1 as limits on applicability for the trip Functions.) Any setpoint adjustment shall be consistent with the assumptions of the current unit specific setpoint analysis.

The Frequency of 31 days is based on unit operating experience with regard to channel OPERABILITY and drift, which demonstrates that failure of more than one channel of a given function in any 31 day interval is a rare event.

SR 3.3.11.3

CHANNEL CALIBRATION is a complete check of the instrument channel including the sensor. The test verifies the channel responds to a measured parameter within the necessary range

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.11.3 (continued)

and accuracy. CHANNEL CALIBRATION leaves the channels adjusted to account for instrument drift to ensure that the instrument channel remains operational between successive tests. CHANNEL CALIBRATION shall find that measurement errors and bistable setpoint errors are within the assumptions of the unit specific setpoint analysis. CHANNEL CALIBRATIONS must be performed consistent with the assumptions of the unit specific setpoint analysis.

The Frequency is based on the assumption of an [18] month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

SR 3.3.11.4

This SR verifies individual channel actuation response times are less than or equal to the maximum value assumed in the accident analysis.

Response time testing acceptance criteria are included in "Unit Specific Response Time Acceptance Criteria" (Ref. 6).

Individual component response times are not modeled in the analysis. The analysis models the overall or total elapsed time, from the point at which the parameter exceeds the actuation setpoint value at the sensor, to the point at which the end device is actuated.

EFIC RESPONSE TIME tests are conducted on an [18] month STAGGERED TEST BASIS. Testing of the final actuation devices, which make up the bulk of the EFIC RESPONSE TIME, is included in the testing of each channel. Therefore, staggered testing results in response time verification of these devices every [18] months. The [18] month test Frequency is based on unit operating experience, which shows that random failures of instrumentation components causing serious response time degradation, but not channel failure, are infrequent occurrences. EFIC RESPONSE TIMES cannot be determined at power since equipment operation is required.

(continued)

BASES (continued)

REFERENCES

1. FSAR, Section [14.1].
2. 10 CFR 50.49.
3. [Unit Name], "[Unit Specific Setpoint Methodology]."
4. FSAR, Chapter [7].
5. IEEE-279-1971, April 1972.
6. "Unit Specific Response Time Acceptance Criteria."

B 3.3 INSTRUMENTATION

B 3.3.12 Emergency Feedwater Initiation and Control (EFIC) Manual Initiation

BASES

BACKGROUND	The EFIC manual initiation capability provides the operator with the capability to actuate EFIC Functions from the control room in the absence of any other initiation condition. Manually actuated Functions include main feedwater (MFW) Isolation for once through steam generator (SG) A, MFW Isolation for SG B, Main Steam Line Isolation for SG A, Main Steam Line Isolation for SG B, and Emergency Feedwater (EFW) Actuation. These Functions are provided in the event the operator determines that an EFIC Function is needed and does not automatically actuate. These are backup Functions to those performed automatically by EFIC. The EFIC manual initiation circuitry satisfies the manual initiation and single-failure criterion requirements of IEEE-279-1971 (Ref. 1).
APPLICABLE SAFETY ANALYSES	EFIC Functions credited in the safety analysis are automatic. However, the manual initiation Functions are required by design as backups to the automatic trip Functions and allow operators to actuate EFW, Main Steam Line Isolation, or MFW Isolation whenever these Functions are needed. Furthermore, the manual initiation of EFW Actuation, Main Steam Line Isolation, and MFW Isolation may be specified in unit operating procedures. The EFIC manual initiation functions satisfy Criterion 3 of the NRC Policy Statement.
LCO	All instrumentation performing an EFIC manual initiation Function shall be OPERABLE. Failure of any instrument renders the affected channel(s) inoperable and reduces the reliability of the affected Functions. Two manual initiation switches per actuation channel (A and B) of each Function (A and B MFW Isolation, A and B Main Steam Line Isolation, and EFW Actuation) are required to be OPERABLE whenever the SGs are being relied on to remove

(continued)

BASES

LCO (continued)

heat. Each Function (MFW Isolation, Main Steam Line Isolation, and EFW Initiation) has two actuation or "trip" channels, channels A and B. Within each channel A actuation logic there are two manual trip switches. When one manual switch is depressed, a half trip occurs. When both manual switches are depressed, a full trip of channel A actuation occurs for that particular Function. Similarly, channel B actuation logic for each Function has two manual trip switches. Both switches per actuation channel must be OPERABLE and must be depressed to get a full manual trip of that channel. The use of two manual trip switches for each channel of actuation logic allows for testing without actuating the end devices and also reduces the possibility of accidental manual actuation.

APPLICABILITY

The MFW and Main Steam Line Isolation manual initiation Functions shall be OPERABLE in MODES 1, 2, and 3 because SG inventory can be at a sufficiently high energy level to contribute significantly to the peak containment pressure during a secondary side break. In MODES 4, 5, and 6, the SG energy level is low and secondary side feedwater flow rate is low or nonexistent.

The EFW manual initiation Function shall be OPERABLE in MODES 1, 2, and 3 because the SGs are relied on for Reactor Coolant System heat removal. In MODES 4, 5, and 6, heat removal requirements are reduced and can be provided by the Decay Heat Removal System.

ACTIONS

A Note has been added to the ACTIONS indicating that separate Condition entry is allowed for each EFIC manual initiation Function.

A.1

With one or both manual initiation switches of one or more EFIC Function(s) inoperable in one channel, the channel for the associated EFIC Function(s) must be placed in the tripped condition within 72 hours. With the channel in the tripped condition, the single-failure criterion is met and the operator can still initiate one actuation channel given

(continued)

BASES

ACTIONS

A.1 (continued)

a single failure in the other channel. Failure to perform Required Action A.1 could allow a single failure of another switch to prevent manual actuation of at least one of two trip channels. The Completion Time allotted to trip the channel allows the operator to take all the appropriate actions for the failed channel and still ensure that the risk involved in operating with the failed channel is acceptable.

B.1

With one or both manual initiation switches of one or more EFIC Function(s) inoperable in both actuation channels, one actuation channel for each Function must be restored to OPERABLE status within 1 hour. With the channel restored, the second channel must be placed in the tripped condition within 72 hours (Required Action A.1). With the channel in the tripped condition, the single-failure criterion is met and the operator can still initiate one actuation channel given a single failure in the other channel. The Completion Time allotted to restore the channel allows the operator to take all the appropriate actions for the failed channel and still ensures that the risk involved in operating with the failed channel is acceptable.

C.1 and C.2

If Required Action A.1 or Required Action B.1 cannot be met within the required Completion Time, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required MODES from full power conditions in an orderly manner and without challenging unit systems.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.3.12.1

This SR requires the performance of a CHANNEL FUNCTIONAL TEST to ensure that the channels can perform their intended functions. However, for MFW and Main Steam Line Isolation, the test need not include actuation of the end device. This is due to the risk of a unit transient caused by the closure of valves associated with MFW and Main Steam Line Isolation or actuating EFW during testing at power. The Frequency of 31 days is based on operating experience that demonstrates the rarity of more than one channel failing within the same 31 day interval.

REFERENCES

1. IEEE-279-1971, April 1972.

B 3.3 INSTRUMENTATION

B 3.3.13 Emergency Feedwater Initiation and Control (EFIC) Logic

BASES

BACKGROUND

Main Steam Line and Main Feedwater (MFW) Isolation

The four emergency feedwater initiation and control (EFIC) channels sensing a steam generator (SG) low outlet pressure condition input their initiate commands to the trip logic modules. Figure [ ], FSAR, Chapter [7] (Ref. 1), illustrates the Main Steam Line and MFW Isolation Logics. The trip logic modules are physically located in the "A" and "B" EFIC channel cabinets. Channel "A" actuation logic initiates when instrumentation channel "A" or "B" initiates and channel "C" or "D" initiates, which in simplified logic is:

"A" actuation = (A and C) or (A and D) or (B and C) or (B and D)

Channel "B" actuation logic initiates when instrumentation channel "A" or "C" initiates and channel "B" or "D" initiates, which in simplified logic is:

"B" actuation = (A and B) or (A and D) or (C and B) or (C and D)

Each of the four Functions (SG A Main Feedwater Isolation, SG B Main Feedwater Isolation, SG A Main Steam Line Isolation, and SG B Main Steam Line Isolation) has a channel "A" and a channel "B" of automatic actuation logic.

Both channels "A" and "B" of the SG A Main Feedwater Isolation automatic actuation logic send closure signals to the SG A main feedwater pump suction valve, the three SG A block valves, and the MFW pump discharge cross connect valve. In addition, the instrumentation trips MFW pump "A."

Both channels "A" and "B" of the SG A Main Steam Line Isolation automatic actuation logic send closure signals to both of the SG A Main Steam Isolation valves.

SG B MFW and Main Steam Line Isolation automatic actuation logics respond similarly for the SG B valves and MFW pump "B."

(continued)

BASES

BACKGROUND (continued)

Emergency Feedwater (EFW) Actuation

The four EFIC instrumentation channels for each of the parameters being sensed input their initiate commands to the trip logic modules. Figure [ ], FSAR, Chapter [7] (Ref. 1), illustrates the EFW initiation logic. These trip logic modules are physically located in the "A" and "B" EFIC channel cabinets.

EFW Actuation functions are the same logic combinations as MFW and Main Steam Line Isolation. EFW initiation also occurs on high pressure injection (HPI) initiation. Both trains of HPI initiation are input into each EFW initiate logic channel.

EFIC automatically initiates the EFW System when any of the following conditions exist:

a. All four reactor coolant pumps are tripped;

b. Both MFW pumps are tripped and reactor power is > 20% RTP with the nuclear instrumentation Reactor Protection System not in shutdown bypass;

c. Low level in either once through SG;

d. Low pressure in either SG; or

e. HPI Actuation on both A and B Engineered Safety Feature Actuation System channels.

Vector Valve Enable Logic

The EFW module logic is responsible for sending open or close signals to the EFW control and isolation valves. Figure [ ], FSAR, Chapter [7] (Ref. 1), illustrates the vector valve logic. The vector module logic outputs are in a neutral state (neither commanding open nor close) until a signal is received from the Vector Valve Enable Logic. The Vector Valve Enable Logic monitors the channel A and B EFW Actuation logics. When an EFW Actuation occurs, the vector enable logic enables the vector logic to generate open or close signals to the EFW valves depending on the relative values of SG pressures.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

Automatic isolation of MFW and main steam line was assumed in the safety analyses to mitigate the consequences of main steam line or MFW line ruptures. The FSAR analyses for steam line breaks (SLBs) was generated before the development and installation of the safety grade EFIC System, which currently performs these automatic safety functions. The FSAR analysis, for example, assumes main steam line isolation through turbine stop valve closure based on an integrated control system signal. This same function is provided by the EFIC System by a safety grade signal that closes the Main Steam Line Isolation valves. The analyses are bounding, and the use of the EFIC System is consistent with the licensing position to take credit for safety grade systems to mitigate the consequences of an accident.

Similarly, vector valve control was not credited in the FSAR SLB analysis. Operator action was credited with isolating EFW to the affected SG within the first 60 seconds. This function would be automatically performed by EFIC. Therefore, the FSAR analysis remains conservative relative to the inclusion of the vector valve logic.

Automatic initiation of EFW is credited in the loss of main feedwater analysis. The automatic actuation was based on the SG low level function of EFIC, although EFIC would initiate EFW based on the loss of both MFW pumps as well.

The EFIC logic satisfies Criterion 3 of the NRC Policy Statement.

LCO

Two channels each of MFW and Main Steam Line Isolation, Vector Valve Enable, and EFW Actuation logics shall be OPERABLE. There are only two channels of automatic actuation logic per Function. Therefore, violation of this LCO could result in a complete loss of the automatic Function assuming a single failure of the other channel.

APPLICABILITY

The MFW and Main Steam Line Isolation automatic actuation logics shall be OPERABLE in MODES 1, 2, and 3 because SG inventory can be at a high energy level and can contribute significantly to the peak containment pressure during a

(continued)

BASES

APPLICABILITY (continued)

secondary side line break. In MODES 4, 5, and 6, the energy level is low and the secondary side feedwater flow rate is low or nonexistent.

The EFW automatic actuation and vector enable logics shall be OPERABLE in MODES 1, 2, and 3 because the SGs are being used for heat removal from the primary system. During these MODES, the core power and heat removal requirements are the greatest, and if the normal source of feedwater is lost, EFW must be initiated rapidly to minimize the overheating of the primary system.

For portions of MODE 4 and for all of MODES 5 and 6, the primary system temperatures are too low to allow the SGs to effectively remove energy.

ACTIONS

If a channel is found inoperable, then all affected logic Functions provided by that channel must be declared inoperable and the LCO Condition entered for the particular protection function affected.

For this LCO, a Note has been added to the ACTIONS indicating that separate Condition entry is allowed for each EFIC logic Function.

A.1

Condition A applies when one or more EFIC logic Functions in a single channel are inoperable (i.e., channel A could be inoperable for all four EFIC logic Functions and Condition A would still be applicable) with all Functions in the other channel OPERABLE. This Condition is equivalent to failure of one EFW, Main Steam Line Isolation, and MFW Isolation train.

With one automatic actuation logic channel of one or more EFIC Functions inoperable, the associated EFIC train must be restored to OPERABLE status. Since there are only two automatic actuation logic channels per EFIC Function, the condition of one channel inoperable is analogous to having one train of a two train Engineered Safety Feature (ESF) System inoperable. The system safety function can be accomplished; however, a single failure cannot be taken.

(continued)

BASES

ACTIONS

A.1 (continued)

Therefore, the failed channel(s) must be restored to OPERABLE status to re-establish the system's single-failure tolerance.

Condition A can be thought of as equivalent to failure of a single train of a two train safety system (e.g., the safety function can be accomplished, but a single failure cannot be taken). Thus, the Completion Time of 72 hours has been chosen to be consistent with Completion Times for restoring one inoperable ESF System train.

The EFIC System has not been analyzed for failure of one train of one Function and the opposite train of the same Function. In this condition, the potential for system interactions that disable heat removal capability on EFW has not been evaluated. Consequently, any combination of failures in both channels A and B is not covered by Condition A and must be addressed by entry into LCO 3.0.3.

B.1 and B.2

If Required Action A.1 cannot be met within the required Completion Time, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required MODES from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.3.13.1

This SR requires the performance of a CHANNEL FUNCTIONAL TEST to ensure that the channels can perform their intended functions. This test verifies MFW and Main Steam Line Isolation and EFW initiation automatic actuation logics are functional. This test simulates the required inputs to the logic circuit and verifies successful operation of the automatic actuation logic. The test need not include actuation of the end device. This is due to the risk of a unit transient caused by the closure of valves associated

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.13.1 (continued)

with MFW and Main Steam Line Isolation or actuation of EFW during testing at power. The Frequency of 31 days is based on operating experience, which has demonstrated the rarity of more than one channel failing within the same 31 day interval.

REFERENCES

1. FSAR, Chapter [7].

B 3.3 INSTRUMENTATION

B 3.3.14 Emergency Feedwater Initiation and Control (EFIC)—Emergency Feedwater (EFW)—Vector Valve Logic

BASES

BACKGROUND

The function of the EFW vector valve logic is to determine whether EFW should not be fed to one or the other steam generator. This is to preclude the continued addition of EFW to a depressurized once through steam generator (SG) and, thus, minimize the overcooling effects of a steam leak. Each vector logic may isolate EFW to one SG or the other, never both.

There are four sets of vector valve logic; one in each channel of EFIC. Each set of vector valve logic receives SG pressure information from bistables located in the input logic of the same EFIC channel. The pressure information received is:

a. SG "A" pressure less than 600 psig;

b. SG "B" pressure less than 600 psig;

c. SG "A" pressure 125 psid greater than SG "B" pressure; and

d. SG "B" pressure 125 psid greater than SG "A" pressure.

Each vector valve logic also receives a vector/control enable signal from both EFIC channel A and channel B when EFW is actuated.

The vector valve logic develops signals for open and close control of SG "A" and "B" EFW valves.

The vector valve logic outputs are in a neutral state with the valves fully open until enabled by the control/vector enable from the channel A or B trip logics. When enabled, the vector valve logic can issue close commands to the EFW control valves and open or close commands to the EFW isolation valves per the selected channel assignments.

(continued)

BASES

BACKGROUND (continued)

The valve open/close commands are determined by the relative values of steam generator pressures as follows:

PRESSURE STATUS	SG VALVES "A"	SG VALVES "B"
If SG "A" & SG "B" > 600 psig	Open	Open
If SG "A" > 600 psig & SG "B" < 600 psig	Open	Close
If SG "A" < 600 psig & SG "B" > 600 psig	Close	Open
If SG "A" & SG "B" < 600 psig		
<u>AND</u>		
SG "A" & SG "B" within 125 psid	Open	Open
SG "A" 125 psid > SG "B"	Open	Close
SG "A" 125 psid > SG "A"	Close	Open

APPLICABLE SAFETY ANALYSES

Automatic isolation of main feedwater (MFW) and main steam line was assumed in the safety analyses to mitigate the consequences of main steam line or MFW line ruptures. The FSAR analysis for steam line breaks (SLBs) was generated before the development and installation of the safety grade EFIC System, which currently performs these automatic safety functions. The FSAR analysis, for example, assumes main steam line isolation through turbine stop valve closure based on an integrated control system signal. This same function is provided by the EFIC System by a safety grade signal that closes the main steam line isolation valves. The analyses are bounding, and the use of the EFIC System is consistent with the licensing position to take credit for safety grade systems to mitigate the consequences of an accident.

Similarly, vector logic valve control was not credited in the FSAR SLB analysis. Operator action was credited with isolating EFW to the affected SG within the first

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

60 seconds. This function would be automatically performed by EFIC. Therefore, the FSAR analysis remains conservative relative to the inclusion of the vector valve logic.

EFW vector valve logic response time is included in the required response time for each EFW actuation initiation function instrumentation and is not specified separately.

The EFIC—EFW—vector valve logic satisfies Criterion 3 of the NRC Policy Statement.

LCO

Four channels of the EFIC—EFW—vector valve logic module are required to be OPERABLE. The necessity for four channels is discussed in the BASES for ACTIONS. The 600 psig and 125 psid setpoints were chosen as discussed in Specification B 3.3.11, "EFIC System Instrumentation." The feed only good generator verification study assumed a differential pressure vector value of 150 psid. The 125 psid setpoint conservatively assumes a 25 psi margin for instrument error. Failure to meet this LCO results in not being able to meet the single-failure criterion.

APPLICABILITY

EFIC—EFW—vector valve logic is required in MODES 1, 2, and 3 because the SGs are relied on in these MODES for required RCS heat removal. In MODES 4, 5, and 6, heat removal requirements are reduced and may be provided by the Decay Heat Removal System. Therefore, vector valve logic is not required to be OPERABLE in these MODES.

ACTIONS

A.1

The function of the EFIC-EFW control/isolation valves and the vector valve logic is to meet the single-failure criterion while maintaining the capability to:

a. Provide EFW on demand; and

b. Isolate an SG when required.

(continued)

BASES

ACTIONS

A.1 (continued)

These conflicting requirements result in the necessity for two valves in series, in parallel with two valves in series, and a four channel valve command system.

With one channel inoperable, the system cannot meet the single-failure criterion and still meet the dual functional criteria previously described. Therefore, when one vector valve logic channel is inoperable, the channel must be restored to OPERABLE status within 72 hours. This is analogous to having one EFW train inoperable; wherein a 72 hour Completion Time is provided by the Required Actions of LCO 3.7.4, "EFW System." As such, the Completion Time of 72 hours is based on engineering judgement.

B.1 and B.2

If Required Action A.1 cannot be met within the required Completion Time, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.3.14.1

SR 3.3.14.1 is the performance of a CHANNEL FUNCTIONAL TEST every 31 days. This test demonstrates that the EFIC—EFW—vector valve logic performs its function as desired. The Frequency is based on operating experience that demonstrates the rarity of more than one channel failing within the same 31 day interval.

REFERENCES

None.

B 3.3 INSTRUMENTATION

B 3.3.15 Reactor Building (RB) Purge Isolation—High Radiation

BASES

BACKGROUND

The RB Purge Isolation—High Radiation Function closes the RB purge valves. This action isolates the RB atmosphere from the environment to minimize releases of radioactivity in the event an accident occurs. The high radiation signal indicates a failure of a barrier to the fuel radioactivity, and most likely a loss of coolant accident. The purge valves must begin to shut rapidly to ensure they reach a completely closed position prior to excessive pressures in the RB, against which the valves may not close.

The radiation monitoring system measures the activity in a representative sample of air drawn in succession through a particulate sampler, an iodine sampler, and a gas sampler. The LCO addresses only the gas sampler portion of this system. The sensitive volume of the gas sampler is shielded with lead and monitored by a Geiger-Mueller detector. The air sample is taken from the center of the purge exhaust duct through an isokinetic nozzle installed in the duct at a point selected for reduced turbulence.

If a gaseous activity flow rate of approximately 1E-2 μCi/sec (Kr-85) is exceeded, the monitor will alarm and initiate closure of the purge valves. This activity flow rate is selected on the basis of 50,000 scfm flow rate in the purge exhaust and on the basis of a gas monitor setpoint equal to two times the expected background at the location of the monitor, which will provide fast detection of any release. The alarm setpoints for the particulate and iodine channels indicate that an alarm is obtained after the monitor samples a maximum permissible concentration level for 8 hours. Therefore, a maximum of 1.3 mCi of Cs-137 or 67 μCi of DOSE EQUIVALENT I-131 will be released to the atmosphere during this period.

The closure of the purge valves ensures the RB remains as a barrier to fission product release. There is no bypass for this function. The closure of the purge valves provides an RB isolation assumed in the accident analysis.

(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 band for CHANNEL CALIBRATION accuracy (i.e., ± [rack calibration + comparator setting accuracy]).

The trip setpoints used in the bistables are based on the analytical limits derived from the FSAR, Section [14.1] (Ref. 1). The selection of these trip setpoints is such that adequate protection is provided when all sensor and processing time delays are taken into account to allow for calibration tolerances, instrumentation uncertainties, instrument drift, and severe environment errors for those channels that must function in harsh environments as defined by 10 CFR 50.49 (Ref. 2). Allowable Values specified in LCO 3.3.15 are conservatively adjusted with respect to the analytical limits. A detailed description of the methodology used to calculate the trip setpoints, including their explicit uncertainties, is provided in the "Unit Specific Setpoint Methodology" (Ref. 3). The actual nominal trip setpoint entered into the bistable is normally still more conservative than that specified by the Allowable Value to account for changes in random measurement errors detectable by a CHANNEL FUNCTIONAL TEST. One example of such a change in measurement error is drift during the surveillance interval. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value.

The Allowable Value in SR 3.3.15.3 is based on the methodology described in Reference 3, which incorporates all of the known uncertainties applicable for each channel. The magnitudes of these uncertainties are factored into the determination of each trip setpoint. All field sensors and signal processing equipment for these channels are assumed to operate within the allowances of these uncertainty magnitudes.

These Allowable Values are established to prevent violation of the accident acceptance criteria during anticipated operational occurrences (AOOs).

Setpoints in accordance with the Allowable Value will ensure that the consequences of Design Basis Accidents (DBAs) will

(continued)

BASES

BACKGROUND

Trip Setpoints and Allowable Values (continued)

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.

APPLICABLE SAFETY ANALYSES

The analysis for the maximum hypothetical accident assumes the RB remains intact, with penetrations that are unnecessary for core cooling isolated early in the event, within approximately [60] seconds. The closure of the purge valves ensures the RB integrity assumed in the analysis is maintained. The isolation of the RB has not been analyzed mechanistically in the dose calculations, although its rapid isolation is assumed.

The RB Purge Isolation System satisfies Criterion 3 of the NRC Policy Statement.

LCO

For sampling systems, OPERABILITY requires correct valve lineups, sample pump operation, filter motor operation, and detector OPERABILITY, when these sampling features are necessary to initiate a trip as assumed by the safety analysis or setpoint analysis.

Only the Allowable Values are specified for each RB Purge Isolation trip Function in the LCO. Nominal trip setpoints are specified in the unit specific setpoint calculations. The nominal setpoints are selected to ensure the setpoint measured by CHANNEL FUNCTIONAL TESTS does not exceed the Allowable Value if the bistable is performing as required. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable provided that operation and testing are consistent with the assumptions of the unit specific setpoint calculations. Each Allowable Value specified is more conservative than the analytical limit assumed in the safety analysis to account for instrument uncertainties associated with the trip function. These uncertainties are defined in the "Unit Specific Setpoint Methodology" (Ref. 3) and the Offsite Dose Calculation Manual.

(continued)

BASES

LCO (continued)

[For this unit, the basis for the setpoint Allowable Value is as follows:]

APPLICABILITY

The RB purge isolation—high radiation shall be OPERABLE in MODES 1, 2, 3, and 4. Outside of these MODES, the purge isolation must be OPERABLE whenever CORE ALTERATIONS or movement of irradiated fuel assemblies within the RB is taking place. These conditions are those under which the potential for fuel damage, and thus radiation release, is the greatest. While in MODES 5 and 6, without fuel handling in progress, the Purge Valve Isolation System does not need to be OPERABLE because the potential for a radioactive release is minimized and operator action is sufficient to ensure post accident offsite doses are maintained within the limits of 10 CFR 100. The need to use the purge valves in MODES 5 and 6 is in preparation for entry. This capability is required to minimize doses for personnel entering the building and is independent of the automatic isolation capability.

ACTIONS

A.1

With one channel inoperable in MODE 1, 2, 3, or 4, the RB purge valves must be placed and maintained in the closed position. This action accomplishes the safety function of the RB Purge Isolation—High Radiation Function. The 1 hour Completion Time is reasonable considering the time required to isolate the penetration and the relative importance of maintaining containment OPERABILITY during MODES 1, 2, 3, and 4.

B.1 and B.2

If Required Action A.1 cannot be met within the required Completion Time, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

(continued)

BASES

ACTIONS (continued)

C.1, C.2.1, C.2.2

Condition C applies to failure of the high radiation purge function during CORE ALTERATIONS or during movement of irradiated fuel assemblies within the RB.

With one channel inoperable during CORE ALTERATIONS or during movement of irradiated fuel assemblies within the RB, the RB purge valves must be closed, or CORE ALTERATIONS and movement of irradiated fuel assemblies within the RB must be suspended. Required Action C.1 accomplishes the function of the high radiation channel. Required Action C.2.1 and Required Action C.2.2 place the unit in a configuration in which purge isolation on high radiation is not required. The Completion Time of "Immediately" is consistent with the urgency associated with the loss of RB isolation capability under conditions in which the fuel handling accidents are possible and the high radiation function provides the only automatic actions to mitigate radiation release.

SURVEILLANCE REQUIREMENTS

SR 3.3.15.1

SR 3.3.15.1 is the performance of the CHANNEL CHECK for the RB purge isolation—high radiation instrumentation once every 12 hours to ensure that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. Performance of the CHANNEL CHECK helps to ensure that the instrumentation continues to operate properly between each CHANNEL CALIBRATION. The high radiation instrumentation should be compared to similar unit instruments located throughout the unit. If the radiation monitor uses keep alive sources or check sources OPERABLE from the control room, the CHANNEL CHECK should also note the detector's response to these sources.

Agreement criteria are determined by the unit staff, based on a combination of the channel instrument uncertainties,

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.15.1 (continued)

including isolation, indication, and readability. If a channel is outside the criteria, it may be an indication that the transmitter or the signal processing equipment has drifted outside its limit. If the channels are within the criteria, it is an indication that the channels are OPERABLE. The Frequency, about once every shift, is based on operating experience that demonstrates channel failure is rare. [At this unit, the following administrative controls and design features (e.g., downscale alarms) immediately alert operators to loss of function.]

SR 3.3.15.2

This SR requires the performance of a CHANNEL FUNCTIONAL TEST once every 92 days to ensure that the channels can perform their intended functions. This test verifies the capability of the instrumentation to provide the RB isolation. Any setpoint adjustment shall be consistent with the assumptions of the current unit specific setpoint analysis.

In MODES 1, 2, 3, and 4, the test does not include the actuation of the purge valves, as these valves are normally closed.

The justification of a 92 day Frequency, in view of the fact that there is only one channel, is Draft NUREG-1366 (Ref. 4).

SR 3.3.15.3

CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations to ensure that the channel remains operational between successive tests. CHANNEL CALIBRATIONS must be performed consistent with the unit specific setpoint analysis.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.15.3 (continued)

The CHANNEL CALIBRATION is a complete check of the instrumentation and detector. In MODES 1, 2, 3, and 4, the CHANNEL CALIBRATION does not include the actuation of the purge valves, since they are normally closed.

The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

REFERENCES

1. FSAR, Section [14.1].
2. 10 CFR 50.49.
3. "Unit Specific Setpoint Methodology."
4. Draft NUREG-1366.

B 3.3 INSTRUMENTATION

B 3.3.16 Control Room Isolation—High Radiation

BASES

BACKGROUND

The principal function of the Control Room Isolation—High Radiation is to provide an enclosed environment from which the unit can be operated following an uncontrolled release of radioactivity. The high radiation isolation function provides assurance that under the required conditions, an isolation signal will be given. The noble gas monitors located in the station vent stack provide isolation and shutdown of the normal Control Room Emergency Ventilation System (CREVS).

The control room isolation signal is provided by a single channel containing an iodine monitor with a scintillation detector and a gaseous monitor with a Geiger-Mueller detector. The iodine channel includes a particulate prefilter with the charcoal cartridge. If a radioactivity concentration above normal background level is detected or if sampling capability is lost, the monitor will initiate a shutdown of the normal duty supply fans and will place the ventilation dampers in their recirculation mode.

Trip Setpoints and Allowable Values

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

The trip setpoints used in the bistables are based on the analytical limits derived from the FSAR, Section [14.1] (Ref. 1). The selection of these trip setpoints indicates that adequate protection is provided when all sensor and processing time delays are taken into account. To allow for calibration tolerances, instrumentation uncertainties, and instrument drift, Allowable Values specified in LCO 3.3.15 are conservatively adjusted with respect to the analytical limits. A detailed description of the methodology used to calculate the trip setpoints, including their explicit uncertainties, is provided in the "Unit Specific Setpoint Methodology" (Ref. 2). The actual nominal trip setpoint

(continued)

BASES

BACKGROUND

Trip Setpoints and Allowable Values (continued)

entered into the bistable is more conservative than that specified by the Allowable Value to account for changes in random measurement errors that are detectable by a CHANNEL FUNCTIONAL TEST. One example of a change in measurement error is drift during the surveillance interval. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value.

APPLICABLE SAFETY ANALYSES

The CREVS is isolated when a reactor building high pressure Engineered Safety Feature Actuation System signal or a high radiation signal is received. For the first 4 days following a loss of coolant accident, the CREVS is operated in the total recirculation mode. Four days after the start of the accident, the CREVS is started in the intake and recirculation mode and continues to operate in this mode for 30 days. This intake slightly pressurizes the control room. In both cases, the air flows through charcoal filters that are 95% efficient for elemental, particulate, and organic materials. The high radiation function only performs the initial isolation function to begin the recirculation mode of operation.

The Control Room Isolation—High Radiation satisfies Criterion 3 of the NRC Policy Statement.

LCO

Only the Allowable Value is specified for each Control Room Isolation—High Radiation trip Function in the LCO. Nominal trip setpoints are specified in the unit specific setpoint calculations. The nominal setpoints are selected to ensure the setpoint measured by the CHANNEL FUNCTIONAL TEST does not exceed the Allowable Value if the bistable is performing as required. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable provided that operation and testing is consistent with the assumptions of the unit specific setpoint calculations. Each Allowable Value specified is more conservative than the analytical limit assumed in the safety analysis to account for instrument uncertainties appropriate to the trip function. These

(continued)

BASES

LCO (continued)

uncertainties are defined in the "Unit Specific Setpoint Methodology" (Ref. 2).

[At this unit, the basis for the Allowable Value is as follows:]

APPLICABILITY

The control room isolation capability on high radiation shall be OPERABLE whenever there is a chance for an accidental release of radioactivity. This includes MODES 1, 2, 3, 4, [5, and 6] [and during CORE ALTERATIONS] and all MODES and conditions during movement of irradiated fuel assemblies. If a radioactive release were to occur during any of these conditions, the control room would have to remain habitable to ensure reactor shutdown and cooling can be controlled from the main control room.

ACTIONS

A.1

Condition A applies to failure of the Control Room Isolation—High Radiation Function in MODE 1, 2, 3, or 4.

With one channel of Control Room Isolation—High Radiation inoperable, the CREVS must be placed in a condition that does not require the isolation to occur. To ensure that the ventilation system has been placed in a state equivalent to that which occurs after the high radiation isolation has occurred, one OPERABLE train of the CREVS is placed in the emergency recirculation mode of operation. Reactor operation can continue indefinitely in this state. The 1 hour Completion Time is a sufficient amount of time in which to take the Required Action.

The Required Action is modified by a Note, which requires the CREVS be placed in the toxic gas protection mode if automatic transfer to the toxic gas protection mode is inoperable, since the pressurization mode would increase vulnerability to toxic gas releases.

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

If the CREVS cannot be placed into recirculation mode while in MODE 1, 2, 3, or 4, actions must be taken to minimize the chances of an accident that could lead to radiation releases. The unit must be placed in at least MODE 3 within 6 hours, with a subsequent cooldown to MODE 5 within 36 hours. This places the reactor in a low energy state that allows greater time for operator action if habitation of the control room is precluded. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

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

Required Action C.1 is the same as discussed earlier for Condition A, except for Completion Time. If the CREVS cannot be placed into recirculation mode during [CORE ALTERATIONS or while] moving irradiated fuel assemblies, then Required Action C.2.1 and Required Action C.2.2 suspend actions that could lead to an accident that could release radioactivity resulting from a fuel handling accident.

Required Action C.2.1 and Required Action C.2.2 place the core in a safe and stable configuration in which it is less likely to experience an accident that could result in a release of radioactivity. The reactor must be maintained in these conditions until the automatic isolation capability is returned to operation or when manual action places one train of the CREVS into the emergency recirculation mode. The Completion Time of "Immediately" for Required Action C.2.1 and Required Action C.2.2 is consistent with the urgency of the situation and accounts for the high radiation function, which provides the only automatic Control Room Isolation Function capable of responding to radiation release due to a fuel handling accident. The Completion Time does not preclude placing any fuel assembly into a safe position before ceasing any such movement.

Note that in certain circumstances, such as fuel handling in the fuel building during power operation, both Condition A and Condition C may apply in the event of channel failure.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.3.16.1

SR 3.3.16.1 is the performance of a CHANNEL CHECK for the Control Room Isolation—High Radiation actuation instrumentation once every 12 hours to ensure that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious.

Performance of the CHANNEL CHECK helps ensure that the instrumentation continues to operate properly between each CHANNEL CALIBRATION. The high radiation instrumentation should be compared with similar unit instruments located throughout the unit. If the radiation monitor uses keep alive sources or check sources operated from the control room, the CHANNEL CHECK should also note the detector's response to these sources.

Agreement criteria are determined by the unit staff, based on a combination of the channel instrument uncertainties, including isolation, indication, and readability. If a channel is outside the criteria, it may be an indication that the transmitter or the signal processing equipment has drifted outside its limit. If the channels are within the criteria, it is an indication that the channels are OPERABLE. The Frequency, about once every shift, is based on operating experience that demonstrates channel failure is rare. [At this unit, the following administrative controls and design features (e.g., downscale alarms) immediately alert operators to loss of function.]

SR 3.3.16.2

A Note defines a channel as being OPERABLE for up to 3 hours while bypassed for surveillance testing. The Note allows channel bypass for testing without defining it as inoperable, although during this time period it cannot actuate a control room isolation. This is based on the

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.16.2 (continued)

average time required to perform channel surveillance. It is not acceptable to routinely remove channels from service for more than 3 hours to perform required surveillance testing.

SR 3.3.16.2 is the performance of a CHANNEL FUNCTIONAL TEST once every 92 days to ensure that the channels can perform their intended functions. This test verifies the capability of the instrumentation to provide the automatic Control Room Isolation. Any setpoint adjustment shall be consistent with the assumptions of the current unit specific setpoint analysis.

The justification of a 92 day Frequency, in view of the fact that there is only one channel, is Draft NUREG-1366 (Ref. 3).

SR 3.3.16.3

This SR requires the performance of a CHANNEL CALIBRATION with a setpoint Allowable Value of ≤ [25] mR/hr to ensure that the instrument channel remains operational with the correct setpoint. This test is a complete check of the instrument loop and the transmitter.

CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations to ensure that the channel remains operational between successive tests. CHANNEL CALIBRATIONS must be performed consistent with the unit specific setpoint analysis.

The Frequency is based on the assumption of an [18] month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis and is consistent with the typical refueling cycle.

(continued)

BASES (continued)

REFERENCES

1. FSAR, Section [14.1].
2. "Unit Specific Setpoint Methodology."
3. Draft NUREG-1366.

B 3.3 INSTRUMENTATION

B 3.3.17 Post Accident Monitoring (PAM) Instrumentation

BASES

BACKGROUND

The primary purpose of the PAM instrumentation is to display unit variables that provide information required by the control room operators during accident situations. This information provides the necessary support for the operator to take the manual actions for which no automatic control is provided and that are required for safety systems to accomplish their safety functions for Design Basis Events.

The OPERABILITY of the accident monitoring instrumentation ensures that there is sufficient information available on selected unit parameters to monitor and to assess unit status and behavior following an accident.

The availability of accident monitoring instrumentation is important so that responses to corrective actions can be observed, and so that the need for and magnitude of further actions can be determined. These essential instruments are identified by [Unit Specific Documents] (Ref. 1) addressing the recommendations of Regulatory Guide 1.97 (Ref. 2) as required by Supplement 1 to NUREG-0737 (Ref. 3).

The instrument channels required to be OPERABLE by this LCO equate to two classes of parameters identified during unit specific implementation of Regulatory Guide 1.97 as Type A and Category I variables.

Type A variables are included in this LCO because they provide the primary information that permits the control room operator to take specific manually controlled actions that are required when no automatic control is provided and that are required for safety systems to accomplish their safety functions for Design Basis Accidents (DBAs). Because the list of Type A variables widely differs between units, Table 3.3.17-1 in the accompanying LCO contains only those examples of Type A variables that may also be Category I.

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

- Determine whether systems important to safety are performing their intended functions;

(continued)

BASES

BACKGROUND (continued)

- Provide information to the operators that will enable them to determine the potential for causing a gross breach of the barriers to radioactivity release; and
- Provide information regarding the release of radioactive materials to allow for early indication of the need to initiate action necessary to protect the public and to estimate the magnitude of any impending threat.

These key variables are identified by unit specific Regulatory Guide 1.97 analysis (Ref. 1). This analysis identifies the unit specific Type A and Category I variables and provides justification for deviating from the NRC proposed list of Category I variables.

[Reviewer's Note: Table 3.3.17-1 provides a list of variables typical of those identified by a unit specific Regulatory Guide 1.97 analysis (Ref. 1). Table 3.3.17-1 in unit specific Technical Specifications shall list all Type A and Category I variables identified by the unit specific Regulatory Guide 1.97 analysis, as amended by the NRC's Safety Evaluation Report (SER).

The specific instrument Functions listed in Table 3.3.17-1 are discussed in the LCO Section.]

APPLICABLE SAFETY ANALYSES

The PAM instrumentation ensures the availability of information so that the control room operating staff can:

- Perform the diagnosis specified in the emergency operating procedures. These variables are restricted to preplanned actions for the primary success path of DBAs (e.g., loss of coolant accident (LOCA));
- Take the specified, preplanned, manually controlled actions, for which no automatic control is provided, which are required for safety systems to accomplish their safety functions;
- Determine whether systems important to safety are performing their intended functions;

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

- Determine the potential for causing a gross breach of the barriers to radioactivity release;
- Determine if a gross breach of a barrier has occurred; and
- Initiate action necessary to protect the public and estimate the magnitude of any impending threat.

The unit specific Regulatory Guide 1.97 analysis documents the process that identifies Type A and Category I non-Type A variables.

PAM instrumentation that meets the definition of Type A in Regulatory Guide 1.97 satisfies Criterion 3 of the NRC Policy Statement. Category I, non-type A, instrumentation must be retained in Technical Specifications because it is intended to assist operators in minimizing the consequences of accidents. Therefore, Category I, non-Type A variables are important for reducing public risk.

LCO

LCO 3.3.17 requires two OPERABLE channels for all but one Function to ensure no single failure prevents the operators from being presented with the information necessary to determine the status of the unit and to bring the unit to, and maintain it in, a safe condition following that accident.

Furthermore, provision of two channels allows a CHANNEL CHECK during the post accident phase to confirm the validity of displayed information. [More than two channels may be required at some units if the Regulatory Guide 1.97 analysis determines that failure of one accident monitoring channel results in information ambiguity (i.e., the redundant displays disagree) that could lead operators to defeat or to fail to accomplish a required safety function.]

The exception to the two channel requirement is containment isolation valve position. In this case, the important information is the status of the containment penetrations. The LCO requires one position indicator for each active containment isolation valve. This is sufficient to redundantly verify the isolation status of each isolable penetration either via indicated status of the active valve

(continued)

BASES

LCO (continued)

and prior knowledge of the passive valve or via system boundary status. If a normally active containment isolation valve is known to be closed and deactivated, position indication is not needed to determine status. Therefore, the position indication for valves in this state is not required to be OPERABLE.

The following list is a discussion of the specified instrument Functions listed in Table 3.3.17-1. These discussions are intended as examples of what should be provided for each Function when the unit specific list is prepared.

1. <u>Wide Range Neutron Flux</u>

   Wide Range Neutron Flux indication is provided to verify reactor shutdown. [For this unit, the Wide Range Neutron Flux channels consist of the following:]

2, 3. <u>Reactor Coolant System (RCS) Hot and Cold Leg Temperature</u>

   RCS Hot and Cold Leg Temperature instrumentation are Category I variables provided for verification of core cooling and long term surveillance. Reactor outlet temperature inputs to the RPS are provided by two fast response resistance elements and associated transmitters in each loop. The channels provide indication over a range of 32°F to 700°F.

4. <u>RCS Pressure (Wide Range)</u>

   RCS Pressure (Wide Range) instrumentation is provided for verification of core cooling and RCS integrity long term surveillance.

   Wide range RCS loop pressure is measured by pressure transmitters with a span of 0 psig to 3000 psig. The pressure transmitters are located outside the RB. Redundant monitoring capability is provided by two trains of instrumentation. Control room indications are provided through the inadequate core cooling plasma display. The inadequate core cooling plasma

(continued)

BASES

LCO

4. RCS Pressure (Wide Range) (continued)

display is the primary indication used by the operator during an accident. Therefore, the accident monitoring specification deals specifically with this portion of the instrument string.

In some units, RCS Pressure is a Type A variable because the operator uses this indication to monitor the cooldown of the RCS following a steam generator (SG) tube rupture or small break LOCA. Operator actions to maintain a controlled cooldown, such as adjusting SG pressure or level, would use this indication. In addition, high pressure injection (HPI) flow is throttled based on RCS Pressure and subcooled margin. For some small break LOCAs, low pressure injection (LPI) may actuate with system pressure stabilizing above the shutoff head of the LPI pumps. If this condition exists, the operator is instructed to verify HPI flow and then terminate LPI flow prior to exceeding 30 minutes of LPI pump operation against a deadhead pressure. RCS Pressure, in conjunction with LPI flow, is also used to determine if a core flood line break has occurred.

5. Reactor Vessel Water Level

Reactor Vessel Water Level instrumentation is provided for verification and long term surveillance of core cooling. The reactor vessel level monitoring system provides a direct measurement of the collapsed liquid level above the fuel alignment plate. The collapsed level represents the amount of liquid mass that is in the reactor vessel above the core. Measurement of the collapsed water level is selected because it is a direct indication of the water inventory.

The collapsed level is obtained over the same temperature and pressure range as the saturation measurements, thereby encompassing all operating and accident conditions where it must function. Also, it functions during the recovery interval. Therefore, it is designed to survive the high steam temperature that may occur during the preceding core recovery interval.

(continued)

BASES

LCO

5. Reactor Vessel Water Level (continued)

The level range extends from the top of the vessel down to the top of the fuel alignment plate. The response time is short enough to track the level during small break LOCA events. The resolution is sufficient to show the initial level drop, the key locations near the hot leg elevation, and the lowest levels just above the alignment plate. This provides the operator with adequate indication to track the progression of the accident and to detect the consequences of its mitigating actions or the functionality of automatic equipment.

[For this unit, the Reactor Vessel Water Level channels consist of the following:]

6. Containment Sump Water Level (Wide Range)

Containment Sump Water Level (Wide Range) instrumentation is provided for verification and long term surveillance of RCS integrity. [For this unit, the Containment Sump Water Level instrumentation consists of the following:]

7. Containment Pressure (Wide Range)

Containment Pressure (Wide Range) instrumentation is provided for verification of RCS and containment OPERABILITY. [For this unit, Containment Pressure instrumentation consists of the following:]

8. Containment Isolation Valve Position

PCIV position is provided for verification of containment integrity. In the case of PCIV position, the important information is the isolation status of the containment penetration. The LCO requires one channel of valve position indication in the control room to be OPERABLE for each active PCIV in a containment penetration flow path, i.e., two total channels of PCIV position indication for a penetration flow path with two active valves. For containment

(continued)

BASES

LCO

8. Containment Isolation Valve Position (continued)

penetrations with only one active PCIV having control room indication, Note (b) requires a single channel of valve position indication to be OPERABLE. This is sufficient to redundantly verify the isolation status of each isolable penetration via indicated status of the active valve, as applicable, and prior knowledge of passive valve or system boundary status. If a penetration flow path is isolated, position indication for the PCIV(s) in the associated penetration flow path is not needed to determine status. Therefore, the position indication for valves in an isolated penetration flow path is not required to be OPERABLE. [For this plant, the PCIV position PAM instrumentation consists of the following:]

9. Containment Area Radiation (High Range)

Containment Area Radiation (High Range) instrumentation is provided to monitor the potential for significant radiation releases and to provide release assessment for use by operators in determining the need to invoke site emergency plans. [For this unit, the Containment Area Radiation instrumentation consists of the following:]

10. Containment Hydrogen Concentration

Containment Hydrogen Concentration instrumentation is provided to detect high hydrogen concentration conditions that represent a potential for containment breach. This variable is also important in verifying the adequacy of mitigating actions. [For this unit, the Containment Hydrogen Concentration instrumentation consists of the following:]

11. Pressurizer Level

Pressurizer Level instrumentation is used to determine whether to terminate safety injection (SI), if still in progress, or to reinitiate SI if it has been stopped. Knowledge of pressurizer water level is also

(continued)

BASES

LCO

11. Pressurizer Level (continued)

used to verify the unit conditions necessary to establish natural circulation in the RCS and to verify that the unit is maintained in a safe shutdown condition. [For this unit, the Pressurizer Level instrumentation consists of the following:]

12. Steam Generator Water Level

Steam Generator Water Level instrumentation is provided to monitor operation of decay heat removal via the SG. The indication of SG level is the extended startup range level instrumentation, covering a span of 6 inches to 394 inches above the lower tubesheet. The measured differential pressure is displayed in inches of water at 68°F. Temperature compensation for this indication is performed manually by the operator. Redundant monitoring capability is provided by two trains of instrumentation. The uncompensated level signal is input to the unit computer, a control room indicator, and the Emergency Feedwater (EFW) Control System.

SG level indication is used by the operator to manually raise and control SG level to establish boiler condenser heat transfer. Operator action is initiated on a loss of subcooled margin. Feedwater flow is increased until the indicated extended startup range level reaches the boiler condenser setpoint.

13. Condensate Storage Tank (CST) Level

CST Level instrumentation is provided to ensure a water supply for EFW. The CST provides the assured, safety grade water supply for the EFW System. The CST consists of two identical tanks connected by a common outlet header. Inventory is monitored by a 0 inch to 144 inch level indication for each tank. CST Level is displayed on a control room indicator, strip chart recorder, and unit computer. In addition, a control room annunciator alarms on low level.

(continued)

BASES

LCO

13. Condensate Storage Tank (CST) Level (continued)

CST Level is the primary indication used by the operator to identify loss of CST volume and replenish the CST or align suction to the EFW pumps from the hotwell.

14. Core Exit Temperature

Core Exit Temperature is provided for verification and long term surveillance of core cooling. An evaluation was made of the minimum number of valid core exit thermocouples (CETs) necessary for inadequate core cooling detection. The evaluation determined the reduced complement of CETs necessary to detect initial core recovery and to trend the ensuing core heatup. The evaluations account for core nonuniformities and cold leg injection. Based on these evaluations, adequate or inadequate core cooling detection is ensured with two sets of five valid CETs.

The subcooling margin monitor takes the average of the five highest CETs for each of the ICCM trains. Two channels ensure that a single failure will not disable the ability to determine the representative core exit temperature.

15. Emergency Feedwater Flow

EFW Flow instrumentation is provided to monitor operation of decay heat removal via the SGs. The EFW Flow to each SG is determined from a differential pressure measurement calibrated to a span of 0 gpm to 1200 gpm. Redundant monitoring capability is provided by two independent trains of instrumentation for each SG. Each differential pressure transmitter provides an input to a control room indicator and the unit computer.

EFW Flow is the primary indication used by the operator to determine the need to throttle flow during an SLB accident to prevent the EFW pumps from operating in runout conditions. EFW Flow is also used by the operator to verify that the EFW System is

(continued)

BASES

LCO

15. Emergency Feedwater Flow (continued)

delivering the correct flow to each SG. However, the primary indication used by the operator to ensure an adequate inventory is SG level.

RCS pressure is used by the operator to monitor the cooldown of the RCS following an SG tube rupture or small break LOCA. In addition, HPI flow is throttled based on RCS pressure and subcooled margin. The indication is also used to identify an LPI pump operating at system pressures above its shutoff head. If this condition exists, the operator is instructed to verify this condition exists, to verify HPI flow, and to terminate LPI flow prior to exceeding 30 minutes of LPI pump operation against a deadhead pressure. RCS pressure, in conjunction with LPI flow, is also used to determine if a core flood line break has occurred.

APPLICABILITY

The PAM instrumentation LCO is applicable in MODES 1, 2, and 3. These variables are related to the diagnosis and preplanned actions required to mitigate DBAs. The applicable DBAs are assumed to occur in MODES 1, 2, and 3. In MODES 4, 5, and 6, unit conditions are such that the likelihood of an event occurring that would require PAM instrumentation is low; therefore, the PAM instrumentation is not required to be OPERABLE in these MODES.

ACTIONS

The ACTIONS are modified by two Notes. Note 1 is added to the ACTIONS to exclude the MODE change restriction of LCO 3.0.4. This exception allows entry into an applicable MODE while relying on the ACTIONS even though the ACTIONS may eventually require a unit shutdown. This exception is acceptable due to the passive function of the instruments, the operator's ability to respond to an accident utilizing alternate instruments and methods, and the low probability of an event requiring these instruments.

Note Two is added to the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each

(continued)

BASES

ACTIONS (continued)

Function listed in Table 3.3.17-1. The Completion Time(s) of the inoperable channels of a Function will be tracked separately for each Function starting from the time the Condition was entered for that Function.

A.1

When one or more Functions have one required channel inoperable, the inoperable channel must be restored to OPERABLE status within 30 days. The 30 day Completion Time is based on operating experience. This takes into account the remaining OPERABLE channel (or, in the case of a Function that has only one required channel, other non-Regulatory Guide 1.97 instrument channels to monitor the Function), the passive nature of the instrument (no critical automatic action is assumed to occur from these instruments), and the low probability of an event requiring PAM instrumentation during this interval.

B.1

Required Action B.1 specifies initiation of action described in Specification 5.6.8, that requires a written report to be submitted to the NRC. This report discusses the results of the root cause evaluation of the inoperability and identifies proposed restorative actions. This action is appropriate in lieu of a shutdown requirement since alternative actions are identified before loss of functional capability and given the likelihood of unit conditions that would require information provided by this instrumentation. The Completion Time of "Immediately" for Required Action B.1 ensures the requirements of Specification 5.6.8 are initiated.

C.1

When one or more Functions have two required channels inoperable (i.e., two channels inoperable in the same Function), one channel in the Function should be restored to OPERABLE status within 7 days. This Condition does not apply to the hydrogen monitor channels. The Completion Time of 7 days is based on the relatively low probability

(continued)

ACTIONS

C.1 (continued)

of an event requiring PAM instrumentation action operation and the availability of alternative means to obtain the required information. Continuous operation with two required channels inoperable in a Function is not acceptable because the alternate indications may not fully meet all performance of qualification requirements applied to the PAM instrumentation. Therefore, requiring restoration of one inoperable channel of the Function limits the risk that the PAM Function will be in a degraded condition should an accident occur.

D.1

When two required hydrogen monitor channels are inoperable, Required Action D.1 requires one channel to be restored to OPERABLE status. This action restores the monitoring capability of the hydrogen monitor. The 72 hour Completion Time is based on the relatively low probability of an event requiring hydrogen monitoring and the availability of alternative means to obtain the required information. Continuous operation with two required channels inoperable is not acceptable because alternate indications are not available.

E.1

Required Action E.1 directs entry into the appropriate Condition referenced in Table 3.3.17-1. The applicable Condition referenced in the Table is Function dependent. Each time an inoperable channel has not met any Required Action of Condition C or D, as applicable, and the associated Completion Time has expired, Condition E is entered for that channel and provides for transfer to the appropriate subsequent Condition.

F.1

If the Required Action and associated Completion Time of Conditions C or D are not met and Table 3.3.17-1 directs entry into Condition F, the unit must be brought to a MODE in which the requirements of this LCO do not apply. To

(continued)

BASES

ACTIONS

F.1 (continued)

achieve this status, the unit must be brought to at least MODE 3 within 6 hours and MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

G.1

At this unit, alternative means of monitoring Containment Area Radiation have been developed and tested. These alternative means may be temporarily installed if the normal PAM channel cannot be restored to OPERABLE status within the allowed time.

If these alternative means are used, the Required Action is not to shut the unit down, but rather to follow the directions of Specification 5.6.8, in the Administrative Controls section of the Technical Specifications. The report provided to the NRC should discuss the alternative means used, describe the degree to which the alternative means are equivalent to the installed PAM channels, justify the areas in which they are not equivalent, and provide a schedule for restoring the normal PAM channels.

In the case of reactor vessel level, Reference 4 determined that the appropriate Required Action was not to shut the unit down, but rather to follow the directions of Specification 5.6.8.

[At this unit, the alternative monitoring provisions consist of the following:]

SURVEILLANCE REQUIREMENTS

As noted at the beginning of the SRs, the SRs apply to each PAM instrumentation Function in Table 3.3.17-1.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.17.1

Performance of the CHANNEL CHECK once every 31 days for each required instrumentation channel that is normally energized ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel with a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; therefore, it is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION. The high radiation instrumentation should be compared with similar unit instruments located throughout the unit. If the radiation monitor uses keep alive sources or check sources OPERABLE from the control room, the CHANNEL CHECK should also note the detector's response to these sources.

Agreement criteria are determined by the unit staff, based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit. If the channels are within the criteria, it is an indication that the channels are OPERABLE. If the channels are normally off scale during times when surveillance is required, the CHANNEL CHECK will only verify that they are off scale in the same direction. Offscale low current loop channels are verified to be reading at the bottom of the range and not failed downscale.

The Frequency is based on unit operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal but more frequent checks of channels during normal operational use of the displays associated with this LCO's required channels.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.17.2

A CHANNEL CALIBRATION is performed every [18] months or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument channel, including the sensor. This test verifies the channel responds to measured parameters within the necessary range and accuracy.

A Note clarifies that the neutron detectors are not required to be tested as part of the CHANNEL CALIBRATION. There is no adjustment that can be made to the detectors. Furthermore, adjustment of the detectors is unnecessary because they are passive devices, with minimal drift. Slow changes in detector sensitivity are compensated for by performing the daily calorimetric calibration and the monthly axial channel calibration.

For the Containment Area Radiation instrumentation, a CHANNEL CALIBRATION may consist of an electronic calibration of the channel, not including the detector, for range decades above 10 R/hr, and a one point calibration check of the detector below 10 R/hr with a gamma source.

The Frequency is based on operating experience and consistency with the typical industry refueling cycle and is justified by the assumption of an [18] month calibration interval in the determination of the magnitude of equipment drift.

REFERENCES

1. [Unit Specific Documents (e.g., FSAR, NRC Regulatory Guide 1.97 SER letter).]

2. Regulatory Guide 1.97.

3. NUREG-0737, 1979.

4. 32-1177256-00, "Technical Basis for Reactor Vessel Level Indication System (RVLIS) Action Statement," April 10, 1990.

B 3.3 INSTRUMENTATION

B 3.3.18 Remote Shutdown System

BASES

BACKGROUND

The Remote Shutdown System provides the control room operator with sufficient instrumentation and controls to place and maintain the unit in a safe shutdown condition from locations other than the control room. This capability is necessary to protect against the possibility that the control room becomes inaccessible. A safe shutdown condition is defined as MODE 3. With the unit in MODE 3, the Emergency Feedwater (EFW) System and the steam generator (SG) safety valves or the SG atmospheric dump valves can be used to remove core decay heat and meet all safety requirements. The long term supply of water for the EFW System and the ability to borate the Reactor Coolant System (RCS) from outside the control room allows extended operation in MODE 3.

In the event that the control room becomes inaccessible, the operators can establish control at the remote shutdown panel and place and maintain the unit in MODE 3. Not all controls and necessary transfer switches are located at the remote shutdown panel. Some controls and transfer switches will have to be operated locally at the switchgear, motor control panels, or other local stations. The unit automatically reaches MODE 3 following a unit shutdown and can be maintained safely in MODE 3 for an extended period of time.

The OPERABILITY of the Remote Shutdown System control and instrumentation Functions ensures that there is sufficient information available on selected unit parameters to place and maintain the unit in MODE 3 should the control room become inaccessible.

APPLICABLE SAFETY ANALYSES

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

The criteria governing the design and the specific system requirements of the Remote Shutdown System are located in 10 CFR 50, Appendix A, GDC 19 (Ref. 1).

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The Remote Shutdown System meets the NRC Policy Statement as a risk significant item for retention in the Technical Specifications.

LCO

The Remote Shutdown System LCO provides the requirements for the OPERABILITY of the instrumentation and controls necessary to place and maintain the unit in MODE 3 from a location other than the control room. The instrumentation and controls typically required are listed in Table 3.3.18-1 in the accompanying LCO.

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

- Core Reactivity Control (initial and long term);
- RCS Pressure Control;
- Decay Heat Removal via the EFW System and the SG safety valves or SG atmospheric dump valves;
- RCS Inventory Control via charging flow; and
- Safety support systems for the above Functions, including service water, component cooling water, and onsite power, including the diesel generators.

A Function of a Remote Shutdown System is OPERABLE if all instrument and control channels needed to support the Function are OPERABLE. In some cases, Table 3.3.18-1 may indicate that the required information or control capability is available from several alternate sources. In these cases, the Function is OPERABLE as long as [one channel of any of] the alternate information or control sources are OPERABLE.

(continued)

BASES

LCO (continued)	The Remote Shutdown System instruments and control circuits covered by this LCO do not need to be energized to be considered OPERABLE. This LCO is intended to ensure the Remote Shutdown System instruments and control circuits will be OPERABLE if unit conditions require that the Remote Shutdown System be placed in operation.
APPLICABILITY	The Remote Shutdown System LCO is applicable in MODES 1, 2, and 3. This is required so that the unit can be placed and maintained in MODE 3 for an extended period of time from a location other than the control room. This LCO is not applicable in MODE 4, 5, or 6. In these MODES, the unit is already subcritical and is in a condition of reduced RCS energy. Under these conditions, considerable time is available to restore necessary instrument and control Functions if control room instruments become unavailable.
ACTIONS	The ACTIONS is modified by two Notes. Note 1 excludes the MODE change restriction of LCO 3.0.4. This exception allows entry into an applicable MODE while relying on the ACTIONS, even though the ACTIONS may eventually require a unit shutdown. This exception is acceptable due to the low probability of an event requiring these instruments. Note 2 has been added to the ACTIONS to clarify the application of Completion Time rules. The Conditions of the Specification may be entered independently for each Function listed in Table 3.3.18-1. The Completion Time(s) of the inoperable channel(s)/train(s) of a Function will be tracked separately for each Function starting from the time the Condition was entered for that Function. <u>A.1</u> Condition A addresses the situation where one or more required Functions of the Remote Shutdown System are inoperable. This includes any Function listed in Table 3.3.18-1 and the control and transfer switches.

(continued)

BASES

ACTIONS

A.1 (continued)

The Required Action is to restore the required Function to OPERABLE status within 30 days. The Completion Time is based on operating experience and takes into account the remaining OPERABLE division and the low probability of an event that would require evacuation of the control room.

B.1 and B.2

If Required Action A.1 cannot be met within the required Completion Time, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.3.18.1

Performance of the CHANNEL CHECK once every 31 days for each required instrumentation channel that is normally energized ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value.

Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; therefore, it is key in verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION. Agreement criteria are determined by the unit staff based on a combination of the channel instrument uncertainties, including indication and readability. If the channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit. If the channels are within the criteria, it is an indication that the channels are

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.18.1 (continued)

OPERABLE. As specified in the Surveillance, a CHANNEL CHECK is only required for those channels that are normally energized. If the channels are normally off scale during times when surveillance is required, the CHANNEL CHECK will only verify that they are off scale in the same direction. Off scale low current loop channels are verified to be reading at the bottom of the range and not failed downscale.

The Frequency is based on unit operating experience, which demonstrates that channel failure is rare.

SR 3.3.18.2

SR 3.3.18.2 verifies each required Remote Shutdown System transfer switch and control circuit performs their intended function. This verification is performed from the remote shutdown panel and locally, as appropriate. Operation of the equipment from the remote shutdown panel is not necessary. The Surveillance can be satisfied by performance of a continuity check. This will ensure that if the control room becomes inaccessible, the unit can be placed and maintained in MODE 3 from the remote shutdown panel and the local control stations. The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience demonstrates that remote shutdown control channels seldom fail to pass the Surveillance when performed at the [18] month Frequency.

SR 3.3.18.3

CHANNEL CALIBRATION is a complete check of the instrument loop and sensor. The test verifies that the channel responds to measured parameters within the necessary range and accuracy.

A Note clarifies that the neutron detectors are not required to be tested as part of the CHANNEL CALIBRATION. There is no adjustment that can be made to the detectors. Furthermore, adjustment of the detectors is unnecessary

(continued)

BASES

SURVEILLANCE REQUIREMENTS	SR 3.3.18.3 (continued) because they are passive devices, with minimal drift. Slow changes in detector sensitivity are compensated for by performing the daily calorimetric calibration and the monthly axial channel calibration. The Frequency is based on operating experience and consistency with the typical industry refueling cycle and is justified by the assumption of an [18] month calibration interval in the determination of the magnitude of equipment drift.
REFERENCES	1. 10 CFR 50, Appendix A, GDC 19.

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This report documents the results of the combined effort of the NRC and the industry to produce improved Standard Technical Specifications (STS), Revision 1 for Babcock & Wilcox Plants. The changes reflected in Revision 1 resulted from the experience gained from license amendment applications to convert to these improved STS or to adopt partial improvements to existing technical specifications. This NUREG is the result of extensive public technical meetings and discussions between the Nuclear Regulatory Commission (NRC) staff and various nuclear power plant licensees, Nuclear Steam Supply System (NSSS) Owners Groups, NSSS vendors, and the Nuclear Energy Institute (NEI). The improved STS were developed based on the criteria in the Final Commission Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors, dated July 22, 1993. The improved STS will be used as the basis for individual nuclear power plant licensees to develop improved plant-specific technical specifications. This report contains three volumes. Volume 1 contains the Specifications for all chapters and sections of the improved STS. Volume 2 contains the Bases for Chapters 2.0 and 3.0, and Sections 3.1 - 3.3 of the improved STS. Volume 3 contains the Bases for Sections 3.4 - 3.9 of the improved STS.

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