B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core Safety Limits (SLs)

BASES

BACKGROUND

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

The restriction of this SL prevents overheating of the fuel and cladding, as well as possible cladding perforation, that would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady state peak linear heat rate (LHR) below the level at which fuel centerline melting occurs. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime, where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

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

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

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

BASES

APPLICABLE SAFETY ANALYSES

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

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

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

The Reactor Trip System (RTS) setpoints (Ref. 2), in combination with all the LCOs, are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System (RCS) temperature, pressure, RCS Flow, ΔI, 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 appropriate operation of the PMS and the steam generator safety valves.

The SLs represent a design requirement for establishing the RTS setpoints. LCO 3.4.1, “RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits,” or the assumed initial conditions of the safety analyses (as indicated in Section 7.2, Ref. 2) provide more restrictive limits to ensure that the SLs are not exceeded.

SAFETY LIMITS

The figure provided in the COLR shows the loci of points of THERMAL POWER, RCS pressure, and cold leg temperature for which the minimum DNBR is not less than the safety analysis limit, that fuel centerline temperature remains below melting, that the average enthalpy in the hot leg is less than or equal to the enthalpy of saturated liquid, or that the exit quality is within the limits defined by the DNBR correlation.

The reactor core SLs are established to preclude violation of the following fuel design criteria:

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

b. There must be at least a 95% probability at a 95% confidence level that the hot fuel pellet in the core does not experience centerline fuel melting.

BASES

SAFETY LIMITS (continued)

The reactor core SLs are used to define the various RPS functions such that the above criteria are satisfied during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). To ensure that the RPS precludes the violation of the above criteria, additional criteria are applied to the Overtemperature and Overpower ΔT reactor trip functions. That is, it must be demonstrated that the average enthalpy in the hot leg is less than or equal to the saturation enthalpy and the core exit quality is within the limits defined by the DNBR correlation. Appropriate functioning of the RPS ensures that for variations in the THERMAL POWER, RCS Pressure, RCS cold leg temperature, RCS flow rate, and ΔI that the reactor core SLs will be satisfied during steady state operation, normal operational transients, and AOOs.

APPLICABILITY

SL 2.1.1 only applies in MODES 1 and 2 because these are the only MODES in which the reactor is critical. Automatic protection functions are required to be OPERABLE during MODES 1 and 2 to ensure operation within the reactor core SLs. The steam generator safety valves or automatic protection actions serve to prevent RCS heatup to the reactor core SL conditions or to initiate a reactor trip function which forces the unit into MODE 3. Setpoints for the reactor trip functions are specified in LCO 3.3.1, “Reactor Trip System (RTS) Instrumentation.” In MODES 3, 4, 5, and 6, applicability is not required since the reactor is not generating significant THERMAL POWER.

SAFETY LIMIT VIOLATIONS

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

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

REFERENCES

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

2. Section 7.2, “Reactor Trip.”

B 2.0 SAFETY LIMITS (SLs)

B 2.1.2 Reactor Coolant System (RCS) Pressure SL

BASES

BACKGROUND

The SL on RCS pressure protects the integrity of the RCS against overpressurization. In the event of fuel cladding failure, fission products are released into the reactor coolant. The RCS then serves as the primary barrier in preventing the release of fission products into the atmosphere. By establishing an upper limit on RCS pressure, the continued integrity of the RCS is ensured. According to 10 CFR 50, Appendix A, GDC 14, “Reactor Coolant Pressure Boundary,” and GDC 15, “Reactor Coolant System Design” (Ref. 1), the reactor coolant pressure boundary (RCPB) design conditions are not to be exceeded during normal operation and anticipated operational occurrences (AOOs). Also, in accordance with GDC 28, “Reactivity Limits” (Ref. 1), reactivity accidents, including rod ejection, do not result in damage to the RCPB greater than limited local yielding.

The design pressure of the RCS is 2500 psia (2485 psig). During normal operation and AOOs, RCS pressure is limited from exceeding the design pressure by more than 10%, in accordance with Section III of the American Society of Mechanical Engineers (ASME) Code (Ref. 2). To ensure system integrity, all RCS components are hydrostatically tested at 125% of design pressure, according to the ASME Code requirements prior to initial operation when there is no fuel in the core. Following inception of unit operation, RCS components shall be pressure tested, in accordance with the requirements of ASME Code, Section XI (Ref. 3).

Overpressurization of the RCS could result in a breach of the RCPB. If such a breach occurs in conjunction with a fuel cladding failure, fission products could enter the containment atmosphere, raising concerns relative to limits on radioactive releases.

APPLICABLE SAFETY ANALYSES

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

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

BASES

APPLICABLE SAFETY ANALYSES (continued)

safety valves on the secondary plant are assumed to open when the steam pressure reaches the secondary plant safety valve settings.

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

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

a. RCS depressurization valves;

b. Steam line relief valves (SG PORVs);

c. Turbine Bypass System;

d. Reactor Control System;

e. Pressurizer Level Control System; or

f. Pressurizer spray.

SAFETY LIMITS

The maximum transient pressure allowed in the RCS pressure vessel, piping, valves, and fittings under the ASME Code, Section III, is 110% of design pressure; therefore, the SL on maximum allowable RCS pressure is 2733.5 psig.

APPLICABILITY

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

SAFETY LIMIT VIOLATIONS

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

Exceeding the RCS pressure SL may cause immediate RCS failure and create a potential for abnormal radioactive releases.

BASES

SAFETY LIMIT VIOLATIONS (continued)

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

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

REFERENCES

1. 10 CFR 50, Appendix A, GDC 14, GDC 15, and GDC 28.
2. ASME, Boiler and Pressure Vessel Code, Section III, Article NB-7000.
3. ASME Boiler and Pressure Vessel Code, Section XI, Article IWX-5000.
4. 10 CFR 50.34.
5. Section 7.2, “Reactor Trip.”

B 3.0 LIMITING CONDITIONS FOR OPERATION (LCO) APPLICABILITY

BASES

LCOs	LCO 3.0.1 through LCO 3.0.8 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 requirements 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 the 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 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.

BASES

LCO 3.0.2 (continued)

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. The reasons for intentionally relying on the ACTIONS include, but are not limited to, performance of Surveillances, preventive maintenance, corrective maintenance, or investigation of operational problems. Entering ACTIONS for these reasons must be done in a manner that does not compromise safety. Intentional entry into ACTIONS should not be made for operational convenience. 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 could 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.

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

BASES

LCO 3.0.3 (continued)

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 into 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. This reduces thermal stresses on components of the Reactor Coolant System and the potential for a plant upset that could challenge safety systems under conditions to which this Specification applies. The use and interpretation of specified times to complete the actions of LCO 3.0.3 are consistent with the discussion of Section 1.3, “Completion Times.”

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

a. The LCO is now met.

b. A Condition exists for which the Required Actions have now been performed.

c. ACTIONS exist that do not have expired Completion Times. These Completion Times are applicable from the point in time that the Condition was initially entered and not from the time LCO 3.0.3 is exited.

BASES

LCO 3.0.3 (continued)

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 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. 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. In MODES 5 and 6, LCO 3.0.8 provides actions for Conditions not covered in other Specifications.

Exceptions to 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 3.7.5, Spent Fuel Pool Water Level. This Specification has an Applicability of “At all times.” Therefore, this LCO can be applicable in any or all MODES. If the LCO and the Required Actions of LCO 3.7.5 are not met while in MODE 1, 2, or 3, there is no safety benefit to be gained by placing the unit in a shutdown condition. The Required Action of LCO 3.7.5 of “Suspend movement of irradiated fuel assemblies in the spent fuel 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 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

BASES

LCO 3.0.4 (continued)

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 provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that results from any unit shutdown.

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

LCO 3.0.4 is only applicable when entering MODE 4 from MODE 5, MODE 3 from MODE 4 or 5, MODE 2 from MODE 3 or 4 or 5, or MODE 1 from MODE 2. Furthermore, LCO 3.0.4 is applicable when entering any other specified condition in the Applicability only while operating in MODE 1, 2, 3, or 4. The requirements of LCO 3.0.4 do not apply in MODES 5 and 6, or 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.

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

BASES

LCO 3.0.4 (continued)

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

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

b. The OPERABILITY of other equipment.

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

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

An example of demonstrating the OPERABILITY of other equipment is taking an inoperable channel or trip system out of the tripped condition to prevent the trip function from occurring during the performance of 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

BASES

LCO 3.0.6 (continued)

to ensure the unit is maintained in a safe condition are specified in the support system LCO's Required Actions. These Required Actions may include entering the supported system's Conditions and Required Actions or may specify other Required Actions.

When a support system is inoperable and there is an LCO specified for it in the TS, the supported system(s) are required to be declared inoperable if determined to be inoperable as a result of the support system inoperability. However it is not necessary to enter into the supported systems' Conditions and Required Actions unless directed to do so by the support system's Required Actions. The potential confusion and inconsistency of requirements related to the entry into multiple support and supported systems' LCOs' Conditions and Required Actions are eliminated by providing all the actions that are necessary to ensure the unit is maintained in a safe condition in the support system's Required Actions.

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

Specification 5.5.7, "Safety Function Determination Program (SFDP)," ensures loss of safety function is detected and appropriate actions are taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other limitations, remedial actions, or compensatory actions may be identified as a result of the support system inoperability and corresponding exception to entering supported system Conditions and Required Actions. The SFDP implements the requirements of LCO 3.0.6.

Cross train checks to identify a loss of safety function for those support systems that support multiple and redundant safety systems are required. The cross train check verifies that the supported systems of the redundant OPERABLE support system are OPERABLE, thereby ensuring safety function is retained. If this evaluation determines that a loss of

BASES

LCO 3.0.6 (continued)

safety function exists, the appropriate Conditions and Required Actions of the LCO in which the loss of safety functions exists are required to be entered.

This loss of safety function does not require the assumption of additional single failures or loss of offsite power. Since operations are being restricted in accordance with the ACTIONS of the support system, any resulting temporary loss of redundancy or single failure protection is taken into account.

When loss of safety function is determined to exist, and the SFDP requires entry into the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists, consideration must be given to the specific type of function affected. Where a loss of function is solely due to a single Technical Specification support system (e.g., loss of automatic start due to inoperable instrumentation, or loss of pump suction source due to low tank level) the appropriate LCO is the LCO for the support system. The ACTIONS for a support system LCO adequately addresses the inoperabilities of that system without reliance on entering its supported system LCO. When the loss of function is the result of multiple support systems, the appropriate LCO is the LCO for the support system.

LCO 3.0.7

There are certain special tests and operations required to be performed at various times over the life of the unit. These special tests and operations are necessary to demonstrate select unit performance characteristics, to perform special maintenance activities, and to perform special evolutions. Test Exception LCO 3.1.8 allows specified Technical Specification (TS) requirements to be changed to permit performance 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

BASES

LCO 3.0.7 (continued)

desired to perform the special operation under the provisions of the Test Exception LCO, the requirements of the Test Exception LCO shall be followed.

LCO 3.0.8

LCO 3.0.8 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.

This Specification delineates the requirements 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 3.0.8, 1 hour is allowed to prepare for an orderly plan of action which optimizes plant safety and equipment restoration. The Shutdown Safety Status Trees provide a systematic method to explicitly determine the status of the plant during shutdown conditions, after entering MODE 5. A set of plant parameters is monitored and if any parameter is outside of its defined limits, a transition is made to the Shutdown Emergency Response Guidelines. These guidelines provide preplanned actions for addressing parameters outside defined limits.

Examples of the required end states specified for inoperable passive systems while in MODES 5 and 6 are provided in Table B 3.0-1, Passive Systems Shutdown MODE Matrix. These requirements are specified in the individual Specifications. The required end states specified for passive systems, when the unit is in MODE 5 or 6, are selected to ensure that the initial conditions and system and equipment availabilities minimize the likelihood and consequences of potential shutdown events.

BASES

LCO 3.0.8 (continued)

ACTIONS required in accordance with LCO 3.0.8 may be terminated and LCO 3.0.8 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 3.0.8 is exited.

In MODES 5 and 6, LCO 3.0.8 provides actions for Conditions not covered in other Specifications and for multiple concurrent Conditions for which conflicting actions are specified.

As an example of the application of LCO 3.0.8, see column 2 of Table B 3.0-1, Passive Systems Shutdown MODE Matrix, for the core makeup tank. This example assumes that the plant is initially in MODE 5 with the RCS pressure boundary intact. In this plant condition, LCO 3.5.3 requires one core makeup tank to be OPERABLE. The table shows the required end state established by the Required Actions of TS 3.5.3 in the event that the core makeup tank cannot be restored to OPERABLE status.

For this initial plant shutdown condition with no OPERABLE core makeup tanks, four conditions are identified in TS 3.5.3, with associated Required Actions and Completion Times. If Conditions A, B, and C cannot be completed within the required Completion Times, then Condition D requires immediately initiating action to place the plant in MODE 5 with the RCS pressure boundary open, and with pressurizer level greater than 20 percent.

LCO 3.0.8 would apply if actions could not immediately be initiated to open the RCS pressure boundary. In this situation, in parallel with the TS 3.5.3 actions to continue to open the RCS pressure boundary, LCO 3.0.8 requires the operators to take actions to restore one core makeup tank to OPERABLE status, and to monitor the Safety System Shutdown Monitoring Trees.

The Shutdown Status Trees monitor seven key RCS parameters and direct the operators to one of six shutdown ERGs in the event that any of the parameters are outside of allowable limits. The shutdown ERGs

BASES

LCO 3.0.8 (continued)

identify actions to be taken by the operators to satisfy the critical safety functions for the plant in the shutdown condition, using plant equipment available in this shutdown condition. LCO 3.0.8 monitoring would continue to be required until one core makeup tank is restored to OPERABLE status or the Required Actions for Condition D can be satisfied. In this case, once the RCS pressure boundary is open as required by Condition D, LCO 3.0.8 would be exited.

Table B 3.0-1 (page 1 of 1)
Passive Systems Shutdown MODE Matrix

LCO Applicability	Automatic Depressurization System	Core Makeup Tank	Passive RHR	IRWST	Containment	Containment Cooling[1]
MODE 5 RCS pressure boundary intact	9 of 10 paths OPERABLE All paths closed LCO 3.4.12	One CMT OPERABLE LCO 3.5.3	System OPERABLE LCO 3.5.5	One injection flow path and one recirculation sump flow path OPERABLE LCO 3.5.7	Closure capability LCO 3.6.8	Three water flow paths OPERABLE LCO 3.6.7
Required End State	MODE 5 RCS pressure boundary open, ≥ 20% pressurizer level	MODE 5 RCS pressure boundary open, ≥ 20% pressurizer level	MODE 5 RCS pressure boundary open, ≥ 20% pressurizer level	MODE 5 RCS pressure boundary intact, ≥ 20% pressurizer level	MODE 5 RCS pressure boundary intact, ≥ 20% pressurizer level	MODE 5 RCS pressure boundary intact, ≥ 20% pressurizer level
MODE 5 RCS pressure boundary open or pressurizer level < 20%	Stages 1, 2, and 3 open 2 stage 4 valves OPERABLE LCO 3.4.13	None	None	One injection flow path and one recirculation sump flow path OPERABLE LCO 3.5.7	Closure capability LCO 3.6.8	Three water flow paths OPERABLE LCO 3.6.7
Required End State	MODE 5 RCS pressure boundary open, ≥ 20% pressurizer level			MODE 5 RCS pressure boundary intact, ≥ 20% pressurizer level	MODE 5 RCS pressure boundary intact, ≥ 20% pressurizer level	MODE 5 RCS pressure boundary intact, ≥ 20% pressurizer level
MODE 6 Upper internals in place	Stages 1, 2, and 3 open 2 stage 4 valves OPERABLE LCO 3.4.13	None	None	One injection flow path and one recirculation sump flow path OPERABLE LCO 3.5.8	Closure capability LCO 3.6.8	Three water flow paths OPERABLE LCO 3.6.7
Required End State	MODE 6 Upper internals removed			MODE 6 Refueling cavity full	MODE 6 Refueling cavity full	MODE 6 Refueling cavity full
MODE 6 Upper internals removed	None	None	None	One injection flow path and one recirc-ulation sump flow path OPERABLE LCO 3.5.8	Closure capability LCO 3.6.8	Three water flow paths OPERABLE LCO 3.6.7
Required End State				MODE 6 Refueling cavity full	MODE 6 Refueling cavity full	MODE 6 Refueling cavity full

(1) Containment cooling via PCS is not required when core decay heat < 9 MWt.

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 ensures that Surveillances are performed to verify the OPERABILITY of systems and components, and that variables are within specified limits. Failure to meet a Surveillance within the specified Frequency, in accordance with SR 3.0.2, constitutes a failure to meet an LCO.

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

a. The systems or components are known to be inoperable, although still meeting the SRs; or

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

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

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

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

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 Actions with a Completion Time that requires the periodic performance of the Required Action on a "once per…" interval.

SR 3.0.2 permits a 25% extension of the interval specified in the Frequency. This extension facilitates Surveillance scheduling and considers plant operating conditions that may not be suitable for conducting the Surveillance (e.g., transient conditions or other ongoing Surveillance or maintenance activities).

The 25% extension does not significantly degrade the reliability that results from performing the Surveillance at its specified Frequency. This is based on the recognition that the most probable result of any particular surveillance being performed is the verification of conformance with the SRs. The exceptions to SR 3.0.2 are those Surveillances for which the 25% extension of the interval specified in the Frequency does not apply. These exceptions are stated in the individual Specifications. The requirements of regulations take precedence over the TS. An example of where SR 3.0.2 does not apply is in the Containment Leakage Rate Testing Program. This program establishes testing requirements and Frequencies in accordance with the requirements of regulations. The TS cannot in and of themselves extend a test interval specified in the regulations.

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 remedial action, is

BASES

SR 3.0.2 (continued)

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 greater, applies from the point in time that it is discovered that the Surveillance has not been performed, in accordance with SR 3.0.2, and not at the time that the specified Frequency was not met.

This delay period provides adequate time to complete Surveillances that have been missed. This delay period permits the completion of a Surveillance before compliance with Required Actions or other remedial measures that might preclude completion of the Surveillance.

The basis for this delay period includes consideration of unit Conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, the safety significance of the delay in completing the required Surveillance, and the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the requirements. When a Surveillance with a Frequency based not on time intervals, but upon specified unit Conditions or operational situations, or requirements of regulations (e.g., prior to entering MODE 1 after each fuel loading, or in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions, etc.) is discovered to not have been performed when specified, SR 3.0.3 allows for the full delay period of up to the specified Frequency to perform the Surveillance. However, since there is not a time interval specified, the missed Surveillance should be performed at the first reasonable opportunity. SR 3.0.3 provides a time limit for, and allowances for the performance of, Surveillances that become applicable as a consequence of MODE changes imposed by Required Actions.

BASES

SR 3.0.3 (continued)

Failure to comply with specified Frequencies for SRs is expected to be an infrequent occurrence. Use of the delay period established by SR 3.0.3 is a flexibility which is not intended to be used as an operational convenience to extend Surveillance intervals. While up to 24 hours or the limit of the specified Frequency is provided to perform the missed Surveillance, it is expected that the missed Surveillance will be performed at the first reasonable opportunity. The determination of the first reasonable opportunity should include consideration of the impact on plant risk (from delaying the Surveillance as well as any plant configuration changes required or shutting the plant down to perform the Surveillance) and impact on any analysis assumptions, in addition to unit conditions, planning, availability of personnel, and the time required to perform the Surveillance. This risk impact should be managed through the program in place to implement 10 CFR 50.65(a)(4) and its implementation guidance, NRC Regulatory Guide 1.182, 'Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants.' This Regulatory Guide addresses consideration of temporary and aggregate risk impacts, determination of risk management action thresholds, and risk management action up to and including plant shutdown. The missed Surveillance should be treated as an emergent condition as discussed in the Regulatory Guide. The risk evaluation may use quantitative, qualitative, or blended methods. The degree of depth and rigor of the evaluation should be commensurate with the importance of the component. Missed Surveillances for important components should be analyzed quantitatively. If the results of the risk evaluation determine the risk increase is significant, this evaluation should be used to determine the safest course of action. All missed Surveillances will be placed in the licensee's Corrective Action Program.

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 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 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.

BASES

SR 3.0.4

SR 3.0.4 establishes the requirement that all applicable SRs must be met before entry into a MODE or other specified condition in the Applicability.

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

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

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

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

BASES

SR 3.0.4 (continued)

required (to be met or performed) until a particular event, condition, or time has been reached. Further discussion of the specific formats of SR's 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 4, 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 MODE 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 MODE 1, 2, 3, or 4) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.1 SHUTDOWN MARGIN (SDM)

BASES

BACKGROUND

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

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

The system design requires that two independent reactivity control systems be provided, and that one of these systems be capable of maintaining the core subcritical under cold conditions. These requirements are provided by the use of movable control assemblies and soluble boric acid in the Reactor Coolant System (RCS). The Plant Control System (PLS) 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 PLS, together with the boration system, provides the SDM during power operation and is capable of making the core subcritical rapidly enough to prevent exceeding acceptable fuel damage limits, assuming that the rod of highest reactivity worth remains fully withdrawn. The soluble boron system can compensate for fuel depletion during operation and xenon burnout reactivity changes and maintain the reactor subcritical under cold conditions.

During power operation, SDM is calculated and monitored by the OPDMS and controlled by operating with RCCAs sufficiently withdrawn to meet the SDM requirement. When the OPDMS is inoperable, SDM control is ensured by operating within the limits of LCO 3.1.5 “Shutdown Bank Insertion Limits,” and LCO 3.1.6, “Control Bank Insertion Limits.” When the unit is in the shutdown and refueling modes, the SDM requirements are met by adjustments to the RCS boron concentration.

BASES

APPLICABLE SAFETY ANALYSES

The minimum required SDM is assumed as an initial condition in safety analyses. The safety analyses (Ref. 2) establish an SDM that ensures that specified acceptable fuel design limits are not exceeded for normal operation and AOOs, with the assumption of the highest worth rod stuck out on scram. For MODE 5, the primary safety analysis that relies on the SDM limits is the boron dilution analysis.

The acceptance criteria for the SDM requirements are that specified acceptable fuel design limits are maintained. This is done by ensuring that:

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

b. The reactivity transients associated with postulated accident conditions are controllable within acceptable limits (departures 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 accidents for the SDM requirements are based on a main steam line break (SLB) and inadvertent opening of a steam generator (SG) relief or safety valve, as described in the accident analyses (Ref. 2). The increased steam flow in the main steam system causes an increased energy removal from the affected SG, and consequently the RCS. This results in a reduction of the reactor coolant temperature. The resultant coolant shrinkage causes a reduction in pressure. In the presence of a negative moderator temperature coefficient (MTC), this cooldown causes an increase in core reactivity. The positive reactivity addition from the moderator temperature decrease will terminate when the affected SG boils dry, thus terminating RCS heat removal and cooldown. Following the SLB or opening of an SG relief or safety valve, a post trip return to power may occur; however, no fuel damage occurs as a result of the post trip return to power, and the THERMAL POWER does not violate the Safety Limit (SL) requirement of SL 2.1.1.

In addition to the limiting SLB and inadvertent opening of an SG relief or safety valve transients, the SDM requirement must also protect against:

a. Inadvertent boron dilution;

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

BASES

APPLICABLE SAFETY ANALYSES (continued)

c. Rod ejection;

d. Inadvertent operation of Passive Residual Heat Removal Heat Exchanger (PRHR HX).

Each of these events is discussed below.

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

The uncontrolled rod withdrawal transient is terminated by a high neutron flux trip. Power level, RCS pressure, linear heat rate, and the DNBR do not exceed allowable limits.

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

The inadvertent actuation of the PRHR HX causes an RCS temperature reduction from an initial injection of relatively cold water and the continued cooling of the RCS by PRHR. In the presence of a negative moderator temperature coefficient, the RCS temperature reduction causes an increase in core reactivity. Safety injection on the low cold leg temperature or low pressurizer pressure signals actuate the core makeup tank (CMT) and bring the plant to a stable condition.

SDM satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii). Even though it is not directly observed from the main control room, SDM is considered an initial condition process variable because it is periodically monitored to provide assurance that the unit is operating within the bounds of accident analysis assumptions.

LCO

SDM is a core design condition that can be ensured during operation through calculations by the Online Power Distribution Monitoring System (OPDMS) and RCCA positioning and through the soluble boron concentration.

BASES

LCO (continued)

The SLB and the boron dilution accidents (Ref. 2) are the most limiting analyses that establish the SDM value of the LCO. For SLB accidents, if the LCO is violated, there is a potential to exceed the DNBR limit and to exceed 10 CFR 50.34 limits (Ref. 3). For the boron dilution accident, if the LCO is violated, the minimum required time assumed for automatic action to terminate dilution may no longer be applicable.

APPLICABILITY

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

ACTIONS

A.1

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

In the determination of the required combination of boration flow rate and boron concentration, there is no unique 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 concentrated solution. The operator should begin boration 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 hot shutdown conditions when boron concentration is highest at 1502 ppm. Assuming that a value of 1.0% $\Delta k/k$ must be recovered and the boration flow rate is 100 gpm, it is possible to increase the boron concentration of the RCS by 111 ppm in approximately 21 minutes utilizing boric acid solution having a concentration of 4375 ppm. If a boron worth of 9 pcm/ppm is assumed, this combination of parameters will increase the SDM by 1.0% $\Delta k/k$. These boration parameters of 100 gpm and 4375 ppm represent typical values and are provided for the purpose of offering a specific example.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1

In MODES 1 and 2, SDM is verified by observing that the requirements of LCO 3.1.5 and LCO 3.1.6 are met. In the event that an RCCA is known to be untrippable, however, SDM verification must account for the worth of both the untrippable RCCA as well as another RCCA of maximum worth.

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

a. RCS boron concentration;

b. RCCA and GRCA 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 the low probability of an accident occurring without the required SDM. This allows time for the operator to collect the required data, which includes performing a boron concentration analysis, and complete the calculation.

REFERENCES

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

2. Chapter 15, “Accident Analysis.”

3. 10 CFR 50.34.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.2 Core Reactivity

BASES

BACKGROUND

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

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

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

BASES

BACKGROUND (continued)

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 compensate reactivity and maintain constant THERMAL POWER. The boron letdown curve is based on steady state operation at RTP. Therefore, deviations from the predicted boron letdown curve may indicate deficiencies in the design analysis, deficiencies in the calculational models, or abnormal core conditions, and must be evaluated.

APPLICABLE SAFETY ANALYSES

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

Accurate prediction of core reactivity is either an explicit or implicit assumption in the accident analysis evaluations. Certain accident evaluations (Ref. 2) are, 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 sensitive to accurate predictions of core reactivity. These accident analysis evaluations rely on computer codes that have been qualified against available test data, operating plant data, and analytical benchmarks. Monitoring reactivity balance provides additional assurance that the nuclear methods provide an accurate representation of the core reactivity.

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

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

BASES

APPLICABLE SAFETY ANALYSES (continued)

curve that develop 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.

Core reactivity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

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 DBA and transient analyses are no longer valid, or that the uncertainties in the Nuclear Design Methodology are larger than expected. A limit on the reactivity balance of ± 1% $\Delta k/k$ has been established based on engineering judgment. A 1% deviation in reactivity from that predicted is larger than expected for normal operation and should therefore be evaluated.

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

APPLICABILITY

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

BASES

APPLICABILITY (continued)

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

ACTIONS

A.1 and A.2

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

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

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

BASES

ACTIONS (continued)

B.1

If the core reactivity cannot be restored to within the 1% Δk/k limit, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours. If the SDM for MODE 3 is not met, then the boration required by SR 3.1.1.1 would occur. The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1

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

REFERENCES

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

2. Chapter 15, “Accident Analysis.”

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 non-positive MTC over the range of fuel cycle operation. Therefore, a coolant temperature increase will cause a reactivity decrease, so that the coolant temperature tends to return toward its initial value. Reactivity increases that cause a coolant temperature increase will thus be self limiting, and stable power operation will result.

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

The limitations on MTC are provided to ensure that the value of this coefficient remains within the limiting conditions assumed in the Chapter 15 accident and transient analyses (Ref. 2).

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

BASES

BACKGROUND (continued)

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

APPLICABLE SAFETY ANALYSES

The acceptance criteria for the specified MTC are:

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

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

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

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

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

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

BASES

APPLICABLE SAFETY ANALYSES (continued)

MTC satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii). Even though it is not directly observed and controlled from the control room, MTC is considered an initial condition process variable because of its dependence on boron concentration.

LCO

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

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

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

The BOC limit and the EOC limit are established in the COLR to allow specifying limits for each particular cycle. This permits the unit to take advantage of improved fuel management and changes in unit operating schedule.

APPLICABILITY

Technical Specifications place both LCO and SR values on MTC, based on the safety analysis assumptions described above.

In MODE 1, the limits on MTC must be maintained to assure that any accident initiated from THERMAL POWER operation will not violate the design assumptions of the accident analysis. In MODE 2, with the reactor critical, the upper limit must also be maintained to ensure that startup and subcritical accidents (such as the uncontrolled CONTROL ROD assembly or group withdrawal) will not violate the assumptions of the accident analysis. The lower MTC limit must be maintained in MODES 2 and 3, in addition to MODE 1, to ensure that cooldown accidents will not violate the assumptions of the accident analysis. In

BASES

APPLICABILITY (continued)

MODES 4, 5, and 6, this LCO is not applicable, since no Design Basis Accidents (DBAs) using the MTC as an analysis assumption are initiated from these MODES.

ACTIONS

A.1

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

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

B.1

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

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

C.1

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

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

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.3.1

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

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

SR 3.1.3.2

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

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

a. The SR is not required to be performed until 7 effective full power days (EFPDs) after reaching the equivalent of an equilibrium RTP all rods out (ARO) boron concentration of 300 ppm.

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

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

BASES

REFERENCES

1. 10 CFR 50, Appendix A, GDC 11.
2. Chapter 15, “Accident Analysis.”
3. WCAP 9273-NP-A, “Westinghouse Reload Safety Evaluation Methodology,” July 1985.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.4 Rod Group Alignment Limits

BASES

BACKGROUND

The OPERABILITY (e.g., trippability) of the RCCAs is an initial assumption in all safety analyses which assume rod insertion upon reactor trip. Maximum rod misalignment is an initial assumption in the safety analysis that directly affects core power distributions and assumptions of available SDM. Gray Rod Cluster Assemblies (GRCAs) are excluded from this LCO during the planned swap of the gray rod banks, with OPDMS operable. The swap of GRCA banks will be periodically necessary to prevent excessive burnup shadowing of fuel rods near the gray rod assemblies. The bank swap maneuver will purposefully misalign GRCAs from their bank for a short period of time. The exclusion from this LCO is acceptable due to SHUTDOWN MARGIN being calculated exclusive of GRCAs, the relative low worth of individual gray rod assemblies, the short time duration anticipated for the swap maneuver and with OPDMS operable, power peaking and xenon redistribution effects will be monitored and controlled.

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

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

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

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

BASES

BACKGROUND (continued)

The rod control assemblies are divided among control banks and shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control. A group consists of two or more rod control assemblies that are electrically paralleled to step simultaneously. A bank of rod control assemblies consists of two groups that are moved in a staggered fashion, but always within one step of each other. The AP1000 design has seven control banks and four shutdown banks.

The shutdown banks are maintained either in the fully inserted or fully withdrawn position. The control banks are part of the MSHIM (Mechanical Shim) Control System which utilizes two independently OPERABLE groups of control banks for control of reactivity and axial power distribution.

Certain control rods will be pre-selected for inclusion in the Rapid Power Reduction (RPR) system. The purpose of the RPR is to initiate a rapid decrease in the core power during load rejection transients.

Reactivity control is provided primarily by the M banks. The M Banks consist of several control banks operating with a fixed overlap. The bank worth and overlap are defined so as to minimize the impact on axial offset with control bank maneuvering and still retain the reactivity required to meet the desired load changes.

The axial power distribution control is provided by the AO Bank, a relatively high worth bank.

In order to avoid boron adjustment for load follow operation, gray rods are utilized.

There are 16 GRCAs in the AP1000, each composed of 24 rodlets mounted on a common RCCA spider. These have been subdivided into what has been termed as MA, MB, MC, and MD Banks with 4 GRCAs in each.

Each of the MA, MB, MC, and MD Banks has almost the same worth. The primary gray bank function is to provide additional reactivity during the transition periods. During base load operation, two of the gray banks may be fully inserted into the core. Each of the gray banks consists of a relatively low worth bank.

The MA, MB, MC, MD, M1 and M2 Banks function together with a single variable (i.e., criticality or temperature) driving these groups as if they are in one control group.

BASES

BACKGROUND (continued)

The control rods are arranged in a radially symmetric pattern so that control bank motion does not introduce radial asymmetries in the core power distributions.

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

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

The DRPI System provides a highly accurate indication of actual control rod position, at a lower precision than the step counters. This system is based on inductive analog signals from a series of coils spaced along a hollow tube. To increase the reliability of the system, the inductive coils are connected alternately to data system A or B. Thus, if one data system fails, the DRPI will go on half-accuracy. The DRPI System is capable of monitoring rod position within at least ± 12 steps with either full accuracy or half accuracy.

APPLICABLE SAFETY ANALYSES

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

a. There be no violations of:

   1. Specified acceptable fuel design limits, or

   2. Reactor Coolant System (RCS) pressure boundary integrity; and

b. The core remains subcritical after accident transients.

Two types of misalignment are distinguished. During movement of a control 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

BASES

APPLICABLE SAFETY ANALYSES (continued)

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.

Two types of analysis are performed in regard to static rod misalignment (Ref. 3). With control banks at or above their insertion limits, one type of analysis considers the case when any one rod is completely inserted into the core. The second type of analysis considers the case of a completely withdrawn single rod from a bank inserted to its insertion limit. Satisfying limits on departure from nucleate boiling ratio in both of these cases bounds the situation when a rod is misaligned from its group by 12 steps.

Another type of misalignment occurs if one RCCA fails to insert upon a reactor trip and remains stuck fully withdrawn. This condition is assumed in the evaluation to determine that the required SDM is met with the maximum worth RCCA also fully withdrawn (Ref. 3).

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

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

Shutdown and control rod OPERABILITY and alignment are directly related to power distributions and SDM, which are initial conditions assumed in safety analyses. Therefore they satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

BASES

LCO

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

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

Failure to meet the requirements of this LCO may produce unacceptable power peaking factors and linear heating rates (LHR), or unacceptable SDMs, which may constitute initial conditions inconsistent with the safety analysis.

The LCO is modified by a Note to relax the rod alignment limit on gray rods during GRCA swap operations. This operation which occurs frequently throughout the fuel cycle would normally violate the LCO.

APPLICABILITY

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

BASES

ACTIONS

A.1.1 and A.1.2

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

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

A.2

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

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

B.1

When a rod becomes misaligned, it can usually be moved and is still trippable. With the OPDMS OPERABLE adverse peaking factors resulting from the misalignment can be detected. If the rod can be realigned within the Completion Time of 8 hours adverse burnup shadowing in the location of the misaligned rod can be avoided. With the OPDMS inoperable xenon redistribution can potentially cause adverse peaking factors which may not be detected. However, if the rod can be realigned within the Completion Time of 1 hour, local xenon redistribution during this short interval will not be significant and operation may proceed without further restriction.

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

BASES

ACTIONS (continued)

B.2.1.1 and B.2.1.2

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

In many cases, realigning the remainder of the group to the misaligned rod may not be desirable. For example, realigning control bank M2 to a rod that is misaligned 15 steps from the top of the core could require insertion of the M1 bank to maintain overlap limits.

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

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

For continued operation with a misaligned rod, RTP must be reduced, SDM must periodically be verified within limits, hot channel factors ($F_Q(Z)$ and $F^N_{\Delta H}$) must be verified within limits, and the safety analyses must be re-evaluated to confirm continued operation is permissible. A note has been added indicating that Required Actions B.2.4 and B.2.5, F_Q and $F_{\Delta H}$ verification, are only required when the OPDMS is inoperable and therefore unavailable to continuously monitor the core power distribution.

Reduction of power to 75% of RTP ensures that local LHR increases due to a misaligned RCCA will not cause the core design criteria to be exceeded (Ref. 3). The Completion Time of 2 hours gives the operator sufficient time to accomplish an orderly power reduction without challenging the Protection and Safety Monitoring System.

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

Online monitoring of core power distribution by the OPDMS, or verifying that $F_Q(Z)$ and $F^N_{\Delta H}$ are within the required limits when the OPDMS is inoperable, ensures that current operation at 75% of RTP with a rod misaligned is not resulting in power distributions which may invalidate safety analysis assumptions at full power. The Completion Time of

BASES

ACTIONS (continued)

72 hours allows sufficient time to restore OPDMS operable or to obtain and analyze offline flux maps of the core power distribution using the incore detector system and to calculate $F_Q(Z)$ and $F^N_{\Delta H}$.

Once current conditions have been verified acceptable, time is available to perform evaluations of accident analysis to determine that core limits will not be exceeded during a Design Basis Accident (DBA) for the duration of operation under these conditions. The accident analyses presented in Chapter 15 (Ref. 3) that may be adversely affected will be evaluated to ensure that the analysis results remain valid for the duration under these conditions. A Completion Time of 5 days is sufficient time to obtain the required input data and to perform the analysis.

C.1

When Required Actions cannot be completed within their Completion Times, the unit must be brought to a MODE or Condition in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours, which obviates concerns about the development of undesirable xenon or power distributions. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching MODE 3 from full power condition in an orderly manner and without challenging the plant systems.

D.1.1 and D.1.2

More than one control rod becoming misaligned from its group average position is not expected, and has the potential to reduce SDM. Therefore, SDM must be evaluated. One hour allows the operator adequate time to determine SDM.

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

BASES

ACTIONS (continued)

D.2

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

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

SURVEILLANCE REQUIREMENTS

SR 3.1.4.1

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

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 would result in radial or axial power tilts, or oscillations. 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 10 steps will not cause radial or axial power tilts, or oscillations, to occur. The 92 day Frequency takes into consideration other information available to the operator in the control room and SR 3.1.4.1, which is performed more frequently and adds to the determination of OPERABILITY of the rods. Between 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 remains 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. GRCA are excluded from this Surveillance because they are not considered in the calculation of SDM in MODES 1 and 2.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.4.3

Verification of rod drop times allows the operator to determine that the maximum rod drop time permitted is consistent with the assumed rod drop time used in the safety analysis. Measuring rod drop times prior to reactor criticality, after each reactor vessel head removal, ensures that the reactor internals and rod drive mechanism will not interfere with rod motion or rod drop time, and that no degradation in these systems has occurred that would adversely affect control rod motion or drop time. This testing is performed with all RCPs operating and the average moderator temperature ≥ 500°F to simulate a reactor trip under conservative conditions. GRCA are excluded from this Surveillance because they are not considered in the calculation of SDM in MODES 1 and 2.

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

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10 and GDC 26.
2. 10 CFR 50.46.
3. Chapter 15, “Accident Analysis.”

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.5 Shutdown Bank Insertion Limits

BASES

BACKGROUND

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

The applicable criteria for these reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10, “Reactor Design,” GDC 26, “Reactivity Control System Redundancy and Protection,” 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 control rod insertion have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

The rod cluster control assemblies (RCCAs) are divided among control banks and shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control. A group consists of two or more RCCAs that are electrically paralleled to step simultaneously. A bank of RCCAs consists of two groups that are moved in a staggered fashion, but always within one step of each other. The AP1000 design has seven control banks and four shutdown banks. See LCO 3.1.4, “Rod Group Alignment Limits,” for control and shutdown rod OPERABILITY and alignment requirements, and LCO 3.1.7, “Rod Position Indication,” for position indication requirements.

The control banks are used for precise reactivity control of the reactor. The positions of the control banks are normally automatically controlled by the Plant Control System (PLS), but they can also be manually controlled. They are capable of adding negative reactivity very quickly (compared to borating). The control banks must be maintained above designed insertion limits and are typically near the fully withdrawn position during normal full power operations. Hence, they are not capable of adding a large amount of positive reactivity. Boration or dilution of the Reactor Coolant System (RCS) compensates for the reactivity changes associated with large changes in RCS temperature. The design calculations are performed with the assumption that the shutdown banks are withdrawn first. The shutdown banks can be fully withdrawn without the core going critical. This provides available negative reactivity in the event of boration errors. The shutdown banks

BASES

BACKGROUND (continued)

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

APPLICABLE SAFETY ANALYSES

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

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

a. There be no violations of:

   1. specified acceptable fuel design limits, or,

   2. RCS pressure boundary integrity; and

b. The core remains subcritical after accident transients.

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

The shutdown bank insertion limits preserve an initial condition assumed in the safety analyses and satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

BASES

LCO

The shutdown banks must be within their insertion limits any time the reactor is critical or approaching criticality. This in conjunction with LCO 3.1.6, “Control Bank Insertion Limits,” ensures that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM following a reactor trip.

The shutdown bank insertion limits are defined in the COLR.

APPLICABILITY

The shutdown banks must be within their insertion limits with the reactor in MODE 1 and MODE 2. The LCO is not applicable if OPDMS is OPERABLE since OPDMS ensures that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM following a reactor trip. The shutdown banks do not have to be within their insertion limits in MODE 3, unless an approach to criticality is being made. In MODE 3, 4, 5, or 6 the shutdown banks are fully inserted in the Core and contribute to the SDM. 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.

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

ACTIONS

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

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

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

BASES

ACTIONS (continued)

B.1

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

SURVEILLANCE REQUIREMENTS

SR 3.1.5.1

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

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

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10, GDC 26, and GDC 28.
2. 10 CFR 50.46.
3. Chapter 15, “Accident Analysis.”

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.6 Control Bank Insertion Limits

BASES

BACKGROUND

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

The applicable criteria for these reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10, “Reactor Design,” GDC 26, “Reactivity Control System Redundancy and Protection,” 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 control rod insertion have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

The rod cluster control assemblies (RCCAs) are divided among control banks and shutdown banks, gray rod cluster assemblies (GRCAs) are limited to control banks. Each bank may be further subdivided into two groups to provide for precise reactivity control. A group consists of two or more RCCAs or GRCAs that are electrically paralleled to step simultaneously. A bank of RCCAs consists of two groups that are moved in a staggered fashion, but always within 1 step of each other. The AP1000 design has seven control banks and four shutdown banks. See LCO 3.1.4, “Rod Group Alignment Limits,” for control and shutdown rod OPERABILITY and alignment requirements, and LCO 3.1.7, “Rod Position Indication,” for position indication requirements.

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

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

The power density at any point in the core must be limited so that the fuel design criteria are maintained. Together, LCO 3.1.4, “Rod Group Alignment Limits,” LCO 3.1.5, “Shutdown Bank Insertion Limits,”

BASES

BACKGROUND (continued)

LCO 3.1.6, “Control Bank Insertion Limits,” and LCO 3.2.5, “OPDMS – Monitored Powered Distribution Parameters,” when the OPDMS is OPERABLE, or LCO 3.2.3, “AXIAL FLUX DIFFERENCE (AFD),” and LCO 3.2.4, “QUADRANT POWER TILT RATIO (QPTR),” when the OPDMS is inoperable, provide limits on control component operation and on monitored process variables which ensure that the core operates within the fuel design criteria.

The shutdown and control bank insertion and alignment limits and power distribution limits are process variables that together characterize and control the three dimensional power distribution of the reactor core. Additionally, the control bank insertion limits control the reactivity that could be added in the event of a rod ejection accident, and the shutdown and control bank insertion limits assure the required SDM is maintained.

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

APPLICABLE SAFETY ANALYSES

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

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

a. There be no violations of:

   1. specified fuel design limits, or

   2. Reactor Coolant System (RCS) pressure boundary integrity; and

b. The core remains subcritical after accident transients.

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

BASES

APPLICABLE SAFETY ANALYSES (continued)

The SDM requirement is ensured by limiting the control and shutdown bank insertion limits so that allowable inserted worth of the RCCAs is such that sufficient reactivity is available in the rods to shut down the reactor to hot zero power with a reactivity margin which assumes the maximum worth RCCA remains fully withdrawn upon trip (Ref. 3).

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

The control and shutdown bank insertion limits ensure that safety analyses assumptions for SDM, ejected rod worth, and power distribution peaking factors are preserved (Ref. 3).

The insertion limits satisfy Criterion 2 of 10 CFR 50.36©(2)(ii) in that they are initial conditions assumed in the safety analysis.

LCO

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

APPLICABILITY

The control bank sequence, overlap, and physical insertion limits shall be maintained with the reactor in MODES 1 and 2 with $k_{eff} \geq 1.0$. The LCO is not applicable if OPDMS is OPERABLE since OPDMS ensures that the limits are met. These limits must be maintained since they preserve the assumed power distribution, ejected rod worth, SDM, and reactivity rate insertion assumptions.

Applicability in MODES 3, 4, and 5 is not required, since neither the power distribution nor ejected rod worth assumptions would be exceeded in these MODES.

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

BASES

ACTIONS

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

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

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

b. Moving rods to be consistent with power.

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

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

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

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

C.1

If Required Actions A.1 and A.2, or B.1 and B.2 cannot be completed within the associated Completion Times, the plant must be brought to MODE 2 with $k_{eff} < 1.0$, 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 condition in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1

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

BASES

SURVEILLANCE REQUIREMENTS (continued)

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

SR 3.1.6.2

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

SR 3.1.6.3

When control banks are maintained within their insertion limits as checked by SR 3.1.6.2 above, it is unlikely that their sequence and overlap will not be in accordance with requirements provided in the COLR. A Frequency of 12 hours is consistent with the insertion limit check above in SR 3.1.6.2.

REFERENCES

1. 10CFR50, Appendix A, GDC 10, GDC 26, and GDC 28.
2. 10CFR50.46.
3. Chapter 15, “Accident Analysis.”

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.7 Rod Position Indication

BASES

BACKGROUND

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

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

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

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

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

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

BASES

BACKGROUND (continued)

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

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

APPLICABLE SAFETY ANALYSES

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

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

BASES

LCO

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

a. The DRPI System indicates within 12 steps of the group step counter demand position as required by LCO 3.1.4, “Rod Group Alignment Limits”;

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

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

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

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

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

APPLICABILITY

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

ACTIONS

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

BASES

ACTIONS (continued)

A.1

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

A.2

Reduction of THERMAL POWER to ≤ 50% RTP puts the core into a condition where rod position is not significantly affecting core peaking factors (Ref. 2).

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

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

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

Monitoring and recording reactor coolant T_{avg} help assure that significant changes in power distribution and SDM are avoided. The once per hour Completion Time is acceptable because only minor fluctuations in RCS temperature are expected at steady state plant operating conditions.

The position of the rods may be determined indirectly by use of the incore detectors. The Required Action may also be satisfied by ensuring at least once per 8 hours that F_Q satisfies LCO 3.2.1, $F^N_{\Delta H}$ satisfies LCO 3.2.2, and SDM is within the limits provided in the COLR,

BASES

ACTIONS (continued)

provided the nonindicating rods have not been moved. Verification of control rod position once per 8 hours is adequate for allowing continued full power operation for a limited, 24 hour period, since the probability of simultaneously having a rod significantly out of position and an event sensitive to that rod position is small. The 24 hour Completion Time provides sufficient time to troubleshoot and restore the DRPI system to operation while avoiding the plant challenges associated with the shutdown without full rod position indication.

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

C.1 and C.2

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

If, within 4 hours, the rod positions have not been determined, THERMAL POWER must be reduced to ≤ 50% RTP within 8 hours to avoid undesirable power distributions that could result from continued operation at > 50% RTP, if one or more rods are misaligned by more than 24 steps. The allowed Completion Time of 4 hours provides an acceptable period of time to verify the rod positions.

D.1.1 and D.1.2

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

D.2

Reduction of THERMAL POWER to ≤ 50% RTP puts the core into a condition where rod position is not significantly affecting core peaking factor limits (Ref. 2). The allowed Completion Time of 8 hours provides an acceptable period of time to verify the rod positions per Required Actions D.1.1 and D.1.2 or reduce power to ≤ 50% RTP.

BASES

ACTIONS (continued)

E.1

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

SURVEILLANCE REQUIREMENTS

SR 3.1.7.1

Verification that the DRPI agrees with the demand position within 12 steps provides assurance that the DRPI is operating correctly. Since the DRPI does not display the actual shutdown rod positions between 18 and 249 steps, only points within the indicated ranges are compared.

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

REFERENCES

1. 10 CFR 50, Appendix A, GDC 13.

2. Chapter 15, “Accident Analysis.”

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.8 PHYSICS TESTS Exceptions – MODE 2

BASES

BACKGROUND

The primary purpose of the MODE 2 PHYSICS TESTS exceptions is to permit relaxations of existing LCOs to allow certain PHYSICS TESTS to be performed.

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

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

a. Ensure that the facility has been adequately designed;

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

c. Verify the assumptions used to predict unit response;

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

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

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

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

BASES

BACKGROUND (continued)

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

a. Critical Boron Concentration – Control Rods Withdrawn;

b. Control Rod Worth;

c. Isothermal Temperature Coefficient (ITC).

These tests are performed in MODE 2. These and other supplementary tests may be required to calibrate the nuclear instrumentation or to diagnose operational problems. These tests may cause the operating controls and process variables to deviate from their LCO requirements during their performance.

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

b. The Control Rod Worth Test is used to measure the reactivity worth of selected control banks. This test is performed at HZP and has four alternative methods of performance. The first method, the Boron Exchange Method, varies the reactor coolant boron concentration and moves the selected control bank in response to the changing boron concentration. The reactivity changes are measured with a reactivity computer. This sequence is repeated for the remaining control banks. The second method, the Rod Swap Method, measures the worth of a predetermined reference bank using the Boron Exchange Method above. The reference bank is then nearly fully inserted into the core. The selected bank is then inserted into the core as the reference bank is withdrawn. The HZP critical conditions are then determined with the selected bank fully inserted into the core. The worth of the selected bank is calculated based on the position of the reference bank with respect to the selected bank. This sequence is repeated as necessary for the remaining control banks. The third method, the Boron Endpoint Method, moves the selected control bank over its entire length of travel and while varying the reactor coolant boron concentration to maintain HZP criticality again. The difference in boron concentration is the worth of the selected control bank. This sequence is repeated for the remaining control banks. The fourth method, Dynamic Rod

BASES

BACKGROUND (continued)

Worth Measurement (DRWM), moves each bank, individually, into the core to determine its worth. The bank is dynamically inserted into the core while data is acquired from the excore channel. While the bank is being withdrawn, the data is analyzed to determine the worth of the bank. This is repeated for each control and shutdown bank. Performance of this test will violate LCO 3.1.4, “Rod Group Alignment Limits,” LCO 3.1.5, “Shutdown Bank Insertion Limit,” or LCO 3.1.6, “Control Bank Insertion Limits.”

c. The ITC Test measures the ITC of the reactor. This test is performed at HZP. The method is to vary the RCS temperature in a slow and continuous manner. The reactivity change is measured with a reactivity computer as a function of the temperature change. The ITC is the slope of the reactivity versus the temperature plot. The test is repeated by reversing the direction of the temperature change and the final ITC is the average of the two calculated ITCs. Performance of this test could violate LCO 3.4.2, “RCS Minimum Temperature for Criticality.”

APPLICABLE SAFETY ANALYSES

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

Chapter 14 defines requirements for initial testing of the facility, including low power PHYSICS TESTS. Sections 14.2.10.2 and 14.2.10.3 (Ref. 6) summarize the initial criticality and low power tests.

Requirements for reload fuel cycle PHYSICS TESTS are defined in ANSI/ANS-19.6.1-1985 (Ref. 4). Although these PHYSICS TESTS are generally accomplished within the limits for the LCOs, conditions may occur when one or more LCOs must be suspended to make completion of PHYSICS TESTS possible or practical. This is acceptable as long as the fuel design criteria are not violated. When one or more of the requirements specified in:

LCO 3.1.3 “Moderator Temperature Coefficient (MTC),”
LCO 3.1.4 “Rod Group Alignment Limits,”
LCO 3.1.5 “Shutdown Bank Insertion Limit,”

BASES

APPLICABLE SAFETY ANALYSES (continued)

LCO 3.1.6 "Control Bank Insertion Limits," and
LCO 3.4.2 "Minimum Temperature for Criticality,"

are suspended for PHYSICS TESTS, the fuel design criteria are preserved as long as the power level is limited to ≤ 5% RTP, the reactor coolant temperature is kept ≥ 541°F, and SDM is within the limits provided in the COLR.

PHYSICS TESTS include measurement of core nuclear parameters or the exercise of control components that affect process variables. Also involved are the movable control components (control and shutdown rods), which are required to shut down the reactor. The limits for these variables are specified for each fuel cycle in the COLR.

As described in LCO 3.0.7, compliance with Test Exception LCOs is optional, and therefore no criteria of 10 CFR 50.36(c)(2)(ii) apply. Test Exception LCOs provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs. A discussion of the criteria satisfied for the other LCOs is provided in their respective Bases.

Reference 7 allows special test exceptions (STE) to be included as part of the LCO that they affect. It was decided, however, to retain this STE as a separate LCO because it was less cumbersome and provided additional clarity.

LCO

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

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

a. RCS lowest loop average temperature is ≥ 541°F,

b. SDM is within the limits provided in the COLR, and

c. THERMAL POWER is < 5% RTP.

BASES

APPLICABILITY

This LCO is applicable when performing low power PHYSICS TESTS. The Applicability is stated as "During PHYSICS TESTS initiated in MODE 2" to ensure that the 5% RTP maximum power level is not exceeded. Should the THERMAL POWER EXCEED 5% RTP, and consequently the unit enter MODE 1, this Applicability statement prevents exiting this Specification and its Required Actions.

ACTIONS

A.1 and A.2

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

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

B.1

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

C.1

When the RCS lowest T_{avg} is < 541°F, the appropriate action is to restore T_{avg} to within its specified limit. The allowed Completion Time of 15 minutes provides time for restoring T_{avg} to within limits without allowing the plant to remain in an unacceptable condition for an extended period of time. Operation with the reactor critical and with temperature below 541°F could violate the assumptions for accidents analyzed in the safety analyses.

D.1

If the Required Action of Condition C cannot be completed within the associated Completion Time, the plant must be placed in a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within an additional 15 minutes. The Completion Time of 15 additional minutes is reasonable, based on operating experience, to reach MODE 3 from MODE 2 HZP conditions in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.8.1

The power range and intermediate range neutron detectors must be verified to be OPERABLE in MODE 2 by LCO 3.3.1 "Reactor Trip System (RTS) Instrumentation." A CHANNEL OPERATIONAL TEST is performed on each power range and intermediate range channel prior to initiation of the PHYSICS TESTS. This will ensure that the RTS is properly aligned to provide the required degree of core protection during the performance of the PHYSICS TESTS.

SR 3.1.8.2

Verification that the RCS lowest loop T_{avg} is ≥ 541°F will ensure that the unit is not operating in a condition that could invalidate the safety analyses. Verification of the RCS temperature at a Frequency of 30 minutes during the performance of the PHYSICS TESTS will provide assurance that the initial conditions of the safety analyses are not violated.

SR 3.1.8.3

Verification that the THERMAL POWER is < 5% RTP will ensure that the plant is not operating in a condition that could invalidate the safety analyses. Verification of the THERMAL POWER at a Frequency of 30 minutes during the performance of the PHYSICS TESTS will ensure that the initial conditions of the safety analyses are not violated.

SR 3.1.8.4

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

a. RCS boron concentration;

b. Control bank position;

c. RCS average temperature;

d. Fuel burnup based on gross thermal energy generation;

e. Xenon concentration;

f. Samarium concentration; and

g. Isothermal temperature coefficient (ITC).

BASES

SURVEILLANCE REQUIREMENTS (continued)

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

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

REFERENCES

1. 10 CFR 50, Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants.”

2. 10 CFR 50.59, “Changes, Tests and Experiments.”

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

4. ANSI/ANS-19.6.1-1997, “Reload Startup Physics Tests for Pressurized Water Reactors,” American National Standards Institute, August 22, 1997.

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

6. Chapter 14, “Initial Testing Program.”

7. WCAP-11618, including Addendum 1, April 1989.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.9 Chemical and Volume Control System (CVS) Demineralized Water Isolation Valves and Makeup Line Isolation Valves

BASES

BACKGROUND

One of the principle functions of the CVS system is to maintain the reactor coolant chemistry conditions by controlling the concentration of boron in the coolant for plant startups, normal dilution to compensate for fuel depletion, and shutdown boration. In the dilute mode of operation, unborated demineralized water may be supplied directly to the reactor coolant system.

Although the CVS is not considered a safety related system, certain functions of the system are considered safety related functions. The appropriate components have been classified and designed as safety related. The safety related functions provided by the CVS include containment isolation of chemical and volume control system lines penetrating containment, termination of inadvertent boron dilution, and preservation of the Reactor Coolant System (RCS) pressure boundary, including isolation of CVS letdown from the RCS.

APPLICABLE SAFETY ANALYSES

One of the initial assumptions in the analysis of an inadvertent boron dilution event (Ref. 1) is the assumption that the increase in core reactivity, created by the dilution event, can be detected by the source range instrumentation. The source range instrumentation will then supply a signal to the demineralized water isolation valves and the makeup line isolation valves in the CVS causing these valves to close and terminate the boron dilution event. Thus the makeup line isolation valves and the demineralized water isolation valves are components which function to mitigate or prevent an AOO.

CVS isolation valves satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The requirement that at least two demineralized water isolation valves and two makeup line isolation valves be OPERABLE assures that there will be redundant means available to terminate or prevent an inadvertent boron dilution event.

APPLICABILITY

The requirement that at least two demineralized water isolation valves and two makeup line isolation valves be OPERABLE is applicable in MODES 1, 2, 3, 4, and 5 because a boron dilution event is considered possible in these MODES, and the automatic closure of these valves is assumed in the safety analysis.

BASES

APPLICABILITY (continued)

In MODES 1 and 2, the detection and mitigation of a boron dilution event does not assume the detection of the event by the source range instrumentation. In these MODES, the event would be signalled by an intermediate range trip, a trip on the Power Range Neutron Flux - High (low setpoint nominally at 25% RTP), or Overtemperature delta T. The two demineralized water isolation valves close automatically upon reactor trip.

In MODE 6, a dilution event is precluded by the requirement in LCO 3.9.2 to close, lock and secure at least one valve in each unborated water source flow path.

ACTIONS

A.1

If only one of the demineralized water isolation valve and/or the makeup line isolation valve is/are OPERABLE, the redundant valve must be restored to OPERABLE status in 72 hours. The allowed Completion Time assures expeditious action will be taken, and is acceptable because the safety function of automatically isolating the clean water source can be accomplished by the redundant isolation valve(s).

B.1

If the Required Actions and associated Completion Time of Condition A are not met, or if both CVS demineralized water isolation valves or both makeup line isolation valves are not OPERABLE (i.e., not able to be closed automatically), then the demineralized water supply flow path to the RCS must be isolated. Isolation can be accomplished by manually isolating the CVS demineralized water isolation valve(s) or by positioning the 3-way blend valve to only take suction from the boric acid tank. Alternatively, the dilution path may be isolated by closing appropriate isolation valve(s) in the flow path(s) from the demineralized water storage tank to the reactor coolant system.

The Action is modified by a Note allowing the flow path to be unisolated intermittently under administrative controls. These administrative controls consist of stationing a dedicated operator at the valve controls, who is in continuous communication with the main control room. In this way, the flow path can be rapidly isolated when a need for isolation is indicated.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.9.1

Verification that the CVS demineralized water isolation valves and makeup line isolation valves are OPERABLE, by stroking each valve closed, demonstrates that the valves can perform their safety related function. The Frequency is in accordance with the Inservice Testing Program.

REFERENCES

1. Chapter 15, “Accident Analysis.”

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1 Heat Flux Hot Channel Factor ($F_Q(Z)$) (F_Q Methodology)

BASES

BACKGROUND

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

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

During power operation with the On-line Power Distribution Monitoring System (OPDMS) inoperable, the global power distribution is limited by LCO 3.2.3, “AXIAL FLUX DIFFERENCE (AFD),” and LCO 3.2.4, “QUADRANT POWER TILT RATIO (QPTR),” which are directly and continuously measured process variables. These LCOs along with LCO 3.1.6, “Control Bank Insertion Limits,” maintain the core limits on power distributions on a continuous basis.

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

With the OPDMS OPERABLE, peak kw/ft (Z) (which is proportional to $F_Q(Z)$) is measured continuously. With the OPDMS inoperable, $F_Q(Z)$ is measured periodically using the incore detector system. These measurements are generally taken with the core at or near steady state conditions.

With the measured three dimensional power distributions, it is possible to derive a measured value for $F_Q(Z)$ with the OPDMS inoperable. However, because this value represents a steady state condition, it does not include the variations in the value of $F_Q(Z)$ which are present during a nonequilibrium situation such as load following.

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

Core monitoring and control under non-equilibrium conditions and the OPDMS inoperable are accomplished by operating the core within the limits of the appropriate LCOs, including the limits on AFD, QPTR, and control rod insertion.

BASES

APPLICABLE SAFETY ANALYSES

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

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

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

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

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

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

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

$F_Q(Z)$ satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

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

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

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

where: CFQ is the $F_Q(Z)$ limit at RTP provided in the COLR,

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

BASES

LCO (continued)

The actual values of CFQ are given in the COLR; however, CFQ is normally a number on the order of 2.60. For the AP1000, the normalized $F_Q(Z)$ as a function of core height is 1.0.

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

An $F_Q^C(Z)$ evaluation requires obtaining an incore flux map in MODE 1. From the incore flux map results the measured value of $F_Q(Z)$, called $F_Q^M(Z)$ is obtained. Then,

$$F_Q^C(Z) = F_Q^M(Z) * F_Q^{MU}(Z)$$

where $F_Q^{MU}(Z)$ is a factor that accounts for fuel manufacturing tolerances and flux map measurement uncertainty. $F_Q^{MU}(Z)$ is provided in the COLR.

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

The expression for $F_Q^W(Z)$ is:

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

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

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

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

BASES

LCO (continued)

Violating the LCO limits for $F_Q(Z)$ may result in an unanalyzed condition while $F_Q(Z)$ is outside its specified limits.

APPLICABILITY

When the OPDMS is inoperable and core power distribution parameters cannot be continuously monitored, it is necessary to determine $F_Q(Z)$ on a periodic basis. Furthermore, the $F_Q(Z)$ limits must be maintained in MODE 1 to prevent core power distributions from exceeding the limits assumed in the safety analyses. Applicability in other MODES is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the reactor coolant to require a limit on the distribution of core power.

ACTIONS

A.1

Reducing THERMAL POWER by ≥ 1% of RTP for each 1% by which $F_Q^C(Z)$ exceeds its limit, maintains an acceptable absolute power density. $F_Q^C(Z)$ is $F_Q^M(Z)$ multiplied by a factor accounting for fuel manufacturing tolerances and flux map measurement uncertainties. $F_Q^M(Z)$ is the measured value of $F_Q(Z)$. The Completion Time of 15 minutes provides an acceptable time to reduce power in an orderly manner without allowing the plant to remain in an unacceptable condition for an extended period of time. The maximum allowable power level initially determined by Required Action A.1 may be affected by subsequent determinations of $F_Q^C(Z)$ and would require power reductions within 15 minutes of the $F_Q^C(Z)$ determination, if necessary to comply with the decreased maximum allowable power level. Decreases in $F_Q^C(Z)$ would allow increasing the maximum allowable power level and increasing power up to this revised limit.

A.2

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

BASES

ACTIONS (continued)

Neutron Flux – High trip setpoints initially determined by Required Action A.2 may be affected by subsequent determinations of $F_Q^C(Z)$ and would require Power Range Neutron Flux – High trip setpoint reductions within 8 hours of the $F_Q^C(Z)$ determination, if necessary to comply with the decreased maximum allowable Power Range Neutron Flux – High trip setpoints. Decreases in $F_Q^C(Z)$ would allow increasing the maximum allowable Power Range Neutron Flux – High trip setpoints.

A.3

Reduction in the Overpower ΔT Trip setpoints (value of K_4) by ≥ 1% for each 1% by which $F_Q^C(Z)$ exceeds its limit is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 72 hours is sufficient considering the small likelihood of a severe transient in this time period and the prompt reduction in THERMAL POWER in accordance with Required Action A.1. The maximum allowable Overpower ΔT trip setpoints initially determined by Required Action A.3 may be affected by subsequent determinations of $F_Q^C(Z)$ and would require Overpower ΔT trip setpoint reductions within 72 hours of the $F_Q^C(Z)$ determination, if necessary to comply with the decreased maximum allowable Overpower ΔT trip setpoints. Decreases in $F_Q^C(Z)$ would allow increasing the maximum allowable Overpower ΔT trip setpoints.

A.4

Verification that $F_Q^C(Z)$ has been restored to within its limit by performing SR 3.2.1.1 and SR 3.2.1.2 prior to increasing THERMAL POWER above the limit imposed by Required Action A.1, assures that core conditions during operation at higher power levels and future operation are consistent with safety analyses assumptions.

Condition A is modified by a Note that requires Required Action A.4 to be performed whenever the Condition is entered. This ensures that SR 3.2.1.1 and SR 3.2.1.2 will be performed prior to increasing THERMAL POWER above the limit of Required Action A.1, even when Condition A is exited prior to performing Required Action A.4. Performance of SR 3.2.1.1 and SR 3.2.1.2 are necessary to assure $F_Q(Z)$ is properly evaluated prior to increasing THERMAL POWER.

BASES

ACTIONS (continued)

B.1

If it is found that the maximum calculated value of $F_Q(Z)$ which can occur during normal maneuvers, $F_Q^W(Z)$, exceeds its specified limits, there exists a potential for $F_Q^C(Z)$ to become excessively high if a normal operational transient occurs. Reducing the AFD by ≥ 1% for each 1% by which $F_Q^W(Z)$ exceeds its limit within the allowed Completion Time of 4 hours restricts the axial flux distribution such that even if a transient occurred, core peaking factors would not be exceeded.

The implicit assumption is that if W(Z) values were recalculated (consistent with the reduced AFD limits), then $F_Q^C(Z)$ times the recalculated W(Z) values would meet the $F_Q(Z)$ limit. Note that complying with this action (of reducing AFD limits) may also result in a power reduction. Hence the need for B.2, B.3, and B.4.

B.2

A reduction of the Power Range Neutron Flux-High trip setpoints by ≥ 1% for each 1% by which the maximum allowable power is reduced, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 72 hours is sufficient considering the small likelihood of a severe transient in this time period and the preceding prompt reduction in THERMAL POWER as a result of reducing AFD limits in accordance with Required Action B.1.

B.3

Reduction in the Overpower ΔT trip setpoints value of K_4 by ≥ 1% for each 1% by which the maximum allowable power is reduced, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 72 hours is sufficient considering the small likelihood of a severe transient in this time period, and the preceding prompt reduction in THERMAL POWER as a result of reducing AFD limits in accordance with Required Action B.1.

BASES

ACTIONS (continued)

B.4

Verification that $F_Q^W(Z)$ has been restored to within its limit, by performing SR 3.2.1.1 and SR 3.2.1.2 prior to increasing THERMAL POWER above the maximum allowable power limit imposed by Required Action B.1 ensures that core conditions during operation at higher power levels and future operation are consistent with safety analyses assumptions.

Condition B is modified by a Note that requires Required Action B.4 to be performed whenever the Condition is entered. This ensures that SR 3.2.1.1 and SR 3.2.1.2 will be performed prior to increasing THERMAL POWER above the limit of Required Action B.1, even when Condition A is exited prior to performing Required Action B.4. Performance of SR 3.2.1.1 and SR 3.2.1.2 are necessary to assure $F_Q(Z)$ is properly evaluated prior to increasing THERMAL POWER.

C.1

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

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

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1 and SR 3.2.1.2 are modified by two Notes. The first note applies to the situation where the OPDMS is inoperable at the beginning of cycle startup. Note 1 applies during the first power ascension after a refueling. It states that THERMAL POWER may be increased until an equilibrium power level has been achieved at which a power distribution map can be obtained. This allowance is modified, however, by one of the Frequency conditions that requires verification that $F_Q^C(Z)$ and $F_Q^W(Z)$ are within their specified limits after a power rise of more than 10% of RTP over the THERMAL POWER at which they were last verified to be within specified limits. Because $F_Q^C(Z)$ and $F_Q^W(Z)$ could not have previously been measured in this reload core, there is a second Frequency condition, applicable only for reload cores, that requires determination of these parameters before exceeding 75% RTP. This ensures that some determination of $F_Q^C(Z)$ and $F_Q^W(Z)$ are made at a lower power level at

BASES

SURVEILLANCE REQUIREMENTS (continued)

which adequate margin is available before going to 100% RTP. Also, this Frequency condition, together with the Frequency condition requiring verification of $F_Q^C(Z)$ and $F_Q^W(Z)$ following a power increase of more than 10%, ensures that they are verified as soon as RTP (or any other level for extended operation) is achieved. In the absence of these Frequency conditions, it is possible to increase power to RTP and operate for 31 days without verification of $F_Q^C(Z)$ and $F_Q^W(Z)$. The Frequency condition is not intended to require verification of these parameters after every 10% increase in power level above the last verification. It only requires verification after a power level is achieved for extended operation that is 10% higher than that power at which $F_Q(Z)$ was last measured.

The second Note applies to the situation where the OPDMS becomes inoperable while the plant is in MODE 1. Without the continuous monitoring capability of the OPDMS, F_Q limits must be monitored on a periodic basis. The first measurement must be made within 31 days of the most recent date where the OPDMS data has verified peak kw/ft (Z) (and therefore also F_Q) to be within its limit. This is consistent with the 31 day Surveillance Frequency.

<u>SR 3.2.1.1</u>

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

The limit to which $F_Q^C(Z)$ is compared varies inversely with power above 50% RTP.

Performing the Surveillance in MODE 1 prior to exceeding 75% RTP assures that the $F_Q^C(Z)$ limit is met when RTP is achieved because Peaking Factors generally decrease as power level is increased.

BASES

SURVEILLANCE REQUIREMENTS (continued)

If THERMAL POWER has been increased by ≥ 10% RTP since the last determination of $F_Q^C(Z)$, another evaluation of this factor is required 12 hours after achieving equilibrium conditions at this higher power level (to assure that $F_Q^C(Z)$ values are being reduced sufficiently with power increase to stay within the LCO limits).

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

SR 3.2.1.2

The nuclear design process includes calculations performed to determine that the core can be operated within the $F_Q(Z)$ limits. Because flux maps are taken in steady state conditions, the variations in power distribution resulting from normal operational maneuvers are not present in the flux map data. These variations are, however, conservatively calculated by considering a wide range of unit maneuvers in normal operation. The maximum peaking factor increase over steady state values, calculated as a function of core elevation, Z, is called W(Z). Multiplying the measured total peaking factor, $F_Q^C(Z)$, by W(Z) gives the maximum $F_Q(Z)$ calculated to occur in normal operation, $F_Q^W(Z)$.

The limit to which $F_Q^W(Z)$ is compared varies inversely with power.

The W(Z) curve is provided in the COLR for discrete core elevations. $F_Q^W(Z)$ evaluations are not applicable for the following axial core regions, measured in percent of core height:

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

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

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

This Surveillance has been modified by a Note, which may require that more frequent surveillances be performed. If $F_Q^W(Z)$ is evaluated and found to be within its limit, an evaluation of the expression below is

BASES

SURVEILLANCE REQUIREMENTS (continued)

required to account for any increase to $F_Q^M(Z)$ which could occur and cause the $F_Q(Z)$ limit to be exceeded before the next required $F_Q(Z)$ evaluation.

If the two most recent FQ(Z) evaluations show an increase in $F_Q^C(Z)$, it is required to meet the $F_Q(Z)$ limit with the last $F_Q^W(Z)$ increased by the greater of a factor of 1.02 or by an appropriate factor as specified in the COLR or to evaluate $F_Q(Z)$ more frequently, each 7 EFPDs. These alternative requirements will prevent $F_Q(Z)$ from exceeding its limit for any significant period of time without detection.

Performing the Surveillance in MODE 1 prior to exceeding 75% of RTP ensures that the $F_Q(Z)$ limit will be met when RTP is achieved, because peaking factors are generally decreased as power level is increased.

The Surveillance Frequency of 31 EFPDs is adequate to monitor the change of power distribution because such a change is sufficiently slow, when the plant is operated in accordance with Technical Specifications, to preclude the occurrence of adverse peaking factors between 31 EFPD Surveillances. The Surveillance may be done more frequently if required by the results of $F_Q(Z)$ evaluations.

$F_Q(Z)$ is verified at power increases of at least 10% RTP above the THERMAL POWER of its last verification, 12 hours after achieving equilibrium conditions, to assure that $F_Q(Z)$ will be within its limit at higher power levels.

REFERENCES

1. 10 CFR 50.46, “Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors,” 1974.

2. Regulatory Guide 1.77, Rev. 0, “Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors,” May 1974.

3. 10 CFR 50, Appendix A, GDC 26.

4. WCAP-7308-L-P-A, “Evaluation of Nuclear Hot Channel Factor Uncertainties,” June 1988 (Westinghouse Proprietary) and WCAP-7308-L-A (Non-Proprietary).

5. WCAP-10216-P-A, Revision 1A, “Relaxation of Constant Axial Offset Control FQ Surveillance Technical Specification,” February 1994 (Westinghouse Proprietary) and WCAP-10217-A (Non-Proprietary).

B 3.2 POWER DISTRIBUTION LIMITS

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

BASES

BACKGROUND

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

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

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

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

The COLR provides peaking factor limits that ensure that the design basis value of the departure from nucleate boiling (DNB) is met for normal operation, operational transients, and any transient condition arising from events of moderate frequency. The DNB design basis precludes DNB and is met by limiting the minimum local DNB heat flux ratio. Transient

BASES

BACKGROUND (continued)

events that may be DNB limited are assumed to begin with a $F^N_{\Delta H}$ that satisfies the LCO requirements.

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

APPLICABLE SAFETY ANALYSES

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

a. There must be at least a 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the hottest fuel rod in the core does not experience a DNB condition;

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

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 280 cal/gm (Ref. 1); and

d. Fuel design limits required by GDC 26 (Ref. 2) for the condition when 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.

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

The allowable $F^N_{\Delta H}$ limit increases with decreasing power level. This functionality in $F^N_{\Delta H}$ is included in the analyses that provide the Reactor Core Safety Limits (SLs) of SL 2.1.1. Therefore, any DNB events in which the calculation of the core limits is modeled implicitly use this

BASES

APPLICABLE SAFETY ANALYSES (continued)

variable value of $F^N_{\Delta H}$ in the analyses. Likewise, all transients that may be DNB limited are assumed to begin with an initial $F^N_{\Delta H}$ as a function of power level defined by the COLR limit equation.

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

The fuel is protected in part by Technical Specifications, which provide assurance that the initial conditions assumed in the safety and accident analyses remain valid. With the OPDMS OPERABLE, peak kw/ft(Z) and $F^N_{\Delta H}$ are directly monitored. Should the OPDMS become inoperable, the following LCOs assure that the conditions assumed for the safety analysis remain valid: LCO 3.2.3, “AXIAL FLUX DIFFERENCE (AFD),” LCO 3.2.4, “QUADRANT POWER TILT RATIO (QPTR),” LCO 3.1.6, “Control Bank Insertion Limits,” LCO 3.2.2, “Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$),” and LCO 3.2.1, “Heat Flux Hot Channel Factor ($F_Q(Z)$).”

When the OPDMS is not available to measure power distribution parameters continuously, $F^N_{\Delta H}$ and $F_Q(Z)$ are measured periodically using the incore detector system. Measurements are generally taken with the core at, or near, steady-state conditions. Without the OPDMS, core monitoring and control under transient conditions (Condition 1 events) are accomplished by operating the core within the limits of the LCOs on AFD, QPTR, and Bank Insertion Limits.

$F^N_{\Delta H}$ satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

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

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

BASES

LCO (continued)

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

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

APPLICABILITY

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

ACTIONS

A.1.1

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

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

BASES

ACTIONS (continued)

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

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

A.1.2.1 and A.1.2.2

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

The allowed Completion Time of 72 hours to reset the trip setpoints per Required Action A.1.2.2 recognizes that, once power is reduced, the safety analysis assumptions are satisfied and there is no urgent need to reduce the trip setpoints. This is a sensitive operation that may cause an inadvertent reactor trip.

A.2

Once the power level has been reduced to < 50% RTP per Required Action A.1.2.1, an incore flux map (SR 3.2.2.1) must be obtained and the measured value of $F^N_{\Delta H}$ verified not to exceed the allowed limit at the lower power level. The unit is provided 20 additional hours to perform this task over and above the 4 hours allowed by either Action A.1.1 or Action A.1.2.1. The Completion Time of 24 hours is acceptable because

BASES

ACTIONS (continued)

of the increase in the DNB margin, which is obtained at lower power levels, and the low probability of having a DNB limiting event within this 24 hour period. Additionally, operating experience has indicated that this Completion Time is sufficient to obtain the incore flux map, perform the required calculations, and evaluate $F^N_{\Delta H}$.

A.3

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

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

When Required Actions A.1.1 through A.3 cannot be completed within their required Completion Times, the plant must be placed in a mode in which the LCO requirements are not applicable. This is done by placing the plant in at least MODE 2 within 6 hours. The allowed Completion Time of 8 hours is reasonable, based on operating experience regarding the time required to reach MODE 2 from full power conditions in an orderly manner without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1

When the OPDMS is OPERABLE, the value of $F^N_{\Delta H}$ is directly and continuously monitored. With the OPDMS inoperable, the value of $F^N_{\Delta H}$ is determined by using the incore detector system to obtain a flux distribution map. A data reduction computer program then calculates the maximum value of $F^N_{\Delta H}$ from the measured flux distributions. The measured value of $F^N_{\Delta H}$ must be multiplied by a measurement uncertainty factor before making comparisons to the $F^N_{\Delta H}$ limit.

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

BASES

SURVEILLANCE REQUIREMENTS (continued)

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

REFERENCES

1. Regulatory Guide 1.77, Rev. 0, May 1979.
2. 10 CFR 50, Appendix A, GDC 26.
3. 10 CFR 50.46, “Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors.”

B 3.2 POWER DISTRIBUTION LIMITS

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

BASES

BACKGROUND

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

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

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

Although the RAOC defines limits that must be met to satisfy safety analyses, typically, without the OPDMS, an operating scheme, Constant Axial Offset Control (CAOC), is used to control axial power distribution in day-to-day operation (Ref. 1). CAOC requires that the AFD be controlled within a narrow tolerance band around a burnup-dependent target to minimize the variation of axial peaking factors and axial xenon distribution during unit maneuvers.

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

BASES

APPLICABLE SAFETY ANALYSES

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

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

Three dimensional power distribution calculations are performed to demonstrate that normal operation power shapes are acceptable for the LOCA, the loss of flow accident, and for initial conditions of anticipated transients. The tentative limits are adjusted as necessary to meet the safety analysis requirements.

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

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

LCO

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

Signals are available to the operator from the Protection and Safety Monitoring System (PMS) excore neutron detectors (Ref. 3). Separate signals are taken from the top and bottom detectors. The AFD is defined as the difference in normalized flux signals between the top and bottom

BASES

LCO (continued)

excore detectors in each detector well. For convenience, this flux difference is converted to provide flux difference units expressed as a percentage and labeled as %Δ flux or %ΔI.

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

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

APPLICABILITY

The AFD requirements are applicable in MODE 1 greater than or equal to 50% RTP where the combination of THERMAL POWER and core peaking factors are of primary importance in safety analysis.

For AFD limits developed using RAOC methodology, the value of the AFD does not affect the limiting accident consequences with THERMAL POWER < 50% RTP and for lower operating power MODES. With the OPDMS inoperable, it is necessary to monitor AFD via the excore detectors to ensure that it remains within the RAOC limits.

ACTIONS

A.1

Required Action A.1 requires a THERMAL POWER reduction to < 50% RTP. This places the core in a condition where the value of the AFD is not important in the applicable safety analyses. A Completion Time of 30 minutes is reasonable, based on operating experience, to reach 50% RTP without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.3.1

This surveillance verifies that the AFD, as indicated by the PMS excore channel, is within its specified limits. The Surveillance Frequency of 7 days is adequate considering that the AFD is monitored by a computer and any deviation from requirements is alarmed.

BASES

REFERENCES

1. WCAP-8385, “Power Distribution Control and Load Following Procedures,” Westinghouse Electric Corporation, September 1974 (Westinghouse Proprietary) and WCAP-8403 (Non-Proprietary).

2. R. W. Miller et al., “Relaxation of Constant Axial Offset Control: F_Q Surveillance Technical Specification,” WCAP-10216-P-A, June 1983 (Westinghouse Proprietary) and WCAP-10217-A (Non-Proprietary).

3. Chapter 15, “Accident Analysis.”

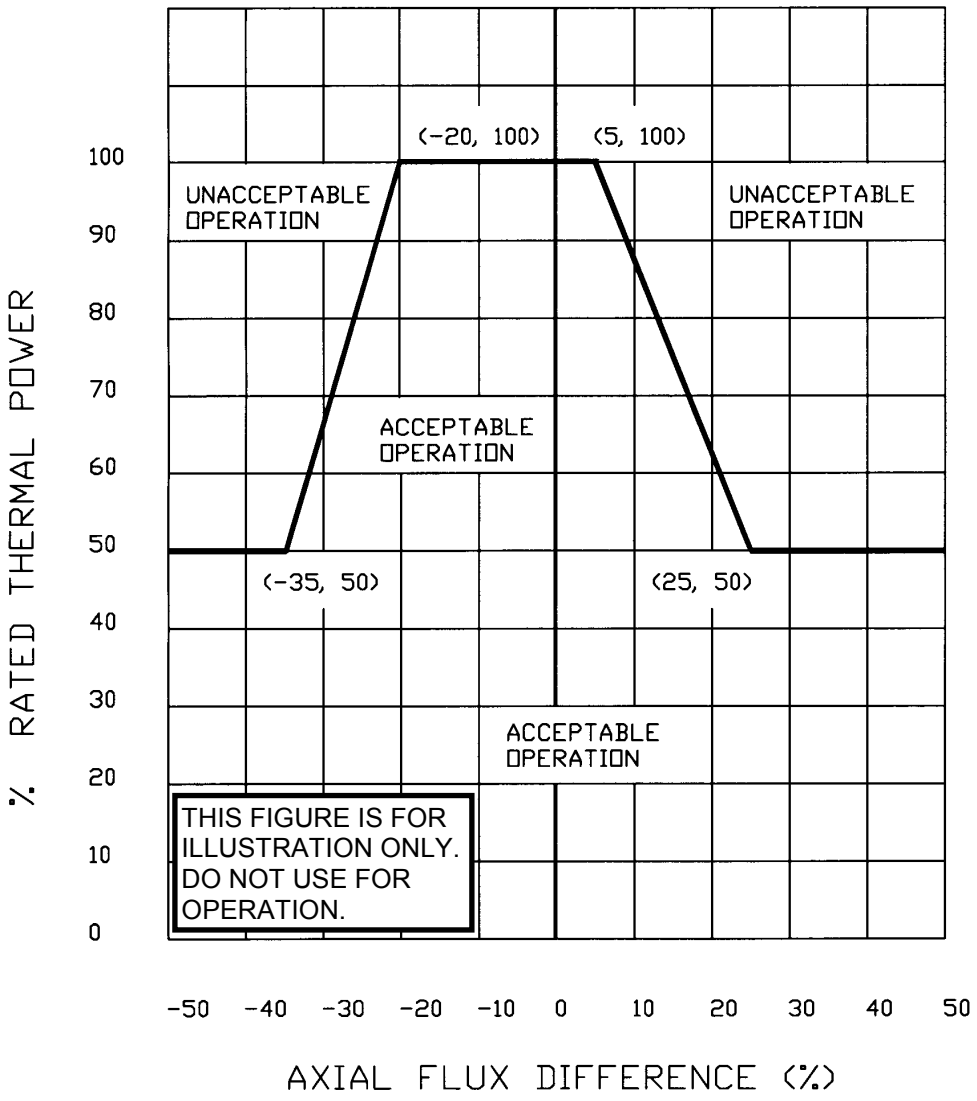

Figure B 3.2.3-1 (page 1 of 1)
AXIAL FLUX DIFFERENCE Limits as a Function
of RATED THERMAL POWER

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 QUADRANT POWER TILT RATIO (QPTR)

BASES

BACKGROUND

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

The power density at any point in the core must be limited so that the fuel design criteria are maintained. With the OPDMS OPERABLE, the peak kw/ft(Z) is continuously and directly monitored. With the OPDMS inoperable, LCO 3.2.3, “AXIAL FLUX DIFFERENCE (AFD),” LCO 3.2.4, “QUADRANT POWER TILT RATIO (QPTR),” and LCO 3.1.6, “Control Rod Insertion Limits,” provide limits on process variables that characterize and control the three dimensional power distribution of the reactor core. Control of these variables ensures that the core operates within the fuel design criteria and that the power distribution remains within the bounds used in the safety analyses.

APPLICABLE SAFETY ANALYSES

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

a. During a large break loss of coolant accident (LOCA), the peak cladding temperature (PCT) 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 a 95% confidence level (the 95/95 departure from nucleate boiling (DNB) criterion) that the hot fuel rod in the core does not experience a DNB condition;

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

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

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

BASES

APPLICABLE SAFETY ANALYSES (continued)

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

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

The QPTR satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

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

APPLICABILITY

The QPTR limit must be maintained in MODE 1 with THERMAL POWER > 50% RTP to preclude core power distributions from exceeding the design limits. With the OPDMS inoperable, a continuous on-line indication of core peaking factors is not available. Therefore, QPTR must be monitored and the limits on QPTR ensure that peaking factors will be within design limits.

Applicability in MODE 1 ≤ 50% RTP and in other MODES is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the reactor coolant to require the implementation of a QPTR limit on the distribution of core power. The QPTR limit in these conditions is, therefore, not important. Note that the $F^N_{\Delta H}$ and $F_Q(Z)$ LCOs still apply, but allow progressively higher peaking factors at 50% RTP or lower.

ACTIONS

A.1

With the QPTR exceeding its limit, and the OPDMS inoperable, a power level reduction of 3% RTP for each 1% by which the QPTR exceeds 1.00 is a conservative tradeoff of total core power with peak linear power. The Completion Time of 2 hours allows sufficient time to identify the cause and correct the tilt. Note that the power reduction itself may cause a change in the tilted condition.

BASES

ACTIONS (continued)

The maximum allowable power level initially determined by Required Action A.1 may be affected by subsequent determinations of QPTR. Increases in QPTR would require power reduction within 2 hours of QPTR determination, if necessary to comply with the decreased maximum allowable power level and increasing power up to this revised limit.

A.2

After completion of Required Action A.1, the QPTR alarm may be in its alarmed state. As such, any additional changes in the QPTR are detected by requiring a check of the QPTR once per 12 hours thereafter. A 12 hour Completion Time is sufficient because any additional change in QPTR would be relatively slow.

A.3

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

A.4

Although $F_{\Delta H}^N$ and $F_Q(Z)$ are of primary importance as initial conditions in the safety analyses, other changes in the power distribution may occur as the QPTR limit is exceeded and may have an impact on the validity of the safety analysis. A change in the power distribution can affect such

ACTIONS (continued)

reactor parameters as bank worths and peaking factors for rod malfunction accidents. When the QPTR exceeds its limit, it does not necessarily mean a safety concern exists. It does mean that there is an indication of a change in the gross radial power distribution that requires an investigation and evaluation that is accomplished by examining the incore power distribution. Specifically, the core peaking factors and the quadrant tilt must be evaluated because they are the factors which best characterize the core power distribution. This re-evaluation is required to assure that, before increasing THERMAL POWER to above the limit of Required Action A.1, the reactor core conditions are consistent with the assumptions in the safety analyses.

A.5

If the QPTR has exceeded the 1.02 limit and a re-evaluation of the safety analysis is completed and shows that safety requirements are met, the excore detectors are normalized to restore QPTR to within limits prior to increasing THERMAL POWER to above the limit of Required Action A.1. Normalization is accomplished in such a manner that the indicated QPTR following normalization is near 1.00. This is done to detect any subsequent significant changes in QPTR.

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

A.6

Once the flux tilt is restored to within limits (i.e., Required Action A.5 is performed), it is acceptable to return to full power operation. However, as an added check that the core power distribution is consistent with the safety analysis assumptions, Required Action A.6 requires verification that $F_Q(Z)$ as approximated by $F_Q^C(Z)$ and $F_Q^W(Z)$, and $F_{\Delta H}^N$ are within their specified limits within 24 hours of achieving equilibrium conditions at RTP. As an added precaution, if the core power does not reach equilibrium conditions at RTP within 24 hours, but is increased slowly,

BASES

ACTIONS (continued)

then the peaking factor surveillances must be performed within 48 hours after increasing THERMAL POWER above the limit of Required Action A.1. These Completion Times are intended to allow adequate time to increase THERMAL POWER to above the limit of Required Action A.1, while not permitting the core to remain with unconfirmed power distributions for extended periods of time.

Required Action A.6 is modified by a Note that states that the peaking factor surveillances may only be done after the excore detectors have been calibrated to show zero tilt (i.e., Required Action A.5). The intent of this Note is to have the peaking factor surveillances performed at operating power levels, which can only be accomplished after the excore detectors are calibrated to show zero tilt and the core returned to power.

B.1

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

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1

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

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

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

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.2.4.2

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

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

For purposes of monitoring the QPTR when one power range channel is inoperable, the incore detectors are used to confirm that the normalized symmetric power distribution is acceptable.

With the OPDMS and one PMS channel inoperable, the surveillance of the incore power distribution on a 12 hour basis is sufficient to maintain peaking factors within their normal limits, especially, considering the other LCOs and ACTIONS required when the OPDMS is out of service.

REFERENCES

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

2. Regulatory Guide 1.77, Rev. 0, “Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors,” May 1974.

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

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.5 OPDMS-Monitored Parameters

BASES

BACKGROUND

The On-line Power Distribution Monitoring System (OPDMS) for the AP1000 is an advanced core monitoring and support package. The OPDMS has the ability to continuously monitor core power distribution parameters. In addition, the OPDMS monitors SDM.

The purpose of the limits on the OPDMS-monitored power distribution parameters is to provide assurance of fuel integrity during Conditions I (Normal Operation) and II (incidents of Moderate Frequency) events by: (1) not exceeding the minimum departure from boiling ratio (DNBR) in the core, and (2) limiting the fission gas release, fuel pellet temperature, and cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the peak cladding temperature (PCT) limit of 2200°F is not exceeded.

The definition of certain quantities used in these specifications are as follows:

Peak kw/ft(Z)	Peak linear power density (axially dependent) as measured in kw/ft.
$F^N_{\Delta H}$	Ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.
Minimum DNBR	Minimum ratio of the critical heat flux to actual heat flux at any point in the reactor that is allowed in order to assure that certain performance and safety criteria requirements are met over the range of plant conditions.

By continuously monitoring the core and following its actual operation, it is possible to significantly limit the adverse nature of power distribution initial conditions for transients which may occur at any time.

BASES

APPLICABLE SAFETY ANALYSES

The limits on the above parameters preclude core power distributions from occurring which would violate the following fuel design criteria:

a. During a large break loss of coolant accident (LOCA), the PCT must not exceed a limit of 2200°F (Ref. 1);

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

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

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn.

Limits on linear power density or peak kw/ft assure that the peak linear power density assumed as a base condition in the LOCA analyses is not exceeded during normal operation.

Limits on $F_{\Delta H}$ ensure that the LOCA analysis assumptions and assumptions made with respect to the Overtemperature ΔT Setpoint are maintained.

The limit on DNBR ensures that if transients analyzed in the safety analyses initiate from the conditions within the limit allowed by the OPDMS, the DNB criteria will be met.

The OPDMS-monitored power distribution parameters of this LCO satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

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

Violating the OPDMS-monitored power distribution parameter limits could result in unanalyzed conditions should a design basis event occur while the parameters are outside their specified limits.

BASES

LCO (continued)

Peak kw/ft limits define limiting values for core power peaking that precludes peak cladding temperatures above 2200°F during either a large or small break LOCA. The highest calculated linear power densities in the core at specific core elevations are displayed for operator visual verification relative to the COLR values.

The determination of $F^N_{\Delta H}$ identifies the coolant flow channel with the maximum enthalpy rise. This channel has the least heat removal capability and thus the highest probability for DNB. Should $F^N_{\Delta H}$ exceed the limit given in the COLR, the possibility exists for DNBR to exceed the value used as a base condition for the safety analysis.

Two levels of alarms on power distribution parameters are provided to the operator. One serves as a warning before the three parameters (kw/ft(Z), $F^N_{\Delta H}$, DNBR) exceed their values used as a base condition for the safety analysis. The other alarm indicates when the parameters have reached their limits.

APPLICABILITY

The OPDMS-monitored power distribution parameter limits must be maintained in MODE 1 above 50% RTD to preclude core power distributions from exceeding the limits assumed in the safety analyses. Applicability in other MODES, and MODE 1 below 50% RTP, is not required because there is either insufficient stored energy in the fuel or insufficient energy transferred to the reactor coolant to require a limit on the distribution of core power. The OPDMS monitoring of SDM must be OPERABLE in MODES 1 and 2.

Specifically for $F^N_{\Delta H}$, the design bases accidents (DBAs) that are sensitive to $F^N_{\Delta H}$ in other MODES (MODES 2 through 5) have significant margin to DNB, and therefore, there is no need to restrict $F^N_{\Delta H}$ in these modes.

In addition to the alarms discussed in the LCO section above (alarms on OPDMS-monitored power distribution parameters), there is an alarm indicating the potential inoperability of the OPDMS itself.

Should the OPDMS be determined to be inoperable for other than reasons of alarms inoperable, this LCO is no longer applicable and LCOs 3.2.1 through 3.2.4 become applicable.

BASES

ACTIONS

A.1

With any of the OPDMS-monitored power distribution parameters outside of their limits, the assumptions used as most limiting base conditions for the DBA analyses may no longer be valid. The 1 hour operator ACTION requirement to restore the parameter to within limits is consistent with the basis for the anticipated operational occurrences and provides time to assess if there are instrumentation problems. It also allows the possibility to restore the parameter to within limits by rod cluster control assembly (RCCA) motion if this is possible. The OPDMS will continuously monitor these parameters and provide an indication when they are approaching their limits.

B.1

If the OPDMS-monitored power distribution parameters cannot be restored to within their limits within the Completion Time of ACTION A.1, it is likely that the problem is not due to a failure of instrumentation. Most of these parameters can be brought within their respective limits by reducing THERMAL POWER because this will reduce the absolute power density at any location in the core thus providing margin to the limit.

If the parameters cannot be returned to within limits as power is being reduced, THERMAL POWER must be reduced to < 50% RTP where the LCOs are no longer applicable.

A Note has been added to indicate that if the power distribution parameters in violation are returned to within their limits during the power reduction, then power operation may continue at the power level where this occurs. This is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions.

The Completion Time of 4 hours provides an acceptable time to reduce power in an orderly manner and without allowing the plant to remain outside the $F^N_{\Delta H}$ limits for an extended period of time.

C.1

If the SDM requirements are not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. It is assumed that boration will be continued until the SDM requirements are met. In the determination of the required combination of boration flow rate and boron concentration, there is no unique requirement that must be satisfied. Since it is imperative to raise the boron concentration of the RCS as soon

BASES

ACTIONS (continued)

as possible, the boron concentration should be a concentrated solution. The operator should begin boration with the best source available for the plant conditions.

SURVEILLANCE REQUIREMENTS

With OPDMS operating, the power distribution parameters are continuously computed and displayed, and compared against their limit. Two levels of alarms are provided to the operator. The first alarm provides a warning before these parameters (kw/ft(Z), $F^N_{\Delta H}$, and DNBR) exceed their limits. The second alarm indicates when they actually reach their limits. A third alarm indicates trouble with the OPDMS system.

SR 3.2.5.1

This Surveillance requires the operator to verify that the power distribution parameters are within their limits. This confirmation is a verification in addition to the automated checking performed by the OPDMS system. A 24 hour Surveillance interval provides assurance that the system is functioning properly and that the core limits are met.

With the OPDMS parameter alarms inoperable, an increased Surveillance Frequency is provided to assure that parameters are not approaching the limits. A 12 hour Frequency is adequate to identify changes in these parameters that could lead to their exceeding their limits.

REFERENCES

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

2. Regulatory Guide 1.77, Rev. 0, “Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors,” May 1974.

B 3.3 INSTRUMENTATION

B 3.3.1 Reactor Trip System (RTS) Instrumentation

BASES

BACKGROUND

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

The Protection and Safety Monitoring System (PMS) has been designed to assure safe operation of the reactor. This is achieved by specifying limiting safety system settings (LSSS) in terms of parameters directly monitored by the RTS, as well as specifying LCOs on other reactor system parameters and equipment performance.

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

1. The Departure from Nucleate Boiling Ratio (DNBR) shall be maintained above the Safety Limit (SL) value to prevent departure from nucleate boiling (DNB);

2. Fuel centerline melt shall not occur; and

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

Operation within the SLs of Specification 2.0, “Safety Limits (SLs),” also maintains the above values and assures that offsite doses are within the acceptance criteria during AOOs.

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

The RTS maintains surveillance on key process variables which are directly related to equipment mechanical limitations, such as pressure, and on variables which directly affect the heat transfer capability of the reactor, such as flow and temperature. Some limits, such as Overtemperature ΔT, are calculated in the protection and safety

BASES

BACKGROUND (continued)

monitoring system cabinets from other parameters when direct measurement of the variable is not possible.

The RTS instrumentation is segmented into four distinct but interconnected modules as identified below:

- Field inputs from process sensors, nuclear instrumentation;
- Protection and Safety Monitoring System Cabinets;
- Voting Logic; and
- Reactor Trip Switchgear Interface.

Field Transmitters and Sensors

Normally, four redundant measurements using four separate sensors are made for each variable used for reactor trip. The use of four channels for protection functions is based on a minimum of two channels being required for a trip or actuation, one channel in test or bypass, and a single failure on the remaining channel. The signal selector algorithm in the Plant Control System (PLS) will function with only three channels. This includes two channels properly functioning and one channel having a single failure. For protection channels providing data to the control system, the fourth channel permits one channel to be in test or bypass. Minimum requirements for protection and control is achieved with only three channels OPERABLE. The fourth channel is provided to increase plant availability, and permits the plant to run for an indefinite time with a single channel out of service. The circuit design is able to withstand both an input failure to the control system, which may then require the protection Function actuation, and a single failure in the other channels providing the protection Function actuation. Again, a single failure will neither cause nor prevent the protection Function actuation. These requirements are described in IEEE-603 (Ref. 5). The actual number of channels required for each plant parameter is specified in Reference 2.

Selected analog measurements are converted to digital form by digital converters within the protection and safety monitoring system cabinets. Signal conditioning may be applied to selected inputs following the conversion to digital form. Following necessary calculations and processing, the measurements are compared against the applicable setpoint for that variable. A partial trip signal for the given parameter is generated if one channel measurement exceeds its predetermined or calculation limit. Processing on all variables for reactor trip is duplicated in each of the four redundant divisions of the protection system. Each

BASES

BACKGROUND (continued)

division sends its partial trip status to each of the other three divisions over isolated multiplexed links. Each division is capable of generating a reactor trip signal if two or more of the redundant channels of a single variable are in the partial trip state.

The reactor trip signal from each division is sent to the corresponding reactor trip actuation division. Each of the four reactor trip actuation divisions consists of two reactor trip circuit breakers. The reactor is tripped when two or more actuation divisions receive a reactor trip signal. This automatic trip demand initiates the following two actions:

1. It de-energizes the undervoltage trip attachment on each reactor trip breaker, and

2. It energizes the shunt trip device on each reactor trip breaker.

Either action causes the breakers to trip. Opening of the appropriate trip breakers removes power to the control rod drive mechanism (CRDM) coils, allowing the rods to fall into the core. This rapid negative reactivity insertion shuts down the reactor.

Protection and Safety Monitoring System Cabinets

The protection and safety monitoring system cabinets contain the necessary equipment to:

- Permit acquisition and analysis of the sensor inputs, including plant process sensors and nuclear instrumentation, required for reactor trip and ESF calculations;

- Perform computation or logic operations on variables based on these inputs;

- Provide trip signals to the reactor trip switchgear and ESF actuation data to the ESF coincidence logic as required;

- Permit manual trip or bypass of each individual reactor trip Function and permit manual actuation or bypass of each individual voted ESF Function;

- Provide data to other systems in the Instrumentation and Control (I&C) architecture;

- Provide separate input circuitry for control Functions that require input from sensors that are also required for protection Functions.

BASES

BACKGROUND (continued)

Each of the four divisions provides signal conditioning, comparable output signals for indications in the main control room, and comparison of measured input signals with established setpoints. The basis of the setpoints are described in References 1, 2, and 3. If the measured value of a unit parameter exceeds the predetermined setpoint, an output is generated which is transmitted to the ESF coincidence logic for logic evaluation.

Within the protection and safety monitoring system redundancy is generally provided for active equipment such as processors and communication hardware. This redundancy is provided to increase plant availability and facilitate surveillance testing. A division or channel is OPERABLE if it is capable of performing its specified safety function(s) and all the required supporting functions or systems are also capable of performing their related support functions. Thus, a division or channel is OPERABLE as long as one set of redundant components within the division or channel is capable of performing its specified safety function(s).

Voting Logic

The voting logic provides a reliable means of opening the reactor trip switchgear in its own division as demanded by the individual protection functions.

Reactor Trip Switchgear Interface

The final stage of the voting logic provides the signal to energize the undervoltage trip attachment on each RTB within the reactor trip switchgear. Loss of the signal de-energizes the undervoltage trip attachments and results in the opening of those reactor trip switchgear. An additional external relay is de-energized with loss of the signal. The normally closed contacts of the relay energize the shunt trip attachments on each switchgear at the same time that the undervoltage trip attachment is de-energized. This diverse trip actuation is performed external to the PMS cabinets. The switchgear interface including the trip attachments and the external relay are within the scope of the PMS. Separate outputs are provided for each switchgear. Testing of the interface allows trip actuation of the breakers by either the undervoltage trip attachment or the shunt trip attachment.

BASES

BACKGROUND (continued)

Trip Setpoints and Allowable Values

The Trip Setpoints are the nominal values at which the trip output is set. Any trip output is considered to be properly adjusted when the “as left” value is within the band for CHANNEL CALIBRATION accuracy (i.e., ± rack calibration accuracy).

The Trip Setpoints used in the trip output are based on the analytical limits stated in Reference 1. The selection of these Trip Setpoints is such that adequate protection is provided when all sensor and processing time delays are taken into account. To allow for calibration tolerances, instrument drift, and severe environment errors for those RTS channels that must function in harsh environments as defined by 10 CFR 50.49 (Ref. 6), the Trip Setpoints and Allowable Values specified in Table 3.3.1-1 in the accompanying LCO are conservatively adjusted with respect to the analytical limits. A detailed description of the methodology used to calculate the Trip Setpoints, including their explicit uncertainties, is provided in the “Westinghouse Setpoint Methodology for Protection Systems” (Refs. 4 and 9). The actual nominal Trip Setpoint entered into the trip output is more conservative than that specified by the Allowable Value to account for changes in random measurement errors detectable by a COT. One example of such a change in measurement error is drift during the surveillance interval. If the measured setpoint does not exceed the Allowable Value, the trip output is considered OPERABLE.

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

The Trip Setpoints and Allowable Values listed in Table 3.3.1-1 are based on the methodology described in Reference 4, which incorporates all of the known uncertainties applicable for each channel. (Reference 4 is an AP600 document that describes a methodology that is applicable to AP1000. AP1000 has some slight differences in instrument spans as a result of the higher power level.) 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. Transmitter and signal processing equipment calibration tolerances and drift allowances must be specified in plant calibration procedures, and must be consistent with the values used in the setpoint methodology.

BASES

BACKGROUND (continued)

The OPERABILITY of each transmitter or sensor can be evaluated when its “as found” calibration data are compared against the “as left” data and are shown to be within the setpoint methodology assumptions. The basis of the setpoints is described in References 1, 2, 3, and 4. Trending of transmitter calibration is required by Generic Letter 91-04, “Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle.”

The protection and safety monitoring system testing features are designed to allow for complete functional testing by using a combination of system self-checking features, functional testing features, and other testing features. Successful functional testing consists of verifying that the capability of the system to perform the safety function has not failed or degraded. For hardware functions this would involve verifying that the hardware components and connections have not failed or degraded. Since software does not degrade, software functional testing involves verifying that the software code has not changed and that the software code is executing. To the extent possible, protection and safety monitoring system functional testing will be accomplished with continuous system self-checking features and the continuous functional testing features.

The protection and safety monitoring system incorporates continuous system self-checking features wherever practical. Self-checking features include on-line diagnostics for the computer system and the hardware and communications tests. These self-checking tests do not interfere with normal system operation.

In addition to the self-checking features, the system includes functional testing features. Functional testing features include continuous functional testing features and manually initiated functional testing features. To the extent practical functional testing features are designed not to interfere with normal system operation.

In addition to the system self-checking features and functional testing features, other test features are included for those parts of the system which are not tested with self-checking features or functional testing features. These test features allow for instruments/sensor checks, calibration verification, response time testing, setpoint verification and component testing. The test features again include a combination of continuous testing features and manual testing features.

BASES

BACKGROUND (continued)

All of the testing features are designed so that the duration of the testing is as short as possible. Testing features are designed so that the actual logic is not modified. To prevent unwanted actuation, the testing features are designed with either the capability to bypass a Function during testing and/or limit the number of signals allowed to be placed in test at one time.

Reactor Trip (RT) Channel

An RT Channel extends from the sensor to the output of the associated reactor trip subsystem in the protection and safety monitoring system cabinets, and includes the sensor (or sensors), the signal conditioning, any associated datalinks, and the associated reactor trip subsystem. For RT Channels containing nuclear instrumentation, the RT Channel also includes the nuclear instrument signal conditioning and the associated Nuclear Instrumentation Signal Processing and Control (NISPAC) subsystem.

Automatic Trip Logic

The Automatic Trip Logic extends from, but does not include, the outputs of the various RT Channels to, but does not include, the reactor trip breakers. Operator bypass of a reactor trip function is performed within the Automatic Trip Logic.

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY

The RTS functions to maintain the SLs during all AOOs and mitigates the consequences of DBAs in all MODES in which the RTBs are closed.

Each of the analyzed accidents and transients which require reactor trip can be detected by one of more RTS Functions. The accident analysis described in Reference 3 takes credit for most RTS trip Functions. RTS trip Functions not specifically credited in the accident analysis were qualitatively credited in the safety analysis and the NRC staff approved licensing basis for the plant. These RTS trip Functions may provide protection for conditions which do not require dynamic transient analysis to demonstrate function performance. These RTS trip Functions may also serve as backups to RTS trip Functions that were credited in the accident analysis.

The LCO requires all instrumentation performing an RTS Function, listed in Table 3.3.1-1 in the accompanying LCO, to be OPERABLE. Failure of any instrument renders the affected channel(s) inoperable and reduces the reliability of the affected Functions.

The LCO generally requires OPERABILITY of three channels in each instrumentation Function.

BASES

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

Reactor Trip System Functions

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

1. Manual Reactor Trip

The Manual Reactor Trip ensures that the main control room operator can initiate a reactor trip at any time by using either of two reactor trip actuation devices in the main control room. A Manual Reactor Trip accomplishes the same results as any one of the automatic trip Functions. It can be used by the reactor operator to shutdown the reactor whenever any parameter is rapidly trending toward its Trip Setpoint. The safety analyses do not take credit for the Manual Reactor Trip.

The LCO requires two Manual Reactor Trip actuation devices be OPERABLE in MODES 1 and 2 and in MODES 3, 4, and 5 with RTBs closed and PLS capable of rod withdrawal. Two independent actuation devices are required to be OPERABLE so that no single random failure will disable the Manual Reactor Trip Function.

In MODE 1 or 2, manual initiation of a reactor trip must be OPERABLE. These are the MODES in which the shutdown rods and/or control rods are partially or fully withdrawn from the core. In MODE 3, 4, or 5, the manual initiation Function must also be OPERABLE if the shutdown or control rods are withdrawn or the PLS is capable of withdrawing the shutdown or control rods. In MODES 3, 4, and 5, manual initiation of a reactor trip does not have to be OPERABLE if the PLS is not capable of withdrawing the shutdown or control rods. If the rods cannot be withdrawn from the core, there is no need to be able to trip the reactor because all of the rods are inserted. In MODE 6, neither the shutdown rods nor the control rods are permitted to be withdrawn and the CRDMs are disconnected from the control rods and shutdown rods. Therefore, the manual initiation Function does not have to be OPERABLE.

2. Power Range Neutron Flux

The PMS power range detectors are located external to the reactor vessel and measure neutrons leaking from the core. The PMS power range detectors provide input to the PLS. Minimum requirements for protection and control is achieved with three channels OPERABLE. The fourth channel is provided to

BASES

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

increase plant availability, and permits the plant to run for an indefinite time with a single channel in trip or bypass. This Function also satisfies the requirements of IEEE 603 (Ref. 5) with 2/4 logic. This Function also provides a signal to prevent automatic and manual rod withdrawal prior to initiating a reactor trip. Limiting further rod withdrawal may terminate the transient and eliminate the need to trip the reactor.

a. Power Range Neutron Flux–High

The Power Range Neutron Flux–High trip Function ensures that protection is provided, from all power levels, against a positive reactivity excursion during power operations. Positive reactivity excursions can be caused by rod withdrawal or reductions in RCS temperature.

The LCO requires four Power Range Neutron Flux–High channels to be OPERABLE in MODES 1 and 2.

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

b. Power Range Neutron Flux–Low

The LCO requirement for the Power Range Neutron Flux–Low trip Function ensures that protection is provided against a positive reactivity excursion from low power or subcritical conditions. The Trip Setpoint reflects only steady state instrument uncertainties as this Function does not provide primary protection for any event that results in a harsh environment.

The LCO requires four of the Power Range Neutron Flux–Low channels to be OPERABLE in MODE 1 below the Power Range Neutron Flux P-10 Setpoint and MODE 2.

BASES

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

In MODE 1, below the Power Range Neutron Flux P-10 setpoint and in MODE 2, the Power Range Neutron Flux–Low trip must be OPERABLE. This Function may be manually blocked by the operator when the respective power range channel is greater than approximately 10% of RTP (P-10 setpoint). This Function is automatically unblocked when the respective power range channel is below the P-10 setpoint. Above the P-10 setpoint, positive reactivity additions are mitigated by the Power Range Neutron Flux–High trip Function.

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

3. Power Range Neutron Flux–High Positive Rate

The Power Range Neutron Flux–High Positive Rate trip Function ensures that protection is provided against rapid increases in neutron flux which are characteristic of a rod cluster control assembly (RCCA) drive rod housing rupture and the accompanying ejection of the RCCA. This Function compliments the Power Range Neutron Flux–High and Low trip Functions to ensure that the criteria are met for a rod ejection from the power range. The Power Range Neutron Flux Rate trip uses the same channels as discussed for Function 2 above.

The LCO requires four Power Range Neutron Flux–High Positive Rate channels to be OPERABLE. In MODE 1 or 2, when there is a potential to add a large amount of positive reactivity from a rod ejection accident (REA), the Power Range Neutron Flux–High Positive Rate trip must be OPERABLE. In MODE 3, 4, 5, or 6, the Power Range Neutron Flux–High Positive Rate trip Function does not have to be OPERABLE because other RTS trip Functions and administrative controls will provide protection against positive reactivity additions. Also, since only the shutdown banks may be withdrawn in MODE 3, 4, or 5, the remaining complement of control bank worth ensures a SDM in the event of an REA. In MODE 6, no rods are withdrawn and the SDM is increased during refueling operations. The reactor vessel head is also removed or the closure

BASES

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

bolts are detensioned preventing any pressure buildup. In addition, the PMS power range detectors cannot detect neutron levels present in this MODE.

4. Intermediate Range Neutron Flux

The Intermediate Range Neutron Flux trip Function ensures that protection is provided against an uncontrolled RCCA bank withdrawal accident from a subcritical condition during startup. This trip Function provides redundant protection to the Power Range Neutron Flux–Low Setpoint trip Function. The PMS intermediate range detectors are located external to the reactor vessel and measure neutrons leaking from the core. The safety analyses do not take credit for the Intermediate Range Neutron Flux trip Function. Even though the safety analyses take no credit for the Intermediate Range Neutron Flux trip, the functional capability at the specified Trip Setpoint enhances the overall diversity of the RTS. The Trip Setpoint reflects only steady state instrument uncertainties as the detectors do not provide primary protection for any events that result in a harsh environment. This trip can be manually blocked by the main control room operator when above the P-10 setpoint, which is the respective PMS power range channel greater than 10% power, and is automatically unblocked when below the P-10 setpoint, which is the respective PMS power range channel less than 10% power. This Function also provides a signal to prevent automatic and manual rod withdrawal prior to initiating a reactor trip. Limiting further rod withdrawal may terminate the transient and eliminate the need to trip the reactor.

The LCO requires four channels of Intermediate Range Neutron Flux to be OPERABLE. Four channels are provided to permit one channel in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function.

In MODE 1 below the P-10 setpoint, and in MODE 2, when there is a potential for an uncontrolled rod withdrawal accident during reactor startup, the Intermediate Range Neutron Flux trip must be OPERABLE. Above the P-10 setpoint, the Power Range Neutron Flux–High Setpoint trip and the Power Range Neutron Flux–High Positive Rate trip provide core protection for a rod withdrawal accident. In MODE 3, 4, or 5, the Intermediate Range Neutron Flux trip does not have to be OPERABLE because the control rods must be fully inserted and only the shutdown rods may be withdrawn. The reactor cannot be started up in this condition. The core also has the

BASES

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

required SDM to mitigate the consequences of a positive reactivity addition accident. In MODE 6, all rods are fully inserted and the core has a required increased SDM. Also, the PMS intermediate range detectors cannot detect neutron levels present in this MODE.

5. Source Range Neutron Flux

The LCO requirement for the Source Range Neutron Flux trip Function ensures that protection is provided against an uncontrolled bank rod withdrawal accident from a subcritical condition during startup. This trip Function provides redundant protection to the Power Range Neutron Flux – Low Setpoint and Intermediate Range Neutron Flux trip Functions. In MODES 3, 4, and 5, administrative controls also prevent the uncontrolled withdrawal of rods. The PMS source range detectors are located external to the reactor vessel and measure neutrons leaking from the core. The safety analyses do not take credit for the Source Range Neutron Flux trip Function. Even though the safety analyses take no credit for the Source Range Neutron Flux trip, the functional capability at the specified Trip Setpoint is assumed to be available and the trip is implicitly assumed in the safety analyses.

The Trip Setpoint reflects only steady state instrument uncertainties as the detectors do not provide primary protection for any events that result in a harsh environment. This trip can be manually blocked by the main control room operator when above the P-6 setpoint (Intermediate Range Neutron Flux interlock) and is automatically unblocked when below the P-6 setpoint. The manual block of the trip function also de-energizes the source range detectors. The source range detectors are automatically re-energized when below the P-6 setpoint. The trip is automatically blocked when above the P-10 setpoint (Power Range Neutron Flux interlock). The source range trip is the only RTS automatic protective Function required in MODES 3, 4, and 5. Therefore, the functional capability at the specified Trip Setpoint is assumed to be available.

The LCO requires four channels of Source Range Neutron Flux to be OPERABLE in MODE 2 below P-6 and in MODE 3, 4, or 5 with RTBs closed and Control Rod Drive System capable of rod withdrawal. Four channels are provided to permit one channel in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function. In MODE 3, 4, or 5 with the RTBs open, the LCO does not require the Source Range Neutron Flux channels for reactor trip Functions to be OPERABLE.

BASES

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

In MODE 2 when below the P-6 setpoint during a reactor startup, the Source Range Neutron Flux trip must be OPERABLE. Above the P-6 setpoint, the Intermediate Range Neutron Flux trip and the Power Range Neutron Flux – Low Setpoint trip will provide core protection for reactivity accidents. Above the P-6 setpoint, the PMS source range detectors are de-energized and inoperable as described above.

In MODE 3, 4, or 5 with the reactor shutdown, the Source Range Neutron Flux trip Function must also be OPERABLE. If the PLS is capable of rod withdrawal, the Source Range Neutron Flux trip must be OPERABLE to provide core protection against a rod withdrawal accident. If the PLS is not capable of rod withdrawal, the source range detectors are required to be OPERABLE to provide monitoring of neutron levels and provide protection for events like an inadvertent boron dilution. These Functions are addressed in LCO 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation." The requirements for the PMS source range detectors in MODE 6 are addressed in LCO 3.9.3, "Nuclear Instrumentation."

6. Overtemperature ΔT

The Overtemperature ΔT trip Function ensures that protection is provided to ensure that the design limit DNBR is met. This trip Function also limits the range over which the Overpower ΔT trip Function must provide protection. The inputs to the Overtemperature ΔT trip include all combinations of pressure, power, coolant temperature, and axial power distribution, assuming full reactor coolant flow. Protection from violating the DNBR limit is assured for those transients that are slow with respect to delays from the core to the measurement system. The Overtemperature ΔT trip Function uses the measured T_{HOT} and T_{COLD} in each loop, together with the measured pressurizer pressure, to compute the reactor core thermal power. Equations to fit the properties of density and enthalpy are programmed in the software, such that the ΔT power signal is presented as a percent of RTP for direct comparison with measured calorimetric power. The overtemperature ΔT setpoint is automatically varied for changes in the parameters that affect DNB as follows:

- reactor core inlet temperature – the Trip Setpoint is varied to correct for changes in core inlet temperature based on measured changes in cold leg temperature with dynamic compensation to account for cold leg-to-core transit time;

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APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

- pressurizer pressure – the Trip Setpoint is varied to correct for changes in system pressure; and

- axial power distribution – the Trip Setpoint is varied to account for imbalances in the axial power distribution as detected by the PMS upper and lower power range detectors. If axial peaks are greater than the design limit, as indicated by the difference between the upper and lower PMS power range detectors, the Trip Setpoint is reduced in accordance with Note 1 of Table 3.3.1-1.

Dynamic compensation of the ΔT power signal is included for system piping delays from the core to the temperature measurement system. The Overtemperature ΔT trip Function is calculated for each loop as described in Note 1 of Table 3.3.1-1. This Function also provides a signal to generate a turbine runback prior to reaching the Trip Setpoint. A turbine runback will reduce turbine power and reactor power. A reduction in power will normally alleviate the Overtemperature ΔT condition and may prevent a reactor trip. No credit is taken in the safety analyses for the turbine runback.

The LCO requires four channels of the Overtemperature ΔT trip Function to be OPERABLE in MODES 1 and 2. Four channels are provided to permit one channel in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function. Note that the Overtemperature ΔT Function receives input from channels shared with other RTS Functions. Failures that affect multiple Functions require entry into the Conditions applicable to all affected Functions.

In MODE 1 or 2, the Overtemperature ΔT trip must be OPERABLE to prevent DNB. In MODE 3, 4, 5, or 6, this trip Function does not have to be OPERABLE because the reactor is not operating and there is insufficient heat production to be concerned about DNB.

7. <u>Overpower ΔT</u>

The Overpower ΔT trip Function ensures that protection is provided to ensure the integrity of the fuel (i.e., no fuel pellet melting and less than 1% cladding strain) under all possible overpower conditions. This trip Function also limits the required range of the Overtemperature ΔT trip function and provides a backup to the Power Range Neutron Flux – High Setpoint trip. The Overpower ΔT

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APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

trip Function ensures that the allowable heat generation rate (kW/ft) of the fuel is not exceeded. It uses the same ΔT power signal generated for the Overtemperature ΔT. The setpoint is automatically varied with the following parameter:

- Axial power distribution – the Trip Setpoint is varied to account for imbalances in the axial power distribution as detected by the PMS upper and lower power range detectors. If axial peaks are greater than the design limit, as indicated by the difference between the upper and lower PMS power range detectors, the Trip Setpoint is reduced in accordance with Note 2 of Table 3.3.1-1.

The Overpower ΔT trip Function is calculated for each loop as per Note 2 of Table 3.3.1-1. The Trip Setpoint reflects the inclusion of both steady state and adverse environmental instrument uncertainties as the detectors provide protection for a steam line break and may be in a harsh environment. Note that this Function also provides a signal to generate a turbine runback prior to reaching the Trip Setpoint. A turbine runback reduces turbine power and reactor power. A reduction in power normally alleviates the Overpower ΔT condition and may prevent a reactor trip.

The LCO requires four channels of the Overpower ΔT trip Function to be OPERABLE in MODES 1 and 2. Four channels are provided to permit one channel in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function. The Overpower ΔT Function receives input from channels shared with other RTS Functions. Failures that affect multiple Functions require entry into the Conditions applicable to a affected Functions.

In MODE 1 or 2, the Overpower ΔT trip Function must be OPERABLE. These are the only times that enough heat is generated in the fuel to be concerned about the heat generation rates and overheating of the fuel. In MODE 3, 4, 5, or 6, this trip Function does not have to be OPERABLE because the reactor is not operating and there is insufficient heat production to be concerned about fuel overheating and fuel damage.

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APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

8. Pressurizer Pressure

The same sensors provide input to the Pressurizer Pressure – High and – Low trips and the Overtemperature ΔT trip.

a. Pressurizer Pressure – Low

The Pressurizer Pressure – Low trip Function ensures that protection is provided against violating the DNBR limit due to low pressure. The Trip Setpoint reflects both steady state and adverse environmental instrument uncertainties as the detectors provide primary protection for an event that results in a harsh environment.

The LCO requires four channels of Pressurizer Pressure – Low to be OPERABLE in MODE 1 above P-10. Four channels are provided to permit one channel in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function.

In MODE 1, when DNB is a major concern, the Pressurizer Pressure – Low trip must be OPERABLE. This trip Function is automatically enabled on increasing power by the P-10 interlock. On decreasing power, this trip Function is automatically blocked below P-10. Below the P-10 setpoint, no conceivable power distributions can occur that would cause DNB concerns.

b. Pressurizer Pressure – High

The Pressurizer Pressure – High trip Function ensures that protection is provided against overpressurizing the RCS. This trip Function operates in conjunction with the safety valves to prevent RCS overpressure conditions. The Trip Setpoint reflects only steady state instrument uncertainties as the detectors do not provide primary protection for any event that results in a harsh environment.

The LCO requires four channels of the Pressurizer Pressure – High to be OPERABLE in MODES 1 and 2. Four channels are provided to permit one channel in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function.

BASES

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

In MODE 1 or 2, the Pressurizer Pressure – High trip must be OPERABLE to help prevent RCS overpressurization and LCOs, and minimizes challenges to the safety valves. In MODE 3, 4, 5, or 6, the Pressurizer Pressure – High trip Function does not have to be OPERABLE because transients which could cause an overpressure condition will be slow to occur. Therefore, the operator will have sufficient time to evaluate plant conditions and take corrective actions. Additionally, low temperature overpressure protection systems provide overpressure protection when below MODE 4.

9. Pressurizer Water Level – High 3

The Pressurizer Water Level – High 3 trip Function provides a backup signal for the Pressurizer Pressure – High 3 trip and also provides protection against water relief through the pressurizer safety valves. These valves are designed to pass steam in order to achieve their design energy removal rate. A reactor trip is actuated prior to the pressurizer becoming water solid. The Trip Setpoint reflects only steady state instrument uncertainties as the detectors do not provide primary protection for any event that results in a harsh environment. The level channels do not actuate the safety valves.

The LCO requires four channels of Pressurizer Water Level – High 3 to be OPERABLE in MODE 1 above P-10. Four channels are provided to permit one channel in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function.

In MODE 1 when there is a potential for overfilling the pressurizer, the Pressurizer Water Level – High 3 trip must be OPERABLE. This trip Function is automatically enabled on increasing power by the P-10 interlock. On decreasing power, this trip Function is automatically blocked below P-10. Below the P-10 setpoint, transients which could raise the pressurizer water level will be slow and the operator will have sufficient time to evaluate plant conditions and take corrective actions.

10. Reactor Coolant Flow – Low

Reactor Coolant Flow

The Reactor Coolant Flow – Low trip Function ensures that protection is provided against violating the DNBR limit due to low flow in one or more RCS hot legs. Above the P-10 setpoint, a loss of flow in any RCS hot leg will actuate a Reactor trip.

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APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

Each RCS hot leg has four flow detectors to monitor flow. The Trip Setpoint reflects only steady state instrument uncertainties as the detectors do not provide primary protection for any event that results in a harsh environment.

The LCO requires four Reactor Coolant Flow – Low channels per hot leg to be OPERABLE in MODE 1 above P-10. Four OPERABLE channels are provided to permit one channel in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function.

In MODE 1 above the P-10 setpoint, when a loss of flow in one RCS hot leg could result in DNB conditions in the core, the Reactor Coolant Flow – Low trip must be OPERABLE.

11. Reactor Coolant Pump (RCP) Bearing Water Temperature – High

RCP Bearing Water Temperature – High

The RCP Bearing Water Temperature – High reactor trip Function ensures that protection is provided against violating the DNBR limit due to a loss of flow in one RCS cold leg. The Trip Setpoint reflects only steady state instrument uncertainties as the detectors do not provide primary protection for any event that results in a harsh environment.

The LCO requires four RCP Bearing Water Temperature – High channels per RCP to be OPERABLE in MODE 1 or 2. Four channels are provided to permit one channel in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function.

In MODE 1 or 2, when a loss of flow in any RCS cold leg could result in DNB conditions in the core, the RCP Bearing Water Temperature – High trip must be OPERABLE.

12. Reactor Coolant Pump Speed – Low

The RCP Speed – Low trip Function ensures that protection is provided against violating the DNBR limit due to a loss of flow in two or more RCS cold legs. The speed of each RCP is monitored. Above the P-10 setpoint a low speed detected on two or more RCPs will initiate a reactor trip. The Trip Setpoint reflects only steady state instrument uncertainties as the detectors do not provide primary protection for any event that results in a harsh environment.

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APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

The LCO requires four RCP Speed – Low channels to be OPERABLE in MODE 1 above P-10. Four channels are provided to permit one channel in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function.

In MODE 1 above the P-10 setpoint, the RCP Speed – Low trip must be OPERABLE. Below the P-10 setpoint, all reactor trips on loss of flow are automatically blocked since no power distributions are expected to occur that would cause a DNB concern at this low power level. Above the P-10 setpoint, the reactor trip on loss of flow in two or more RCS cold legs is automatically enabled.

13. Steam Generator Water Level – Low

The SG Water Level – Low trip Function ensures that protection is provided against a loss of heat sink. The SGs are the heat sink for the reactor. In order to act as a heat sink, the SGs must contain a minimum amount of water. A narrow range low level in any steam

generator is indicative of a loss of heat sink for the reactor. The Trip Setpoint reflects the inclusion of both steady state and adverse environmental instrument uncertainties as the detectors provide primary protection for an event that results in a harsh environment. This Function also contributes to the coincidence logic for the ESFAS Function of opening the Passive Residual Heat Removal (PRHR) discharge valves.

The LCO requires four channels of SG Water Level – Low per SG to be OPERABLE in MODES 1 and 2. Four channels are provided to permit one channel in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function.

In MODE 1 or 2, when the reactor requires a heat sink, the SG Water Level – Low trip must be OPERABLE. The normal source of water for the SGs is the Main Feedwater System (non-safety related). The Main Feedwater System is normally in operation in MODES 1 and 2. PRHR is the safety related backup heat sink for the reactor. During normal startups and shutdowns, the Main and Startup Feedwater Systems (non-safety related) can provide feedwater to maintain SG level. In MODE 3, 4, 5, or 6, the SG Water Level – Low Function does not have to be OPERABLE because the reactor is not operating or even critical.

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APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

14. Steam Generator Water Level – High 2

The SG Water Level – High 2 trip Function ensures that protection is provided against excessive feedwater flow by closing the main feedwater control valves, tripping the turbine, and tripping the reactor. While the transmitters (d/p cells) are located inside containment, the events which this function protects against cannot cause severe environment in containment. Therefore, the Trip Setpoint reflects only steady state instrument uncertainties.

The LCO requires four channels of SG Water Level – High 2 per SG to be OPERABLE in MODES 1 and 2. Four channels are provided to permit one channel in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function.

In MODES 1 and 2 above the P-11 interlock, the SG Water Level – High 2 trip must be OPERABLE. The normal source of water for the SGs is the Main Feedwater System (non-safety related). The Main Feedwater System is only in operation in MODES 1 and 2. In MODE 3, 4, 5, or 6, the SG Water Level – High 2 Function does not have to be OPERABLE because the reactor is not operating or even critical. The P-11 interlock is provided on this Function to permit bypass of the trip Function when the pressure is below P-11. This bypass is necessary to permit rod testing when the steam generators are in wet layup.

15. Safeguards Actuation Signal from Engineered Safety Feature Actuation System

The Safeguards Actuation Signal from ESFAS ensures that if a reactor trip has not already been generated by the RTS, the ESFAS automatic actuation logic will initiate a reactor trip upon any signal which initiates the Safeguards Actuation signal. This is a condition of acceptability for the Loss of Coolant Accident (LOCA). However, other transients and accidents take credit for varying levels of ESFAS performance and rely upon rod insertion, except for the most reactive rod which is assumed to be fully withdrawn, to ensure reactor shutdown.

The LCO requires two manual and four automatic divisions of Safeguards Actuation Signal Input from ESFAS to be OPERABLE in MODES 1 and 2. Four automatic divisions are provided to permit one division bypass indefinitely and still ensure no single random failure will disable this trip Function.

BASES

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

A reactor trip is initiated every time a Safeguards Actuation signal is present. Therefore, this trip Function must be OPERABLE in MODES 1 and 2, when the reactor is critical, and must be shutdown in the event of an accident. In MODE 3, 4, 5, or 6, the reactor is not critical.

16. Reactor Trip System Interlocks

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

a. Intermediate Range Neutron Flux, P-6

The Intermediate Range Neutron Flux, P-6 interlock is actuated when the respective PMS Intermediate Range Neutron Flux channel goes approximately one decade above the minimum channel reading. The LCO requirement for the P-6 interlock ensures that the following Functions are performed:

(1) on increasing power, the P-6 interlock allows the manual block of the respective PMS Source Range, Neutron Flux reactor trip. This prevents a premature block of the source range trip and allows the operator to ensure that the intermediate range is OPERABLE prior to leaving the source range. When the source range trip is blocked, the high voltage to the detectors is also removed.

(2) on decreasing power, the P-6 interlock automatically energizes the PMS source range detectors and enables the PMS Source Range Neutron Flux reactor trip.

(3) on increasing power, the P-6 interlock provides a backup block signal to the source range neutron flux doubling circuit. Normally, this Function is manually blocked by the main control room operator during the reactor startup.

The LCO requires four channels of Intermediate Range Neutron Flux, P-6 interlock to be OPERABLE in MODE 2 when below the P-6 interlock setpoint.

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APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

In MODE 2, when below the P-6 interlock setpoint, the P-6 interlock must be OPERABLE. Above the P-6 interlock setpoint, the PMS Source Range Neutron Flux reactor trip will be blocked; and this Function will no longer be necessary. In MODES 3, 4, 5, and 6, the P-6 interlock does not have to be OPERABLE because the PMS Source Range is providing core protection.

b. Power Range Neutron Flux, P-10

The Power Range Neutron Flux, P-10 interlock is actuated at approximately 10% power as determined by the respective PMS power-range detector. The LCO requirement for the P-10 interlock ensures that the following functions are performed:

(1) on increasing power, the P-10 interlock automatically enables reactor trips on the following Functions:

- Pressurizer Pressure – Low,
- Pressurizer Water Level – High 3,
- Reactor Coolant Flow – Low, and
- RCP Speed – Low.

These reactor trips are only required when operating above the P-10 setpoint (approximately 10% power). These reactor trips provide protection against violating the DNBR limit. Below the P-10 setpoint, the RCS is capable of providing sufficient natural circulation without any RCP running.

(2) on increasing power, the P-10 interlock allows the operator to manually block the Intermediate Range Neutron Flux reactor trip.

(3) on increasing power, the P-10 interlock allows the operator to manually block the Power Range Neutron Flux – Low Setpoint reactor trip.

(4) on increasing power, the P-10 interlock automatically provides a backup block signal to the Source Range Neutron Flux reactor trip and also to de-energize the PMS source range detectors.

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APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

(5) on decreasing power, the P-10 interlock automatically blocks reactor trips on the following Functions:

- Pressurizer Pressure – Low,
- Pressurizer Water Level – High 3,
- Reactor Coolant Flow – Low, and
- RCP Speed – Low.

(6) on decreasing power, the P-10 interlock automatically enables the Power Range Neutron Flux – Low reactor trip and the Intermediate Range Neutron Flux reactor trip (and rod stop).

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

In MODE 1, when the reactor is at power, the Power Range Neutron Flux, P-10 interlock must be OPERABLE. This Function must be OPERABLE in MODE 2 to ensure that core protection is provided during a startup or shutdown by the Power Range Neutron Flux – Low Setpoint and Intermediate

Range Neutron Flux reactor trips. In MODE 3, 4, 5, or 6, this Function does not have to be OPERABLE because the reactor is not at power and the Source Range Neutron Flux reactor trip provides core protection.

c. Pressurizer Pressure, P-11

With pressurizer pressure channels less than the P-11 setpoint, the operator can manually block the Steam Generator Narrow Range Water Level – High 2 reactor Trip. This allows rod testing with the steam generators in cold wet layup. With pressurizer pressure channels > P-11 setpoint, the Steam Generator Narrow Range Water Level – High 2 reactor Trip is automatically enabled. The operator can also enable these actuations by use of the respective manual reset.

BASES

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

17. Reactor Trip Breakers

This trip Function applies to the RTBs exclusive of individual trip mechanisms. There are eight reactor trip breakers with two breakers in each division. The reactor trip circuit breakers are arranged in a two-out-of-four logic configuration, such that the tripping of the two circuit breakers associated with one division does not cause a reactor trip. This circuit breaker arrangement is illustrated in Figure 7.1-7. The LCO requires four divisions of the Reactor Trip Switchgear to be OPERABLE with two trip breakers associated with each required division. This logic is required to meet the safety function assuming a single failure.

These trip Functions must be OPERABLE in MODE 1 or 2 when the reactor is critical. In MODE 3, 4, or 5, these RTS trip Functions must be OPERABLE when the RTBs are closed, and the PLS is capable of rod withdrawal.

18. Reactor Trip Breaker Undervoltage and Shunt Trip Mechanisms

The LCO requires both the Undervoltage and Shunt Trip Mechanisms to be OPERABLE for each RTB that is in service. The trip mechanisms are not required to be OPERABLE for trip breakers that are open, racked out, incapable of supplying power to the PLS, or declared inoperable under Function 17 above. OPERABILITY of both trip mechanisms on each breaker ensures that no single trip mechanism failure will prevent opening the breakers on a valid signal.

These trip Functions must be OPERABLE in MODES 1 and 2 when the reactor is critical. In MODES 3, 4, and 5, these RTS trip Functions must be OPERABLE when the RTBs are closed, and the PLS is capable of rod withdrawal.

19. Automatic Trip Logic

The LCO requirement for the RTBs (Functions 17 and 18) and Automatic Trip Logic (Function 19) ensures that means are provided to interrupt the power to the CRDMs and allow the rods to fall into the reactor core. Each RTB is equipped with an undervoltage coil and a shunt trip coil to trip the breaker open when needed.

The automatic trip logic includes the ESF coincidence logic and the voting logic.

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APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

The LCO requires four divisions of RTS Automatic Trip Logic to be OPERABLE. Four OPERABLE divisions are provided to ensure that a random failure of a single logic channel will not prevent reactor trip.

These trip Functions must be OPERABLE in MODE 1 or 2 when the reactor is critical. In MODE 3, 4, or 5, these RTS trip Functions must be OPERABLE when the RTBs are closed and the PLS is capable of rod withdrawal.

20. ADS Stages 1, 2 and 3 Actuation Input from Engineered Safety Feature Actuation System

The LCO requirement for this Function provides a reactor trip for any event that may initiate depressurization of the reactor.

The LCO requires four divisions of RTS Automatic Trip Logic to be OPERABLE. Four OPERABLE divisions are provided to ensure that a random failure of a single logic channel will not prevent reactor trip.

These trip Functions must be OPERABLE in MODE 1 or 2 when the reactor is critical. In MODE 3, 4, or 5, these RTS trip Functions must be OPERABLE when the RTBs are closed and the PLS is capable of rod withdrawal.

21. Core Makeup Tank (CMT) Actuation Input from Engineered Safety Feature Actuation System

The LCO requirement for this Function provides a reactor trip for any event that may initiate CMT injection.

The LCO requires four divisions of RTS Automatic Trip Logic to be OPERABLE. Four OPERABLE divisions are provided to ensure that random failure of a single logic channel will not prevent reactor trip.

These trip Functions must be OPERABLE in MODES 1 and 2 when the reactor is critical. In MODE 3, 4, and 5 these RTS trip Functions must be OPERABLE when the RTBs are closed and the PLS is capable of rod withdrawal.

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

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

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ACTIONS

In the event the transmitter, instrument loop, signal processing electronics, or trip output is found inoperable, then all affected Functions provided by that channel must be declared inoperable and the LCO Condition(s) entered for the protection Function(s) affected.

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

A.1

Condition A applies to all RTS protection Functions. Condition A addresses the situation where one or more required channels for one or more Functions are inoperable at the same time. The Required Action is to refer to Table 3.3.1-1 and to take the Required Actions for the protection Functions affected. The Completion Times are those from the referenced Conditions and Required Actions.

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

Condition B applies to the Manual Reactor Trip, Manual Safeguards Actuation, Manual ADS Stages 1, 2, and 3 Actuation and Manual Core Makeup Tank Actuation in MODES 1 and 2 and in MODES 3, 4, and 5 with the reactor trip breakers closed and the plant control system capable of rod withdrawal. These Required Actions address inoperability of one manual initiation device of the Manual Reactor Trip Function, Manual Safeguards Actuation Function, Manual ADS Stages 1, 2, and 3 Actuation Function and/or Manual Core Makeup Tank Actuation Function. One device consists of an actuation switch and the associated hardware (such as contacts and wiring) up to but not including the eight Reactor Trip Breakers. With one device inoperable, the inoperable device must be restored to OPERABLE status within 48 hours. In this Condition, the remaining OPERABLE device is adequate to perform the safety function.

If the manual Function(s) cannot be restored to OPERABLE status in the allowed 48 hour Completion Time, the unit must be brought to a MODE in which the requirement does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 additional hours (54 hours total time) followed by opening the RTBs within 1 additional hour (55 hours total time). The 6 additional hours to reach MODE 3 and the

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1 hour to open the RTBs are reasonable, based on operating experience, to reach MODE 3 and open the RTBs from full power operation in an orderly manner and without challenging unit systems. With the RTBs open and the unit in MODE 3, this trip Function is no longer required to be OPERABLE.

C.1 and C.2

Condition C applies to the Manual Reactor Trip in MODES 3, 4, and 5 with the RTBs closed and the PLS capable of rod withdrawal. These Required Actions address inoperability of one manual initiation device of the Manual Reactor Trip Function. One device consists of an actuation switch and the associated hardware (such as contacts and wiring) up to but not including the eight Reactor Trip Breakers. With one device inoperable, the inoperable device must be restored to OPERABLE status within 48 hours. In this Condition, the remaining OPERABLE device is adequate to perform the safety function.

If the Manual Reactor Trip Function cannot be restored to OPERABLE status in the allowed 48 hour Completion Time, the unit must be placed in a MODE in which the requirement does not apply. To achieve this status, the RTBs must be opened within the next 1 hour. With the RTBs open, this Function is no longer required.

D.1.1, D.1.2, D.1.3, D.2.1, D.2.2, and D.3

Condition D applies to the Power Range Neutron Flux – High Function in MODES 1 and 2.

With one or two channels inoperable, one affected channel must be placed in a bypass or trip condition within 6 hours. If one channel is bypassed, the logic becomes two-out-of-three, while still meeting the single failure criterion. (A failure in one of the three remaining channels will not prevent the protective function.) If one channel is tripped, the logic becomes one-out-of-three, while still meeting the single failure criterion. (A failure in one of the three remaining channels will not prevent the protective function.) If one channel is bypassed and one channel is tripped, the logic becomes one-out-of-two, while still meeting the single failure criterion. The 6 hours allowed to place the inoperable channel(s) in the bypassed or tripped condition is justified in Reference 7.

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ACTIONS (continued)

In addition to placing the inoperable channel(s) in the bypassed or tripped condition, THERMAL POWER must be reduced to ≤ 75% RTP within 12 hours. Reducing the power level prevents operation of the core with radial power distributions beyond the design limits. With one or two of the PMS power range detectors inoperable, partial radial power distribution monitoring capability is lost. However, the protective function would still function even with a single failure of one of the two remaining channels.

As an alternative to reducing power, the inoperable channel(s) can be placed in the bypassed or tripped condition within 6 hours and the QPTR monitored every 12 hours as per SR 3.2.4.2, QPTR verification. Calculating QPTR compensates for the lost monitoring capability and allows continued plant operation at power levels > 75% RTP. The 12 hour Frequency is consistent with LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)."

Required Action D.2.2 has been modified by a Note which only requires SR 3.2.4.2 to be performed if OPDMS and the Power Range Neutron Flux input to QPTR become inoperable. Power distribution limits are normally verified in accordance with LCO 3.2.5, "OPDMS - Monitored Power Distribution Parameters." However, if OPDMS becomes inoperable, then LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," becomes applicable. Failure of a component in the Power Range Neutron Flux Channel which renders the High Flux Trip Function inoperable may not affect the capability to monitor QPTR. If either OPDMS or the channel input to QPTR is OPERABLE, then performance of SR 3.2.4.2 once per 12 hours is not necessary.

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

E.1.1, E.1.2, and E.2

Condition E applies to the following reactor trip Functions:

- Power Range Neutron Flux – Low;
- Overtemperature ΔT;

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ACTIONS (continued)

- Overpower ΔT;
- Power Range Neutron Flux – High Positive Rate;
- Pressurizer Pressure – High;
- RCP Bearing Water Temperature – High;
- SG Water Level – Low; and
- SG Water Level – High 2.

With one or two channels inoperable, one affected channel must be placed in a bypass or trip condition within 6 hours. If one channel is bypassed, the logic becomes two-out-of-three, while still meeting the single failure criterion. (A failure in one of the three remaining channels will not prevent the protective function.) If one channel is tripped, the logic becomes one-out-of-three, while still meeting the single failure criterion. (A failure in one of the three remaining channels will not prevent the protective function.) If one channel is bypassed and one channel is tripped, the logic becomes one-out-of-two, while still meeting the single failure criterion. The 6 hours allowed to place the inoperable channel(s) in the bypassed or tripped condition is justified in Reference 7.

If the Required Actions described above cannot be met within the specified Completion Times, the unit must be placed in a MODE where this Function is no longer required to be OPERABLE. An additional 6 hours is allowed to place the unit in MODE 3. Six hours is a reasonable time, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging plant systems.

F.1.1, F.1.2, F.2, and F.3

Condition F applies to the Intermediate Range Neutron Flux trip when above the P-6 setpoint and below the P-10 setpoint. Above the P-6 setpoint and below the P-10 setpoint, the PMS intermediate range detector performs the monitoring functions.

With one or two channels inoperable, one affected channel must be placed in a bypass or trip condition within 2 hours. If one channel is bypassed, the logic becomes two-out-of-three, while still meeting the single failure criterion. (A failure in one of the three remaining channels will not prevent the protective function.) If one channel is tripped, the logic becomes one-out-of-three, while still meeting the single failure criterion. (A failure in one of the three remaining channels will not prevent

BASES

ACTIONS (continued)

the protective function.) If one channel is bypassed and one channel is tripped, the logic becomes one-out-of-two, while still meeting the single failure criterion. The 2 hours allowed to place the inoperable channel(s) in the bypassed or tripped condition is justified in Reference 7.

As an alternative to placing the channel(s) in bypass or trip if THERMAL POWER is greater than the P-6 setpoint but less than the P-10 setpoint, 2 hours are allowed to reduce THERMAL POWER below the P-6 setpoint or to increase the THERMAL POWER above the P-10 setpoint. The PMS Intermediate Range Neutron Flux channels must be OPERABLE when the power level is above the capability of the source range, P-6, and below the capability of the power range, P-10. If THERMAL POWER is greater than the P-10 setpoint, the PMS power range detectors perform the monitoring and protective functions and the intermediate range is not required. The Completion Times allow for a slow and controlled power adjustment below P-6, and takes into account the redundant capability afforded by the two remaining OPERABLE channels and the low probability of their failure during this period.

G.1 and G.2

Condition G applies to three Intermediate Range Neutron Flux trip channels inoperable in MODE 2 above the P-6 setpoint and below the P-10 setpoint. Required Actions specified in this Condition are only applicable when channel failures do not result in reactor trip. Above the P-6 setpoint and below the P-10 setpoint, the PMS intermediate range detector performs the monitoring Functions. With only one intermediate range channel OPERABLE, the Required Actions are to suspend operations involving positive reactivity additions immediately. This will preclude any power level increase since there are insufficient

OPERABLE Intermediate Range Neutron Flux channels to adequately monitor the power rise. The operator must also reduce THERMAL POWER below the P-6 setpoint within 2 hours. Below P-6, the Source Range Neutron Flux channels will be able to monitor the core power level. The Completion Time of 2 hours will allow a slow and controlled power reduction to less than the P-6 setpoint and takes into account the low probability of occurrence of an event during this period that may require the protection afforded by the PMS Intermediate Range Neutron Flux trip.

BASES

ACTIONS (continued)

H.1

Condition H applies to the Intermediate Range Neutron Flux trip when THERMAL POWER is below the P-6 setpoint and one or two channels is inoperable. Below the P-6 setpoint, the PMS source range performs the monitoring and protective functions. At least three of the four PMS intermediate range channels must be returned to OPERABLE status prior to increasing power above the P-6 setpoint. With the unit in this Condition, below P-6, the PMS source range performs the monitoring and protection functions.

I.1

Condition I applies to one or two Source Range Neutron Flux trip channels inoperable when in MODE 2, below the P-6 setpoint, and performing a reactor startup. With the unit in this Condition, below P-6, the PMS source range performs the monitoring and protection functions. With one or two of the four channels inoperable, operations involving positive reactivity additions shall be suspended immediately.

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

J.1

Condition J applies to three inoperable Source Range Neutron Flux channels when in MODE 2, below the P-6 setpoint, and performing a reactor startup, or in MODE 3, 4, or 5 with the RTBs closed and the CRD System capable of rod withdrawal. With the unit in this Condition, below P-6, the NIS source range performs the monitoring and protection functions. With three source range channels inoperable, the RTBs must be opened immediately. With the RTBs open, the core is in a more stable condition and the unit enters Condition T.

BASES

ACTIONS (continued)

K.1.1, K.1.2, and K.2

Condition K applies to the following reactor trip Functions:

- Pressurizer Pressure – Low;
- Pressurizer Water Level – High 3;
- Reactor Coolant Flow – Low; and
- RCP Speed – Low.

With one or two channels inoperable, one affected channel must be placed in a bypass or trip condition within 6 hours. If one channel is bypassed, the logic becomes two-out-of-three, while still meeting the single failure criterion. (A failure in one of the three remaining channels will not prevent the protective function.) If one channel is tripped, the logic becomes one-out-of-three, while still meeting the single failure criterion. (A failure in one of the three remaining channels will not prevent the protective function.) If one channel is bypassed and one channel is tripped, the logic becomes one-out-of-two, while still meeting the single failure criterion. The 6 hours allowed to place the inoperable channel(s) in the bypassed or tripped condition is justified in Reference 7.

If Required Actions described above cannot be met within the specified Completion Times, the unit must be placed in a MODE where this Function is no longer required to be OPERABLE. A Completion Time of an additional 6 hours is allowed to reduce power < P-10. Allowance of this time interval takes into consideration the redundant capability provided by the remaining two redundant OPERABLE channels and the low probability of occurrence of an event during this period that may require the protection afforded by the Functions associated with Condition K.

L.1 and L.2

Condition L applies to the Safeguards Actuation signal from ESFAS reactor trip, the RTS Automatic Trip Logic, automatic ADS Stages 1, 2, and 3 actuation, and automatic CMT injection in MODES 1 and 2.

With one or two channels or divisions inoperable, the Required Action is to restore three of the four channels/divisions within 6 hours. Restoring all channels/divisions but one to OPERABLE status ensures that a single failure will neither cause nor prevent the protective function. The 6 hour Completion Time is considered reasonable since the protective function will still function.

BASES

ACTIONS (continued)

If Required Actions described above cannot be met within the specified Completion Times, the unit must be placed in a MODE where this Function is no longer required to be OPERABLE. A Completion Time of an additional 6 hours is allowed to place the unit in MODE 3. The Completion Time is a reasonable time, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging plant systems. Allowance of this time interval takes into consideration the redundant capability provided by the remaining two redundant OPERABLE channels/divisions and the low probability of occurrence of an event during this period that may require the protection afforded by this Function.

M.1, M.2.1, M.2.2, and M.3

Condition M applies to the P-6, P-10, and P-11 interlocks. With one or two channels inoperable, the associated interlock must be verified to be in its required state for the existing plant condition within 1 hour, or the Functions associated with inoperable interlocks placed in a bypassed or tripped condition within 7 hours, or the unit must be placed in MODE 3 within 13 hours. Verifying the interlock manually accomplishes the interlock condition.

If one interlock channel is inoperable, the associated Function(s) must be placed in a bypass or trip condition within 7 hours. If one channel is bypassed, the logic becomes two-out-of-three, while still meeting the single failure criterion. (A failure in one of the three remaining channels will not prevent the protective function.) If one channel is tripped, the logic becomes one-out-of-three, while still meeting the single failure criterion. (A failure in one of the three remaining channels will not prevent the protective function.)

If two interlock channels are inoperable, one channel of the associated Function(s) must be bypassed and one channel of the associated Function(s) must be tripped. In this state, the logic becomes one-out-of-two, while still meeting the single failure criterion. The 7 hours allowed to place the inoperable channel(s) in the bypassed or tripped condition is justified in Reference 7.

If placing the associated Functions in bypass or trip is impractical, for instance as the result of other channels in bypass or trip, the Completion Time of an additional 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging plant systems.

N.1, N.2.1, and N.2.2

Condition N applies to the RTBs, and RTB undervoltage and shunt trip mechanisms in MODES 1 and 2, and in MODES 3, 4, and 5 with the

BASES

ACTIONS (continued)

RTBs closed and the PLS capable of rod withdrawal. This Condition is primarily associated with mechanical damage that can prevent the RTBs from opening.

With one division inoperable, the reactor trip breakers in the inoperable division must be opened within 8 hours. A division is inoperable, if, within that division, one or both of the RTBs and/or one or both of the trip mechanisms is inoperable.

With one division inoperable (with its RTBs open) and with three OPERABLE divisions remaining, the trip logic becomes one-out-of-three. The one-out-of-three trip logic meets the single failure criterion. (A failure in one of the three remaining divisions will not prevent the protective function.) If, coincident with RTBs inoperable in one division, the automatic trip logic is inoperable in another division, the trip logic becomes one-out-of-two, which meets the single failure criterion.

If Required Actions described above cannot be met within the specified Completion Times, the unit must be placed in a MODE where this Function is no longer required to be OPERABLE within an additional 6 hours. This is done by opening all of the RTBs. With the RTBs open, these Functions are no longer required.

O.1, O.2.1, and O.2.2

Condition O applies to the RTBs in MODES 1 and 2, and in MODES 3, 4, and 5 with the RTBs closed and the PLS capable of rod withdrawal. With two divisions of RTBs and/or RTB Undervoltage and Shunt Trip Mechanisms inoperable, 1 hour is allowed to restore the three of the four divisions to OPERABLE status or the unit must be placed in MODE 3, 4 or 5 and the RTBs opened within the next 6 hours. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging unit systems. The 1-hour and 6-hour Completion Times are equal to the time allowed by LCO 3.0.3 for shutdown actions in the event of a complete loss of RTS Function. Placing the unit in MODE 3 removes the requirement for this particular Function.

P.1 and P.2

Condition P applies to automatic ADS Stages 1, 2, and 3 Actuation, automatic CMT Actuation and the RTS Automatic Trip Logic in MODES 3, 4, and 5 with the RTBs closed and the PLS capable of rod withdrawal.

With one or two channels/divisions inoperable, three of the four channels/divisions must be restored to OPERABLE status in 48 hours. Restoring all channels but one to OPERABLE ensures that a single failure will neither cause nor prevent the protective function. The 48 hour

BASES

ACTIONS (continued)

Completion Time is considered reasonable since the protective function will still function.

If Required Actions described above cannot be met within the specified Completion Times, the unit must be placed in a MODE where this Function is no longer required to be OPERABLE. A Completion Time of an additional 1 hour is allowed to open the RTBs. With RTBs open, these Functions are no longer required.

Q.1 and Q.2

Condition Q applies to one or two inoperable Source Range Neutron Flux channels in MODE 3, 4, or 5 with the RTBs closed and the PLS capable of rod withdrawal. With the unit in this Condition, below P-6, the NIS source range performs the monitoring and protection functions. With one or two of the source range channels inoperable, 48 hours is allowed to restore three of the four channels to an OPERABLE status. If the channels cannot be returned to an OPERABLE status, 1 additional hour is allowed to open the RTBs. Once the RTBs are open, the core is in a more stable condition and the unit enters Condition R. The allowance of 48 hours to restore the channel to OPERABLE status, and the additional hour to open the RTBs, are justified in Reference 7.

R.1, R.2, and R.3

Condition R applies when the required Source Range Neutron Flux channel is inoperable in MODE 3, 4, or 5 with the RTBs open. With the unit in this Condition, the NIS source range performs the monitoring and protection functions. With the required source range channel inoperable, operations involving positive reactivity additions shall be suspended immediately. This will preclude any power escalation. In addition to suspension of positive reactivity additions, all valves that could add unborated water to the RCS must be closed within 1 hour as specified in LCO 3.9.2. The isolation of unborated water sources will preclude a boron dilution accident.

Also, the SDM must be verified within 1 hour and once every 12 hours thereafter as per SR 3.1.1.1, SDM verification. With no source range channels OPERABLE, core protection is severely reduced. Verifying the SDM within 1 hour allows sufficient time to perform the calculations and determine that the SDM requirements are met. The SDM must also be verified once per 12 hours thereafter to ensure that the core reactivity has not changed. Required Action R.3 precludes any positive reactivity additions; therefore, core reactivity should not be increasing, and a 12 hour Frequency is adequate. The Completion Times of within 1 hour and once per 12 hours are based on operating experience in performing the Required Actions and the knowledge that unit conditions will change slowly.

BASES

SURVEILLANCE REQUIREMENTS

The SRs for each RTS Function are identified in the SRs column of Table 3.3.1-1 for that Function.

A Note has been added to the SR table stating that Table 3.3.1-1 determines which SRs apply to which RTS Functions.

The CHANNEL CALIBRATION and RTCOT are performed in a manner that is consistent with the assumptions used in analytically calculating the required channel accuracies. For channels that include dynamic transfer functions, such as, lag, lead/lag, rate/lag, the response time test may be performed with the transfer function set to one, with the resulting measured response time compared to the appropriate Chapter 7 response time (Ref. 2). Alternately, the response time test can be performed with the time constants set to their nominal value provided the required response time is analytically calculated assuming the time constants are set at their nominal values. The response time may be measured by a series of overlapping tests such that the entire response time is measured.

SR 3.3.1.1

Performance of the CHANNEL CHECK once every 12 hours ensures that gross failure of instrumentation has not occurred. A CHANNEL CHECK is 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 even something more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment have drifted outside its limit.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The channels to be checked are:

Power Range Neutron Flux
Intermediate Range Neutron Flux
Source Range Neutron Flux
Overtemperature Delta T
Overpower Delta T
Pressurizer Pressure
Pressurizer Water Level
Reactor Coolant Flow – each hot leg
RCP Bearing Water Temperature – each RCP
RCP Speed
SG Narrow Range Level – each SG
RCS Loop T-cold – each cold leg
RCS Loop T-hot – each cold leg

The Frequency is based on operating experience that demonstrates the channel failure is rare. Automated operator aids may be used to facilitate the performance of the CHANNEL CHECK.

SR 3.3.1.2

SR 3.3.1.2 compares the calorimetric heat balance to the nuclear instrumentation channel output every 24 hours. If the calorimetric measurement between 70% and 100% RTP, differs from the nuclear instrument channel output by > 1% RTP, the nuclear instrument channel is not declared inoperable, but must be adjusted. If the nuclear instrument channel output cannot be properly adjusted, the channel is declared inoperable.

Three Notes modify SR 3.3.1.2. The first Note indicates that the nuclear instrument channel output shall be adjusted consistent with the calorimetric results if the absolute difference between the nuclear instrument channel output and the calorimetric measurement between 70% and 100% RTP is > 1% RTP. The second Note clarifies that this Surveillance is required only if reactor power is ≥ 15% RTP and that 12 hours is allowed for performing the first Surveillance after reaching 15% RTP. At lower power levels the calorimetric data are inaccurate. The third Note is required because, at power levels between 15% and 70% calorimetric uncertainty and control rod insertion create the potential for miscalibration of the nuclear instrumentation channel in cases where the channel is adjusted downward to match the calorimetric power. Therefore, if the calorimetric heat measurement is less than 70% RTP, and if the nuclear instrumentation channel indicated power is lower than the calorimetric measurement by > 1%, then the nuclear instrumentation

BASES

SURVEILLANCE REQUIREMENTS (continued)

channel shall be adjusted upward to match the calorimetric measurement. No nuclear instrumentation channel adjustment is required if the nuclear instrumentation channel is higher than the calorimetric measurement (see Westinghouse Technical Bulletin NSD-TB-92-14, Rev. 1.)

The Frequency of every 24 hours is adequate. It is based on plant operating experience, considering instrument reliability and operating history data for instrument drift.

Together these factors demonstrate the change in the absolute difference between nuclear instrumentation and heat balance calculated powers rarely exceeds 1% RTP in any 24 hours period.

In addition, main control room operators periodically monitor redundant indications and alarms to detect deviations in channel outputs.

SR 3.3.1.3

SR 3.3.1.3 compares the calorimetric heat balance to the calculated ΔT power ($q_{\Delta T}$) in each Division every 24 hours. If the calorimetric measurement between 70% and 100% RTP, differs from the calculated ΔT power by > 1% RTP, the Function is not declared inoperable, but the conversion factor, ΔT°, must be adjusted. If ΔT° cannot be properly adjusted, the Function is declared inoperable in the affected Division(s).

Three Notes modify SR 3.3.1.3. The first Note indicates that ΔT° shall be adjusted consistent with the calorimetric results if the absolute difference between the calculated ΔT power and the calorimetric measurement between 70% and 100% RTP is > 1% RTP.

The second Note clarifies that this Surveillance is required only if reactor power is ≥ 50% RTP and that 12 hours is allowed for performing the first Surveillance after reaching 50% RTP. At lower power levels, the calorimetric data are inaccurate. The calculated ΔT power is normally stable (less likely to need adjustment or to be grossly affected by changes in the core loading pattern than the nuclear instrumentation), and its calibration should not be unnecessarily altered by a possibly inaccurate calorimetric measurement at low power.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The third Note is required because at power levels below 70%, calorimetric uncertainty creates the potential for non-conservative adjustment of the ΔT° conversion factor, in cases where the calculated ΔT power would be reduced to match the calorimetric power. Therefore, if the calorimetric heat measurement is less than 70% RTP, and if the calculated ΔT power is lower than the calorimetric measurement by > 5%, then the ΔT° conversion factor shall be adjusted so that the calculated ΔT power matches the calorimetric measurement. No ΔT° conversion factor adjustment is required if the calculated ΔT power is higher than the calorimetric measurement.

The Frequency of every 24 hours is based on plant operating experience, considering instrument reliability and the limited effects of fuel burnup and rod position changes on the accuracy of the calculated ΔT power.

SR 3.3.1.4

SR 3.3.1.4 compares the AXIAL FLUX DIFFERENCE determined using the incore system to the nuclear instrument channel AXIAL FLUX DIFFERENCE every 31 EFPD.

If the absolute difference is ≥ 3% AFD the nuclear instrument channel is still OPERABLE, but must be readjusted. If the nuclear instrument channel cannot be properly readjusted, the channel is declared inoperable. This surveillance is performed to verify the f(ΔI) input to the overtemperature ΔT function.

Two Notes modify SR 3.3.1.4. The first Note indicates that the excore nuclear instrument channel shall be adjusted if the absolute difference between the incore and excore AFD is ≥ 3% AFD. Note 2 clarifies that the Surveillance is required only if reactor power is ≥ 20% RTP and that 24 hours is allowed for performing the first Surveillance after reaching 20% RTP. Below 20% RTP, the design of the incore detector system, low core power density, and detector accuracy make use of the incore detectors inadequate for use as a reference standard for comparison to the excore channels.

The Frequency of every 31 EFPD is adequate. It is based on plant operating experience, considering instrument reliability and operating history data for instrument drift. Also, the slow changes in neutron flux during the fuel cycle can be detected during this interval.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.5

SR 3.3.1.5 is a calibration of the excore channels to the incore channels. If the measurements do not agree, the excore channels are not declared inoperable but must be calibrated to agree with the incore detector measurements. If the excore channels cannot be adjusted, the channels are declared inoperable. This Surveillance is performed to verify the f(ΔI) input to the overtemperature ΔT Function.

A Note modifies SR 3.3.1.5. The Note states that this Surveillance is required only if reactor power is > 50% RTP and that 24 hours is allowed for performing the first surveillance after reaching 50% RTP.

The Frequency of 92 EFPD is adequate. It is based on industry operating experience, considering instrument reliability and operating history data for instrument drift.

SR 3.3.1.6

SR 3.3.1.6 is the performance of a TADOT every 92 days on a STAGGERED TEST BASIS. This test shall verify OPERABILITY by actuation of the end devices.

The Reactor Trip Breaker (RTB) test shall include separate verification of the undervoltage and shunt trip mechanisms. Each RTB in a division shall be tested separately in order to minimize the possibility of an inadvertent trip.

The Frequency of every 92 days on a STAGGERED TEST BASIS is adequate. It is based on industry operating experience, considering instrument reliability and operating history data. In addition, the AP1000 design provides additional breakers to enhance reliability.

The SR is modified by a Note to clarify that both breakers in a single division are to be tested during each STAGGERED TEST.

SR 3.3.1.7

SR 3.3.1.7 is the performance of a REACTOR TRIP CHANNEL OPERATIONAL TEST (RTCOT) every 92 days.

A RTCOT is performed on each required channel to provide reasonable assurance that the entire channel will perform the intended Function.

BASES

SURVEILLANCE REQUIREMENTS (continued)

A test subsystem is provided with the protection and safety monitoring system to aid the plant staff in performing the RTCOT. The test subsystem is designed to allow for complete functional testing by using a combination of system self checking features, functional testing features, and other testing features. Successful functional testing consists of verifying that the capability of the system to perform the safety function has not failed or degraded.

For hardware functions this would involve verifying that the hardware components and connections have not failed or degraded. Generally this verification includes a comparison of the outputs from two or more redundant subsystems or channels.

Since software does not degrade, software functional testing involves verifying that the software code has not changed and that the software code is executing.

To the extent possible, protection and safety monitoring system functional testing is accomplished with continuous system self-checking features and the continuous functional testing features. The RTCOT shall include a review of the operation of the test subsystem to verify the completeness and adequacy of the results.

If the RTCOT can not be completed using the built-in test subsystem, either because of failures in the test subsystem or failures in redundant channel hardware used for functional testing, the RTCOT can be performed using portable test equipment.

This test frequency of 92 days is justified based on Reference 7 and the use of continuous diagnostic test features, such as deadman timers, cross-check of redundant channels, memory checks, numeric coprocessor checks, and tests of timers, counters and crystal time bases, which will report a failure within the protection and safety monitoring system cabinets to the operator within 10 minutes of a detectable failure.

SR 3.3.1.7 is modified by a note that provides a 4 hour delay in the requirement to perform this Surveillance for source range instrumentation when entering MODE 3 from MODE 2. This note allows a normal shutdown to proceed without a delay for testing in MODE 2 and for a short time in MODE 3 until the RTBs are open and SR 3.3.1.7 is no longer required to be performed. If the unit is to be in MODE 3 with the RTBs closed for a time greater than 4 hours, this Surveillance must be performed prior to 4 hours after entry into MODE 3.

During the RTCOT, the protection and safety monitoring system cabinets in the division under test may be placed in bypass.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.8

SR 3.3.1.8 is the performance of a RTCOT as described in SR 3.3.1.7, except it is modified by a Note that this test shall include verification that the P-6 and P-10 interlocks are in their required state for the existing unit condition. The Frequency is modified by a Note that allows this surveillance to be satisfied if it has been performed within 92 days of the Frequencies prior to reactor startup and four hours after reducing power below P-10 and P-6. The Frequency of “prior to startup” ensures this surveillance is performed prior to critical operations and applies to the source, intermediate and power range low instrument channels. The Frequency of “4 hours after reducing power below P-10” (applicable to intermediate and power range low channels) and “4 hours after reducing power below P-6” (applicable to source range channels) allows a normal shutdown to be completed and the unit removed from the MODE of Applicability for this surveillance without a delay to perform the testing required by this surveillance. The Frequency of every 92 days thereafter applies if the plant remains in the MODE of Applicability after the initial performances of prior to reactor startup and four hours after reducing power below P-10 or P-6. The MODE of Applicability for this surveillance is < P-10 for the power range low and intermediate range channels and < P-6 for the source range channels. Once the unit is in MODE 3, this surveillance is no longer required. If power is to be maintained < P-10 or < P-6 for more than 4 hours, then the testing required by this surveillance must be performed prior to the expiration of the 4 hour limit. Four hours is a reasonable time to complete the required testing or place the unit in a MODE where this surveillance is no longer required. This test ensures that the NIS source, intermediate, and power range low channels are OPERABLE prior to taking the reactor critical and after reducing power into the applicable MODE (< P-10 or < P-6) for periods > 4 hours.

SR 3.3.1.9

A CHANNEL CALIBRATION is performed every 24 months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy.

Transmitter calibration must be performed consistent with the assumptions of the unit specific setpoint methodology. The difference between the current “as found” values and the previous test “as left” values must be consistent with the transmitter drift allowance used in the setpoint methodology.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The CHANNEL CALIBRATION is assisted by the use of a tester.

The setpoint methodology requires that 30 months drift be used (1.25 times the surveillance calibration interval, 24 months) based on Generic Letter 91-04, “Changes in Technical Specification Surveillance Intervals to Accommodate a 24-month Fuel Cycle.”

SR 3.3.1.9 is modified by a Note stating that this test shall include verification that the time constants are adjusted to the prescribed values where applicable.

SR 3.3.1.10

SR 3.3.1.10 is the performance of a CHANNEL CALIBRATION every 24 months. This SR is modified by a Note stating that neutron detectors are excluded from the CHANNEL CALIBRATION. The CHANNEL CALIBRATION for the power range neutron detectors consists of a normalization of the detectors based on a power calorimetric and flux map performed above 20% RTP. Below 20% RTP, the design of the incore detector system, low core power density, and detector accuracy make use of the incore detectors inadequate for use as a reference standard for comparison to the excore channels. The CHANNEL CALIBRATION for the source range and intermediate range neutron detectors consists of obtaining the detector plateau or preamp discriminator curves, evaluating those curves, and comparing the curves to the manufacturer’s data. This Surveillance is not required for the power range detectors for entry into MODES 2 and 1, and is not required for the intermediate range detectors for entry into MODE 2, because the plant must be in at least MODE 2 to perform the test for the intermediate range detectors and MODE 1 for the power range detectors. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed on the 24 month Frequency.

SR 3.3.1.11

SR 3.3.1.11 is the performance of a TADOT of the Manual Reactor Trip, and the SI, ADS Actuation, and CMT Injection inputs from the ESF logic. This TADOT is performed every 24 months. The test shall independently verify the OPERABILITY of the undervoltage and shunt trip mechanisms for the Manual Reactor Trip Function for the Reactor Trip Breakers.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The Frequency is based on the known reliability of the Functions and the multichannel redundancy available, and has been shown to be acceptable through operating experience.

The SR is modified by a Note that excludes verification of setpoints from the TADOT. The Functions affected have no setpoints associated with them.

SR 3.3.1.12

This SR 3.3.1.12 verifies that the individual channel/division actuation response times are less than or equal to the maximum values assumed in the accident analysis. Response Time testing criteria are included in Reference 2.

For channels that include dynamic transfer Functions (e.g., lag, lead/lag, rate/lag, etc.), the response time test may be performed with the transfer Function set to one, with the resulting measured response time compared to the appropriate FSAR response time. Alternately, the response time test can be performed with the time constants set to their nominal value, provided the required response time is analytically calculated assuming the time constants are set at their nominal values. The response time may be measured by a series of overlapping test such that the entire response time is measured.

Response time may be verified by actual response time tests in any series of sequential, overlapping or total channel measurements, or by the summation of allocated sensor, signal processing and actuation logic response times with actual response time tests on the remainder of the channel. Allocations for sensor response times may be obtained from: (1) historical records based on acceptable response time tests (hydraulic, noise, or power interrupt tests), (2) in place, onsite, or offsite (e.g. vendor) test measurements, or (3) utilizing vendor engineering specifications. WCAP-13632-P-A, Revision 2, “Elimination of Pressure Sensor Response Time Testing Requirements” (Ref. 10), provides the basis and methodology for using allocated sensor response times in the overall verification of the channel response time for specific sensors identified in the WCAP. Response time verification for other sensor types must be demonstrated by test.

Each division response must be verified every 24 months on a STAGGERED TEST BASIS (i.e., all four Protection Channel Sets would be tested after 96 months). Response times cannot be determined during plant operation because equipment operation is required to measure

BASES

SURVEILLANCE REQUIREMENTS (continued)

response times. Experience has shown that these components usually pass this surveillance when performed on a refueling frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

The SR 3.3.1.12 is modified by a note exempting neutron detectors from response time testing. A Note to the Surveillance indicates that neutron detectors may be excluded from RTS 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.

REFERENCES

1. Chapter 6.0, “Engineered Safety Features.”
2. Chapter 7.0, “Instrumentation and Controls.”
3. Chapter 15.0, “Accident Analysis.”
4. WCAP-16361-P, “Westinghouse Setpoint Methodology for Protection Systems – AP1000,” May 2006 (proprietary).
5. Institute of Electrical and Electronic Engineers, IEEE-603-1991, “IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations,” June 27, 1991.
6. 10 CFR 50.49, “Environmental Qualifications of Electric Equipment Important to Safety for Nuclear Power Plants.”
7. APP-GW-GSC-020, “Technical Specification Completion Time and Surveillance Frequency Justification.”
8. NRC Generic Letter No. 83-27, Surveillance Intervals in Standard Technical Specifications.
9. ESBU-TB-97-01, Westinghouse Technical Bulletin, “Digital Process Rack Operability Determination Criteria,” May 1, 1997.
10. WCAP-13632-P-A (Proprietary) and WCAP-13787-A (Non-Proprietary), Revision 2, “Elimination of Pressure Sensor Response Time Testing Requirements,” January 1996.

B 3.3 INSTRUMENTATION

B 3.3.2 Engineered Safety Feature Actuation System (ESFAS) Instrumentation

BASES

BACKGROUND

The ESFAS initiates necessary safety systems, based upon the values of selected unit parameters, to protect against violating core design limits and the Reactor Coolant System (RCS) pressure boundary, and to mitigate accidents.

The ESFAS instrumentation is segmented into distinct but interconnected modules.

Field Transmitters and Sensors

Normally, four redundant measurements using four separate sensors, are made for each variable used for actuation of ESF. The use of four channels for protection Functions is based on a minimum of two channels being required for a trip or actuation, one channel in test or bypass, and a single failure on the remaining channel. The signal selector in the Plant Control System will function correctly with only three channels. This includes two channels properly functioning and one channel having a single failure. Minimum requirements for protection and control is achieved with three channels OPERABLE. The fourth channel is provided to increase plant availability, and permits the plant to run for an indefinite time with a single channel out of service. The circuit design is able to withstand both an input failure to the control system, which may then require the protection Function actuation, and a single failure in the other channels providing the protection Function actuation. Again, a single failure will neither cause nor prevent the protection Function actuation. These requirements are described in IEEE-603 (Ref. 4). The actual number of channels provided for each plant parameter is specified in Reference 2.

Engineered Safety Features (ESF) Channel

An ESF channel extends from the sensor to the output of the associated ESF subsystem and shall include the sensor (or sensors), the signal conditioning, any associated data links, and the associated ESF subsystem. For ESF channels containing nuclear instrumentation, the ESF channel shall also include the nuclear instrument signal conditioning and the associated Nuclear Instrumentation Signal Processing and Control (NISPAC) subsystem. Any manual ESF controls that are associated with a particular ESF channel are also included in that ESF channel.

BASES

BACKGROUND (continued)

Plant Protection Subsystem

The Plant Protection contains the necessary equipment to:

- Permit acquisition and analysis of the sensor inputs, including plant process sensors and nuclear instrumentation, required for reactor trip and ESF calculations;
- Perform computation or logic operations on variables based on these inputs;
- Provide trip signals to the reactor trip switchgear and ESF actuation data to the ESF coincidence logic as required;
- Permit manual trip or bypass of each individual reactor trip Function and permit manual actuation or bypass of each individual voted ESF Function;
- Provide data to other systems in the Instrumentation and Control (I&C) architecture; and
- Provide separate input circuitry for control Functions that require input from sensors that are also required for protection Functions.

Each of the four divisions of plant protection provides signal conditioning, comparable output signals for indications in the main control room, and comparison of measured input signals with established setpoints. The basis of the setpoints are described in References 1, 2, and 3. If the measured value of a unit parameter exceeds the predetermined setpoint, an output is generated which is transmitted to the ESF coincidence logic for logic evaluation.

Within the protection and safety monitoring system, redundancy is generally provided for active equipment such as processors and communication hardware. This redundancy is provided to increase plant availability and facilitate surveillance testing. A division or channel is OPERABLE if it is capable of performing its specified safety function(s) and all the required supporting functions or systems are also capable of performing their related support functions. Thus, a division or channel is OPERABLE as long as one set of redundant components within the division or channel are capable of performing its specified safety function(s).

BASES

BACKGROUND (continued)

ESF Coincidence Logic

The ESF coincidence logic contains the necessary equipment to:

- Permit reception of the data supplied by the four divisions of plant protection and perform voting on the trip outputs;
- Perform system level logic using the input data from the plant protection subsystems and transmit the output to the ESF actuation subsystems; and
- Provide redundant hardware capable of providing system level commands to the ESF actuation subsystems.

ESF Actuation Subsystems

The ESF actuation subsystems contain the necessary equipment to:

- Receive automatic system level signals supplied by the ESF coincidence logic;
- Receive and transmit data to/from main control room multiplexers;
- Receive and transmit data to/from other PLCs on the same logic bus;
- Receive status data from component position switches (such as limit switches and torque switches); and
- Perform logic computations on received data, generate logic commands for final actuators (such as START, STOP, OPEN, and CLOSE).

ESF Coincidence Logic and ESF Actuation Subsystem OPERABILITY Background

Each ESF coincidence logic and ESF actuation subsystem has two subsystems that communicate by means of redundant halves of the logic bus. This arrangement is provided to facilitate testing. If one subsystem is removed from service, the remaining subsystem continues to function and the ESF division continues to provide full protection. At least one of these redundant halves is connected to the battery backed portion of the power system. This provides full functionality of the ESF division even when all ac power sources are lost. As long as one battery subsystem within an ESF coincidence logic or ESF actuation subsystem continues to

BASES

BACKGROUND (continued)

operate, the ESF division is unaffected. An ESF division is only affected when all battery backed subsystems within that division's ESF coincidence logic or ESF actuation subsystem are not OPERABLE.

Trip Setpoints and Allowable Values

The Trip Setpoints are the nominal values at which the trip output is set. Any trip output is considered to be properly adjusted when the "as left" value is within the band for CHANNEL CALIBRATION accuracy.

The Trip Setpoints used in the trip output are based on the analytical limits stated in Reference 2. The selection of these Trip Setpoints is such that adequate protection is provided when all sensor and processing time delays are taken into account. To allow for calibration tolerances, instrument drift, and severe environment errors for those ESFAS channels that must function in harsh environments as defined by 10 CFR 50.49 (Ref. 5), the Trip Setpoints and Allowable Values specified in Table 3.3.2-1 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 "Westinghouse Setpoint Methodology for Protection Systems" (Refs. 9 and 10). (Reference 9 is an AP600 document that describes a methodology that is applicable to AP1000. AP1000 has some slight differences in instrument spans as a result of the higher power level.) 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 COT. One example of such a change in measurement error is drift during the surveillance interval. If the measured setpoint does not exceed the Allowable Value, the trip output is considered OPERABLE.

Setpoints in accordance with the Allowable Value ensure that the consequences of Design Basis Accidents (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.

The Trip Setpoints and Allowable Values listed in Table 3.3.2-1 are based on the methodology described in Reference 9, 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.

BASES

BACKGROUND (continued)

Calibration tolerances and drift allowances must be specified in plant calibration procedures, and must be consistent with the values used in the setpoint methodology.

The OPERABILITY of each transmitter or sensor can be evaluated when its “as found” calibration data are compared against the “as left” data and are shown to be within the setpoint methodology assumptions. The basis of the setpoints is described in References 1, 2, 3, and 9. Trending of transmitter calibration is required by Generic Letter 91-04, “Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle.”

The protection and safety monitoring system testing features are designed to allow for complete functional testing by using a combination of system self-checking features, functional testing features, and other testing features. Successful functional testing consists of verifying that the capability of the system to perform the safety function has not failed or degraded. For hardware functions this would involve verifying that the hardware components and connections have not failed or degraded. Since software does not degrade, software functional testing involves verifying that the software code has not changed and that the software code is executing. To the extent possible, protection and safety monitoring system functional testing will be accomplished with continuous system self-checking features and the continuous functional testing features.

The protection and safety monitoring system incorporates continuous system self-checking features wherever practical. Self-checking features include on-line diagnostics for the computer system and the hardware and communications tests. These self-checking tests do not interfere with normal system operation.

In addition to the self-checking features, the system includes functional testing features. Functional testing features include continuous functional testing features and manually initiated functional testing features. To the extent practical functional testing features are designed not to interfere with normal system operation.

In addition to the system self-checking features and functional testing features, other test features are included for those parts of the system which are not tested with self-checking features or functional testing features. These test features allow for instruments/sensor checks,

BASES

BACKGROUND (continued)

calibration verification, response time testing, setpoint verification and component testing. The test features again include a combination of continuous testing features and manual testing features.

All of the testing features are designed so that the duration of the testing is as short as possible. Testing features are designed so that the actual logic is not modified. To prevent unwanted actuation, the testing features are designed with either the capability to bypass a Function during testing and/or limit the number of signals allowed to be placed in test at one time.

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY

Each of the analyzed accidents can be detected by one or more ESFAS Functions. One of the ESFAS Functions is the primary actuation signal for that accident. An ESFAS Function may be the primary actuation signal for more than one type of accident. An ESFAS Function may also be a secondary, or backup, actuation signal for one or more other accidents. For example, Pressurizer Pressure – Low is a primary actuation signal for small loss of coolant accidents (LOCAs) and a backup actuation signal for steam line breaks (SLBs) outside containment. Functions such as manual initiation not specifically credited in the accident safety analysis are qualitatively credited in the safety analysis and the NRC staff approved licensing basis for the plant. These Functions may provide protection for conditions which do not require dynamic transient analysis to demonstrate Function performance. These Functions may also serve as backups to Functions that were credited in the accident analysis (Ref. 3).

The LCO generally requires OPERABILITY of four channels in each instrumentation/logic Function and two devices for each manual initiation Function. The two-out-of-four configurations allow one channel to be bypassed during maintenance or testing without causing an ESFAS initiation. Two manual initiation channels are required to ensure no single random failure disables the ESFAS.

The required channels of ESFAS instrumentation provide plant protection in the event of any of the analyzed accidents. ESFAS protective functions are as follows:

1. Safeguards Actuation

   The Safeguards Actuation signal actuates the alignment of the Core Makeup Tank (CMT) valves for passive injection to the RCS. The Safeguards Actuation signal provides two primary Functions:

   - Primary side water addition to ensure maintenance or recovery of reactor vessel water level (coverage of the active fuel for

BASES

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

heat removal and clad integrity, peak clad temperature < 2200°F); and

- Boration to ensure recovery and maintenance of SHUTDOWN MARGIN (k_{eff} < 1.0).

These Functions are necessary to mitigate the effects of high energy line breaks (HELBs) both inside and outside of containment. The Safeguards Actuation signal is also used to initiate other Functions such as:

- Containment Isolation;
- Reactor Trip;
- Turbine Trip;
- Close Main Feedwater Control Valves;
- Trip Main Feedwater Pumps and Closure of Isolation and Crossover Valves; and
- Reactor Coolant Pump Trip.

These other Functions ensure:

- Isolation of nonessential systems through containment penetrations;
- Trip of the turbine and reactor to limit power generation;
- Isolation of main feedwater to limit secondary side mass losses;
- Trip of the reactor coolant pumps to ensure proper CMT actuation;
- Enabling automatic depressurization of the RCS on CMT Level – Low 1 to ensure continued safeguards actuated injection.

BASES

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

Manual and automatic initiation of Safeguards Actuation must be OPERABLE in MODES 1, 2, 3, and 4. In these MODES there is sufficient energy in the primary and secondary systems to warrant automatic initiation of ESF systems. Automatic actuation in MODE 4 is provided by the high containment pressure signal.

Manual initiation is required in MODE 5 to support system level initiation. Automatic initiation is not required to be OPERABLE in MODE 5 because parameters are not available to provide automatic actuation, and manual initiation is sufficient to mitigate the consequences of an accident.

These Safeguards Actuation Functions are not required to be OPERABLE in MODE 6 because there is adequate time for the operator to evaluate plant conditions and respond by manually starting individual systems, pumps, and other equipment to mitigate the consequences of an abnormal condition or accident. Plant pressure and temperature are very low and many ESF components are administratively locked out or otherwise prevented from actuating to prevent inadvertent overpressurization of plant systems.

1.a. Manual Initiation

The LCO requires that two manual initiation devices are OPERABLE. The operator can initiate the Safeguards Actuation signal at any time by using either of two switches in the main control room. This action will cause actuation of all components in the same manner as any of the automatic actuation signals.

The LCO on Manual Initiation ensures the proper amount of redundancy is maintained in the manual ESFAS actuation circuitry to ensure the operator has manual ESFAS initiation capability.

Each device consists of one switch and the interconnecting wiring to all four divisions. Each manual initiation device actuates all four divisions. This configuration does not allow testing at power.

BASES

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

1.b. Containment Pressure – High 2

This signal provides protection against the following accidents:

- SLB inside containment;
- LOCA; and
- Feed line break inside containment.

The transmitters (d/p cells) and electronics are located inside of containment. Since the transmitters and electronics are located inside of containment, they will experience adverse environmental conditions and the trip setpoint reflects environmental instrument uncertainties. The Containment Pressure – High 2 setpoint has been specified as low as reasonable, without creating potential for spurious trips during normal operations, consistent with the TMI action item (NUREG-0933, Item II.E.4.2) guidance.

The LCO requires four channels of Containment Pressure – High 2 to be OPERABLE in MODES 1, 2, 3, and 4. Four channels are provided to permit one channel to be in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function.

1.c. Pressurizer Pressure – Low

This signal provides protection against the following accidents:

- Inadvertent opening of a steam generator (SG) safety valve;
- SLB;
- A spectrum of rod cluster control assembly ejection accidents (rod ejection);
- Inadvertent opening of a pressurizer safety valve;
- LOCAs; and
- Steam Generator Tube Rupture (SGTR).

BASES

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

The transmitters are located inside containment, with the taps in the vapor space region of the pressurizer, and thus possibly experiencing adverse environmental conditions (LOCA, SLB inside containment). Therefore, the Trip Setpoint reflects the inclusion of both steady state and adverse environmental instrument uncertainties.

The LCO requires four channels of Pressurizer Pressure – Low to be OPERABLE in MODES 1, 2, and 3 (above P-11, when the RCS boron concentration is below that necessary to meet the SDM requirements at an RCS temperature of 200°F), to mitigate the consequences of a high energy line rupture inside containment. Four channels are provided to permit one channel to be in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function. This signal may be manually blocked by the operator below the P-11 setpoint. Automatic actuation below this pressure is then performed by the Containment Pressure – High 2 signal.

This Function is not required to be OPERABLE in MODE 3 below the P-11 setpoint. Other ESF Functions are used to detect accident conditions and actuate the ESF systems in this MODE. In MODES 4, 5, and 6, this Function is not needed for accident detection and mitigation.

1.d. Steam Line Pressure – Low

Steam Line Pressure – Low provides protection against the following accidents:

- SLB;
- Feed line break; and
- Inadvertent opening of an SG relief or an SG safety valve.

It is possible for the transmitters to experience adverse environmental conditions during a secondary side break. Therefore, the Trip Setpoint reflects both steady state and adverse environmental instrument uncertainties.

This Function is anticipatory in nature and has a typical lead/lag ratio of 50/5.

BASES

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

The LCO requires four channels of Steam Line Pressure – Low to be OPERABLE in MODES 1, 2, and 3 (above P-11, when the RCS boron concentration is below that necessary to meet the SDM requirements at an RCS temperature of 200°F). At these conditions, a secondary side break or stuck open valve could result in the rapid depressurization of the steam lines. Four channels are provided in each steam line to permit one channel to be in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function. This signal may be manually blocked by the operator below the P-11 setpoint. Below P-11, feed line break is not a concern, inside containment SLB will be terminated by automatic actuation via Containment Pressure – High 2, and outside containment SLB will be terminated by the Steam Line Pressure-Negative Rate – High signal for steam line isolation. In MODE 4, 5, or 6, this Function is not needed for accident detection and mitigation because the steam line pressure is below the actuation setpoint. Low steam line pressure in these MODES is not an adequate indication of a feed line or steam line break.

1.e. RCS Cold Leg Temperature (T_{cold}) – Low

This signal provides protection against the following accidents:

- SLB;
- Feed line break; and
- Inadvertent opening of an SG relief or an SG safety valve.

The LCO requires four channels of T_{cold} – Low to be OPERABLE in MODES 1 and 2, and in MODE 3 with any main steam isolation valve open and above P-11 when the RCS boron concentration is below that necessary to meet the SDM requirements at an RCS temperature of 200°F. At these conditions, a secondary side break or stuck open valve could result in the rapid cooldown of the primary side. Four channels are provided in each loop to permit one channel to be in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function. In MODES 4, 5, and 6, this Function is not needed for accident detection and mitigation because the cold leg temperature is reduced below the actuation setpoint.

BASES

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

2. Core Makeup Tank (CMT) Actuation

CMT Actuation provides the passive injection of borated water into the RCS. Injection provides RCS makeup water and boration during transients or accidents when the normal makeup supply from the Chemical and Volume Control System (CVS) is lost or insufficient. Two tanks are available to provide passive injection of borated water. CMT injection mitigates the effects of high energy line breaks by adding primary side water to ensure maintenance or recovery of reactor vessel water level following a LOCA, and by borating to ensure recovery or maintenance of SHUTDOWN MARGIN following a steam line break. CMT Valve Actuation is initiated by the Safeguards Actuation signal, Pressurizer Level – Low 2, ADS Stages 1, 2 and 3 Actuation, or manually.

The LCO requires that manual and automatic CMT Valve Actuation be OPERABLE in MODES 1 through 4. Manual and Automatic actuation of the CMT valves is additionally required in MODE 5 with the RCS pressure boundary intact. Actuation of this Function is not required in MODE 5 with the RCS pressure boundary open, or MODE 6 because the CMTs are not required to be OPERABLE in these MODES.

2.a. Manual Initiation

Manual CMT Valve Actuation is accomplished by either of two switches in the main control room. Either switch activates all four divisions.

2.b. Pressurizer Water Level – Low 2

This Function also initiates CMT Valve Actuation from the coincidence of pressurizer level below the Low 2 Setpoint in any two of the four divisions. This function can be manually blocked when the pressurizer water level is below the P-12 Setpoint. This Function is automatically unblocked when the pressurizer water level is above the P-12 Setpoint. The Setpoint reflects both steady state and adverse environmental instrument uncertainties as the detectors provide protection for an event that results in a harsh environment.

BASES

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

2.c. Safeguards Actuation (Function 1)

CMT Valve Actuation is also initiated by all Functions that initiate the Safeguards Actuation signal. The CMT Valve Actuation Function requirements are the same as the requirements for the Safeguards Actuation Functions, but only apply in MODES 1 through 4, and in MODE 5 with the RCS pressure boundary intact. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead, Function 1 is referenced for all initiating Functions and requirements.

2.d. ADS Stages 1, 2, and 3 Actuation (Function 9)

The CMTs are actuated on an ADS Stages 1, 2, and 3 actuation. The CMT Actuation Function requirements are the same as the requirements for the ADS Stages 1, 2, and 3 Actuation Function, but only apply in MODES 1 through 4, and in MODE 5 with the RCS pressure boundary intact. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead, Function 9 is referenced for all initiating functions and requirements.

3. Containment Isolation

Containment Isolation provides isolation of the containment atmosphere and selected process systems which penetrate containment from the environment. This Function is necessary to prevent or limit the release of radioactivity to the environment in the event of a large break LOCA.

Containment Isolation is actuated by the Safeguards Actuation signal, manual actuation of containment cooling, or manually.

Manual and automatic initiation of Containment Isolation must be OPERABLE in MODES 1, 2, 3, and 4, when containment integrity is required. Manual initiation is required in MODE 5 and MODE 6 for closure of open penetrations providing direct access from the containment atmosphere to the outside atmosphere. Manual initiation of this Function in MODES 5 and 6 is not applicable if the direct access lines penetrating containment are isolated. Initiation of containment isolation by manual initiation of passive containment cooling in MODE 5 or 6 with decay heat ≤ 9.0 MWt is not required because OPERABILITY of the passive containment cooling system is not required when air cooling is sufficient. This provides the capability to

BASES

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

manually initiate containment isolation during all MODES. Automatic Safeguards Actuation is required in MODE 5 for closure of open penetrations providing direct access from the containment atmosphere to the outside atmosphere. Automatic Safeguards Actuation is not required in MODE 6 because manual initiation is sufficient to mitigate the consequences of an accident in this MODE.

3.a. Manual Initiation

Manual Containment Isolation is accomplished by either of two switches in the main control room. Either switch actuates all four ESFAC divisions.

3.b. Manual Initiation of Passive Containment Cooling (Function 12.a)

Containment Isolation is also initiated by Manual Initiation of Passive Containment Cooling. This is accomplished as described for ESFAS Function 12.a, but are not applicable if the direct access flow paths are isolated.

3.c. Safeguards Actuation (Function 1)

Containment Isolation is also initiated by all Functions that initiate the Safeguards Actuation signal. The Containment Isolation Function requirements are the same as the requirements for the Safeguards Actuation Function, but are not applicable if the direct access flow paths are isolated. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead, Function 1 is referenced for all initiating functions and requirements.

4. Steam Line Isolation

Isolation of the main steam lines provides protection in the event of an SLB inside or outside containment. Rapid isolation of the steam lines will limit the steam break accident to the blowdown from one SG at most. For an SLB upstream of the isolation valves, inside or outside of containment, closure of the isolation valves limits the accident to the blowdown from only the affected SG. For a SLB downstream of the isolation valves, closure of the isolation valves terminates the accident as soon as the steam lines depressurize.

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APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

Closure of the turbine stop and control valves and the main steam branch isolation valves is initiated by this Function. Closure of these valves limits the accidental depressurization of the main steam system associated with an inadvertent opening of a single steam dump, relief, safety valve, or a rupture of a main steam line. Closure of these valves also supports a steam generator tube rupture event by isolating the faulted steam generator.

4.a. Manual Initiation

Manual initiation of Steam Line Isolation can be accomplished from the main control room. There are two switches in the main control room and either switch can initiate action to immediately close all main steam isolation valves (MSIVs). The LCO requires two OPERABLE channels in MODES 1, 2, 3, and 4 with any main steam valve open, when there is sufficient energy in the RCS and SGs to have an SLB or other accident resulting in the release of significant quantities of energy to cause a cooldown of the primary system. In MODES 5 and 6, this Function is not required to be OPERABLE because there is insufficient energy in the secondary side of the unit to cause an accident.

4.b. Containment Pressure – High 2

This Function actuates closure of the MSIVs in the event a SLB inside containment to limit the mass and energy release to containment and limit blowdown to a single SG.

The transmitters and electronics are located inside containment, thus, they will experience harsh environmental conditions and the Trip Setpoint reflects environmental instrument uncertainties.

The Containment Pressure – High 2 setpoint has been specified as low as reasonable, without creating potential for spurious trips during normal operations, consistent with the TMI action item (NUREG-0933, Item II.E.4.2) guidance. The LCO requires four channels of Containment Pressure – High 2 to be OPERABLE in MODES 1, 2, 3, and 4, with any main steam valve open, when there is sufficient energy in the primary and secondary side to pressurize the containment following a pipe break. Four channels are provided to permit one channel to be in trip or bypass indefinitely and still ensure

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

no single random failure will disable this trip Function. There would be a significant increase in the containment pressure, thus allowing detection and closure of the MSIVs. In MODES 5 and 6, there is not enough energy in the primary and secondary sides to pressurize the containment to the Containment Pressure – High 2 setpoint.

4.c. Steam Line Pressure

(1) Steam Line Pressure – Low

Steam Line Pressure – Low provides closure of the MSIVs in the event of an SLB to limit the mass and energy release to containment and limit blowdown to a single SG.

The LCO requires four channels of Steam Line Pressure – Low Function to be OPERABLE in MODES 1, 2, and 3 (above P-11, when the RCS boron concentration is below that necessary to meet the SDM requirements at an RCS temperature of 200°F), with any main steam isolation valve open, when a secondary side break or stuck open valve could result in the rapid depressurization of the steam lines. Four channels are provided in each steam line to permit one channel to be in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function. This signal may be manually blocked by the operator below the P-11 setpoint. Below P-11, an inside containment SLB will be terminated by automatic actuation via Containment Pressure – High 2, and stuck open valve transients and outside containment steam line breaks will be terminated by the Steam Line Pressure-Negative Rate – High signal for Steam Line Isolation. In MODES 4, 5, and 6, this Function is not needed for accident detection and mitigation.

(2) Steam Line Pressure-Negative Rate – High

Steam Line Pressure-Negative Rate – High provides closure of the MSIVs for an SLB, when less than the P-11 setpoint, to maintain at least one unfaulted SG as a heat sink for the reactor and to limit the mass and energy release to containment. When the operator manually

BASES

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

blocks the Steam Line Pressure – Low when less than the P-11 setpoint, the Steam Line Pressure-Negative Rate – High signal is automatically enabled.

The LCO requires four channels of Steam Line Pressure-Negative Rate – High to be OPERABLE in MODE 3, with any main steam valve open, when less than the P-11 setpoint, when a secondary side break or stuck open valve could result in the rapid depressurization of the steam line(s). Four channels are provided in each steam line to permit one channel to be in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function. In MODES 1 and 2, and in MODE 3 when above the P-11 setpoint with the RCS boron concentration below that necessary to meet the SDM requirements at an RCS temperature of 200°F, this signal is automatically disabled and the Steam Line Pressure – Low signal is automatically enabled.

In MODES 4, 5, and 6, this Function is not needed for accident detection and mitigation.

While the transmitters may experience elevated ambient temperatures due to a steam line break, the Trip Function is on rate of change, not the absolute accuracy of the indicated steam pressure. Therefore, the Trip Setpoint reflects only steady state instrument uncertainties.

4.d. T_{cold} – Low

This Function provides closure of the MSIVs during a SLB or inadvertent opening of a SG relief or a safety valve to maintain at least one unfaulted SG as a heat sink for the reactor and to limit the mass and energy release to containment.

This Function was discussed as Safeguards Actuation Function 1.e.

The LCO requires four channels of T_{cold} – Low to be OPERABLE in MODES 1 and 2, and in MODE 3 above P-11 when the RCS boron concentration is below that necessary to meet the SDM requirements at an RCS temperature of 200°F, with any main steam isolation valve open, when a secondary side break or stuck open valve could result in the rapid

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

cooldown of the primary side. Four channels are provided in each loop to permit one channel to be in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function. In MODE 3 below P-11 and in MODES 4, 5, and 6, this Function is not needed for accident detection and mitigation because the cold leg temperature is reduced below the actuation setpoint.

5. Turbine Trip

The primary Function of the Turbine Trip is to prevent damage to the turbine due to water in the steam lines. This Function is necessary in MODES 1 and 2, and 3 above P-11 to mitigate the effects of a large SLB or a large Feedline Break (FLB). Failure to trip the turbine following a SLB or FLB can lead to additional mass and energy being delivered to the steam generators, resulting in excessive cooldown and additional mass and energy release in containment. In MODES 3, 4, 5, and 6, the turbine is not in operation and this function is not required to be OPERABLE.

This Function is actuated by Steam Generator Water Level – High 2, by a Safeguards Actuation signal, or manually. The Reactor Trip Signal also initiates a turbine trip signal whenever a reactor trip (P-4) is generated.

5.a. Manual Main Feedwater Isolation

The Turbine Trip is also initiated by the Manual Main Feedwater Control Valve Isolation Function. The requirements for this Function are the same as the requirements for Manual Main Feedwater Control Valve Isolation (Function 6.a), but only apply in MODES 1 and 2. Therefore, the requirements are not repeated in Table 3.3.2-1, and Function 6.a is referenced for all requirements.

5.b. Steam Generator Narrow Range Water Level – High 2

This signal provides protection against excessive feedwater flow by closing the main feedwater control, isolation and crossover valves, tripping of the main feedwater pumps, and tripping the turbine. Four channels are provided to permit one channel to be in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function. The transmitters (d/p cells) are located inside containment.

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APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

However, the events which this Function protect against cannot cause severe environment in containment. Therefore, the Setpoint reflects only steady state instrument uncertainties.

5.c. Safeguards Actuation (Function 1)

Turbine Trip is also initiated by all Functions that initiate the Safeguards Actuation signal. The Turbine Trip Function requirements are the same as the requirements for the Safeguards Actuation Function, but only apply in MODES 1 and 2. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead Function 1 is referenced for all initiating Functions and requirements. The Safeguards Actuation signal closes all main feedwater control, isolation and crossover valves, trips all main feedwater pumps, and trips the turbine.

5.d. Reactor Trip (Function 18.a)

Turbine Trip is also initiated by all functions that initiate Reactor Trip. The turbine trip function requirements are the same as the requirements for the Reactor Trip Function, but only apply in MODES 1 and 2. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead Function 18.a, P-4 (Reactor Trip), is referenced for all initiating Functions and requirements.

6. Main Feedwater Control Valve Isolation

The primary Function of Main Feedwater Control Valve Isolation is to prevent damage to the turbine due to water in the steam lines and to stop the excessive flow of feedwater into the SGs. This Function is actuated by Steam Generator Narrow Range Water Level – High 2, by a Safeguards Actuation signal, or manually. The Reactor Trip Signal also initiates closure of the main feedwater control valves coincident with a low RCS average temperature (T_{avg}) signal whenever a reactor trip (P-4) is generated.

Closing the Main Feedwater Control Valves on Manual Main Feedwater Isolation, SG Narrow Range Water Level-High 2, or Safeguards Actuation is necessary in MODES 1, 2, and 3 to mitigate the effects of a large SLB or a large FLB. This Function is also required to be OPERABLE in MODES 1 and 2 on T_{avg} Low-1 coincident with Reactor Trip (P-4). Failure to close the main feedwater control valves following a SLB or FLB can lead to

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APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

additional mass and energy being delivered to the steam generators, resulting in excessive cooldown and additional mass and energy release in containment. Manual main feedwater isolation is required to be OPERABLE in MODE 4 when the main feedwater control valves are open. This Function is not applicable in MODE 4 for valve isolation if the main feedwater line is isolated. Automatic actuation on a Steam Generator Narrow Range Water Level – High 2 is required to be OPERABLE in MODE 4 when the RCS is not being cooled by the RNS. In MODES 5 and 6, the energy in the RCS and the steam generators is low and this function is not required to be OPERABLE.

6.a. Manual Main Feedwater Isolation

Manual Main Feedwater Isolation can be accomplished from the main control room. There are two switches in the main control room and either switch can initiate action in both divisions to close all main and startup feedwater control, isolation and crossover valves, trip all main and startup feedwater pumps, and trip the turbine.

6.b. Steam Generator Narrow Range Water Level – High 2

This signal provides protection against excessive feedwater flow by closing the main feedwater control, isolation and crossover valves, tripping of the Main Feedwater Pumps, and tripping the turbine.

Four channels are provided to permit one channel to be in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function. The transmitters (d/p cells) are located inside containment. However, the events which this Function protect against cannot cause severe environment in containment. Therefore, the Setpoint reflects only steady state instrument uncertainties.

6.c. Safeguards Actuation (Function 1)

This Function is also initiated by all Functions that initiate the Safeguards Actuation signal. The Main Feedwater Control Valve Isolation Function requirements are the same as the requirements for the Safeguards Actuation Function, but do not apply in MODE 4 with the flow paths isolated. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead

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APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

Function 1 is referenced for all initiating Functions and requirements. The Safeguards Actuation signal closes all main feedwater control, isolation and crossover valves, trips all main feedwater pumps, and trips the turbine.

6.d. T_{avg} Low-1 Coincident with Reactor Trip (P-4)

This signal provides protection against excessive feedwater flow by closing the main feedwater control valves. This signal results from a coincidence of two of the four divisions of reactor loop average temperature below the Low 1 setpoint coincident with the P-4 permissive. Four channels are provided to permit one channel to be in trip or bypass indefinitely and still ensure that no single random failure will disable this trip Function.

7. Main Feedwater Pump Trip and Valve Isolation

The primary function of the Main Feedwater Pump Trip and Isolation is to prevent damage to the turbine due to water in the steam lines and to stop the excessive flow of feedwater into the SGs. Valve isolation includes closing the main feedwater isolation and crossover valves. Isolation of main feedwater is necessary to prevent an increase in heat removal from the reactor coolant system in the event of a feedwater system malfunction. Addition of excessive feedwater causes an increase in core power by decreasing reactor coolant temperature. This Function is actuated by Steam Generator Water Level – High 2, by a Safeguards Actuation signal, or manually. The Reactor Trip Signal also initiates a turbine trip signal whenever a reactor trip (P-4) is generated.

This Function is necessary in MODES 1, 2, 3, and 4 to mitigate the effects of a large SLB or a large FLB except T_{avg} Low 2 coincident with Reactor Trip (P-4) which is required to be OPERABLE in MODES 1 and 2. Failure to trip the turbine or isolate the main feedwater system following a SLB or FLB can lead to additional mass and energy being delivered to the steam generators, resulting in excessive cooldown and additional mass and energy release in containment. Manual main feedwater isolation is required to be OPERABLE in MODE 4 when the main feedwater isolation valves are open. This Function is not applicable in MODE 4 for valve isolation if the main feedwater line is isolated. Automatic actuation on a Steam Generator Narrow Range Water Level – High 2 is required to be OPERABLE in MODE 4 when the RCS is not being

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APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

cooled by the RNS. In MODES 5 and 6, the energy in the RCS and the steam generators is low and this Function is not required to be OPERABLE.

7.a. Manual Main Feedwater Isolation

The Main Feedwater Pump Trip and Valve Isolation is also initiated by the Manual Main Feedwater Control Valve Isolation Function. The requirements for this Function are the same as the requirements for Manual Main Feedwater Control Valve Isolation (Function 6.a). Therefore, the requirements are not repeated in Table 3.3.2-1, and Function 6.a is referenced for all requirements.

7.b. Steam Generator Narrow Range Water Level – High 2

This signal provides protection against excessive feedwater flow by closing the main feedwater control, isolation and crossover valves, tripping of the main feedwater pumps, and tripping the turbine. Four channels are provided to permit one channel to be in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function. The transmitters (d/p cells) are located inside containment. However, the events which this Function protect against cannot cause severe environment in containment. Therefore, the Setpoint reflects only steady state instrument uncertainties.

7.c. Safeguards Actuation (Function 1)

This Function is also initiated by all Functions that initiate the Safeguards Actuation signal. The Main Feedwater Pump Trip and Valve Isolation Function requirements are the same as the requirements for their Safeguards Actuation Function, but do not apply in MODE 4 with the flow paths isolated. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead Function 1 is referenced for all initiating Functions and requirements. The Safeguards Actuation signal closes all main feedwater control, isolation and crossover valves, trips all main feedwater pumps, and trips the turbine.

7.d. T_{avg} Low-2 Coincident with Reactor Trip (P-4)

This signal provides protection against excessive feedwater flow by closing the main feedwater isolation and crossover leg valves, and tripping of the main feedwater pumps. This signal

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APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

results from a coincidence of two out of four divisions of reactor loop average temperature below the Low 2 setpoint coincident with the P-4 permissive. Four channels are provided to permit one channel to be in trip or bypass indefinitely and still ensure that no single random failure will disable this trip Function. This Function may be manually blocked when the pressurizer pressure is below the P-11 setpoint. The block is automatically removed when the pressurizer pressure is above the P-11 setpoint.

8. Startup Feedwater Isolation

The primary Function of the Startup Feedwater Isolation is to stop the excessive flow of feedwater into the SGs. This Function is necessary in MODES 1, 2, 3, and 4 to mitigate the effects of a large SLB or a large FLB. Failure to isolate the startup feedwater system following a SLB or FLB can lead to additional mass and energy being delivered to the steam generators, resulting in excessive cooldown and additional mass and energy release in containment.

Startup feedwater isolation must be OPERABLE in MODES 1, 2, 3, and 4 when there is significant mass and energy in the RCS and the steam generators. This Function is not applicable in MODE 4 when the startup feedwater flow paths are isolated. In MODES 5 and 6, the energy in the RCS and the steam generators is low and this Function is not required to be OPERABLE.

8.a. Steam Generator (SG) Narrow Range Water Level – High 2

If steam generator narrow range level reaches the High 2 setpoint in either steam generator, then all startup feedwater control and isolation valves are closed and the startup feedwater pumps are tripped. Four channels are provided in each steam generator to permit one channel to be in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function.

8.b. T_{cold} – Low

This Function closes the startup feedwater control and isolation valves and trips the startup feedwater pumps if reactor coolant system cold leg temperature is below the T_{cold} setpoint in any loop. Startup feedwater isolation on this condition may be manually blocked when the pressurizer

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APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

pressure is below the P-11 setpoint. This function is automatically unblocked when the pressurizer pressure is above the P-11 setpoint with the RCS boron concentration below that necessary to meet the SDM requirements at an RCS temperature of 200°F. Four channels are provided in each loop to permit one channel to be in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function.

8.c. Manual Main Feedwater Control Valve Isolation (Function 6.a)

The Startup Feedwater Isolation is also initiated by the Manual Main Feedwater Control Valve Isolation Function. The requirements for this Function are the same as the requirements for the Manual Main Feedwater Control Valve Isolation (Function 6.a). Therefore, the requirements are not repeated in Table 3.3.2-1, and Function 6.a is referenced for all requirements.

9. ADS Stages 1, 2, & 3 Actuation

The Automatic Depressurization System (ADS) provides a sequenced depressurization of the reactor coolant system to allow passive injection from the CMTs, accumulators, and the in-containment refueling water storage tank (IRWST) to mitigate the effects of a LOCA. The depressurization is accomplished in four stages, with the first three stages discharging into the IRWST and the last stage discharging into containment. Each of the first three stages consists of two parallel paths with each path containing an isolation valve and a depressurization valve.

The first stage isolation valves open on any ADS Stages 1, 2, and 3 actuation. The first stage depressurization valves are opened following a preset time delay after the actuation of the isolation valves. The second stage isolation valves are opened following a preset time delay after actuation of the first stage depressurization valves open. The second stage depressurization valves are opened following a preset time delay after the second stage isolation valves are actuated, similar to stage one. Similar to the second stage, the third stage isolation valves are opened following a preset time delay after the actuation of the second stage depressurization valves. The third stage depressurization valves are opened following a preset time delay after the third stage isolation valves are actuated.

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APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

9.a. Manual Initiation

The first stage depressurization valves open on manual actuation. Any ADS Stages 1, 2, and 3 actuation also actuates PRHR and trips all reactor coolant pumps. The operator can initiate an ADS Stages 1, 2, and 3 actuation from the main control room by simultaneously actuating two ADS actuation devices in the same set. There are two sets of two switches each in the main control room. Simultaneously actuating the two devices in either set will actuate ADS Stages 1, 2, and 3. This Function must be OPERABLE in MODES 1, 2, 3, and 4. This Function must also be OPERABLE in MODES 5 and 6 when the required ADS valves are not open, and in MODE 6 with the upper internals in place. The required ADS valves or equivalent relief area are specified in LCO 3.4.12, ADS - Shutdown, RCS Intact and LCO 3.4.13, ADS - Shutdown, RCS Open.

9.b. CMT Level – Low 1 Coincident with CMT Actuation

This Function ensures continued passive injection or borated water to the RCS following a small break LOCA. ADS Stages 1, 2 and 3 actuation is initiated when the CMT Level reaches its Low 1 Setpoint coincident with any CMT Actuation signal (Function 2). Four channels are provided in each CMT to permit one channel to be in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function.

The ADS Stages 1, 2, and 3 Actuation Function requirements are the same as the requirements discussed in Function 2 (CMT Actuation). Therefore, the requirements are not repeated in Table 3.3.2-1. Instead Function 2 is referenced for all initiating functions and requirements. This Function must be OPERABLE in MODES 1, 2, 3, and 4.

This Function must also be OPERABLE in MODE 5 with pressurizer level ≥ 20% and the required ADS valves not open. The required ADS valves or equivalent relief area are specified in LCO 3.4.12, ADS - Shutdown, RCS Intact and LCO 3.4.13, ADS - Shutdown, RCS Open. In MODE 5, only one CMT is required to be OPERABLE in accordance with LCO 3.5.3, CMTs - Shutdown, RCS Intact; therefore, CMT level channels are only required on an OPERABLE CMT.

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APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

10. ADS Stage 4 Actuation

The ADS provides a sequenced depressurization of the reactor coolant system to allow passive injection from the CMTs, accumulators, and the IRWST to mitigate the effects of a LOCA. The depressurization is accomplished in four stages, with the first three stages discharging into the IRWST and the fourth stage discharging into containment.

The fourth stage of the ADS consists of four parallel paths. Each of these paths consists of a normally open isolation valve and a depressurization valve. The four paths are divided into two groups with two paths in each group. Within each group, one path is designated to be substage A and the second path is designated to be substage B.

The substage A depressurization valves are opened following a preset time delay after the substage A isolation valve confirmatory open signal. The sequence is continued with substage B. A confirmatory open signal is provided to the substage B isolation valves following a preset time delay after the substage A depressurization valve has been opened. The signal to open the substage B depressurization valve is provided following a preset time delay after the substage B isolation valves confirmatory open signal.

10.a. Manual Initiation Coincident with RCS Wide Range Pressure – Low or ADS Stages 1, 2, and 3 Actuation (Function 9)

The fourth stage depressurization valves open on manual actuation. The operator can initiate Stage 4 of ADS from the main control room. There are two sets of two switches each in the main control room. Actuating the two switches in either set will actuate all 4th stage ADS valves. This manual actuation is interlocked to actuate with either the low RCS pressure signal or with the ADS Stages 1, 2, & 3 actuation (Function 9). These interlocks minimize the potential for inadvertent actuation of this Function. This interlock with Function 9 allows manual actuation of this Function if automatic or manual actuation of the ADS Stages 1, 2, & 3 valves fails to depressurize the RCS due to common-mode failure. This consideration is important in PRA modeling to improve the reliability of reducing the RCS pressure following a small LOCA or transient event. This Function must be OPERABLE

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APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

in MODES 1, 2, 3, and 4. This Function must also be OPERABLE in MODES 5 and 6 when the required ADS valves are not open, and in MODE 6 with the upper internals in place. The required ADS valves or equivalent relief area are specified in LCO 3.4.12, ADS - Shutdown, RCS Intact and LCO 3.4.13, ADS - Shutdown, RCS Open.

10.b. CMT Level – Low 2 Coincident with RCS Wide Range Pressure – Low

The fourth stage depressurization valves open on CMT Level – Low 2 in two-out-of-four channels in either CMT. Actuation of the fourth stage depressurization valves is interlocked with the third stage depressurization signal such that the fourth stage is not actuated unless the third stage has been previously actuated following a preset time delay. Actuation of the fourth stage ADS valves are further interlocked with a low RCS pressure signal such that the ADS Stage 4 actuation is not actuated unless the RCS pressure is below a predetermined setpoint. Four channels of CMT level are provided to permit one channel to be in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function. This Function must be OPERABLE in MODES 1, 2, 3, and 4. This Function must also be OPERABLE in MODE 5 when the required ADS valves are not open and with the pressurizer level ≥ 20%. The required ADS valves or equivalent relief area are specified in LCO 3.4.12, ADS - Shutdown, RCS Intact and LCO 3.4.13, ADS - Shutdown, RCS Open. In MODE 5, only one CMT is required to be OPERABLE in accordance with LCO 3.5.3, CMTs - Shutdown, RCS Intact; therefore, CMT level channels are only required on an OPERABLE CMT.

10.c. Coincident RCS Loop 1 and 2 Hot Leg Level – Low

A signal to automatically open the ADS Stage 4 is also generated when coincident loop 1 and 2 reactor coolant system hot leg level indication decreases below an established setpoint for a duration exceeding an adjustable time delay. This Function is required to be OPERABLE in MODE 4 with the RCS being cooled by the RNS. This Function is also required to be OPERABLE in MODE 5 and in MODE 6 when the required ADS valves are not open. The

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APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

required ADS valves or equivalent relief area are specified in LCO 3.4.12, ADS - Shutdown, RCS Intact and LCO 3.4.13, ADS - Shutdown, RCS Open.

11. Reactor Coolant Pump Trip

Reactor Coolant Pump (RCP) Trip allows the passive injection of borated water into the RCS. Injection provides RCS makeup water and boration during transients or accidents when the normal makeup supply from the CVS is lost or insufficient. Two tanks provide passive injection of borated water by gravity when the reactor coolant pumps are tripped. CMT injection mitigates the effects of high energy line breaks by adding primary side water to ensure maintenance or recovery of reactor vessel water level following a LOCA, and by borating to ensure recovery or maintenance of SHUTDOWN MARGIN following a steam line break. RCP trip on high bearing water temperature protects the RCP coast down. A high bearing water temperature trip signal will result in the tripping of all the RCPs. RCP trip is actuated by High RCP bearing water temperature, ADS Stages 1, 2, and 3 Actuation (Function 9), Manual CMT Actuation (Function 2.a), Pressurizer Water Level – Low 2, and Safeguards Actuation (Function 1).

11.a. ADS Stage 1, 2, and 3 Actuation (Function 9)

The RCPs are tripped any time ADS Stage 1, 2, and 3 actuation is initiated. The RCP trip Function requirements for the ADS Stage 1, 2, and 3 actuation are the same as the requirements for the ADS Function. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead Function 9 is referenced for all initiating functions and requirements.

11.b. Reactor Coolant Pump Bearing Water Temperature – High

The RCPs are tripped if two-out-of-four sensors on any RCP indicate high bearing water temperature. This Function is required to be OPERABLE in MODES 1 and 2. Four channels are provided to permit one channel to be in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function.

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APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

11.c. Manual CMT Actuation (Function 2.a)

RCP trip is also initiated by the manual CMT actuation Function. The RCP trip Function requirements are the same as the requirements for the manual CMT actuation Function. Therefore, the requirements are not repeated in Table 3.3.2-1, and Function 2.a is referenced for all requirements.

11.d. Pressurizer Water Level – Low 2

The RCPs are tripped when the pressurizer water level reaches its Low 2 setpoint. This signal results from the coincidence of pressurizer water level below the Low 2 setpoint in any two-of-four divisions. This Function is required to be OPERABLE in MODES 1, 2, 3, and 4. This Function is also required to be OPERABLE in MODE 5 with pressurizer level ≥ 20%, when the RCS is not being cooled by the RNS. This Function can be manually blocked when the pressurizer water level is below the P-12 setpoint. This Function is automatically unblocked when the pressurizer water level is above the P-12 setpoint.

11.e. Safeguards Actuation (Function 1)

This Function is also initiated by all Functions that initiated the Safeguards Actuation signal. The requirements for the reactor trip Functions are the same as the requirements for the Safeguards Actuation Function. Therefore, the requirements are not repeated in Table 3.3.2.1. Instead, Function 1 is referenced for all initiating Functions and requirements.

12. Passive Containment Cooling Actuation

The Passive Containment Cooling System (PCS) transfers heat from the reactor containment to the environment. This Function is necessary to prevent the containment design pressure and temperature from being exceeded following any postulated DBA (such as LOCA or SLB). Heat removal is initiated automatically in response to a Containment Pressure – High 2 signal or manually.

A Passive Containment Cooling Actuation signal initiates water flow by gravity by opening the isolation valves. The water flows onto the containment dome, wetting the outer surface. The path for natural circulation of air along the outside walls of the containment structure is always open.

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APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

The LCO requires this Function to be OPERABLE in MODES 1, 2, 3, and 4 when the potential exists for a DBA that could require the

operation of the Passive Containment Cooling System. In MODES 5 and 6, with decay heat more than 9.0 MWt, manual initiation of the PCS provides containment heat removal. Section B 3.6.7, Applicability, provides the basis for the decay heat limit.

12.a. Manual Initiation

The operator can initiate Containment Cooling at any time from the main control room by actuating either of the two containment cooling actuation switches. There are two switches in the main control room, either of which will actuate containment cooling in all divisions. Manual Initiation of containment cooling also actuates containment isolation.

12.b. Containment Pressure – High 2

This signal provides protection against a LOCA or SLB inside containment. Four channels are provided to permit one channel to be in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function.

The transmitters and electronics are located inside containment, thus, they will experience harsh environmental conditions and the trip setpoint reflects only steady state instrument uncertainties associated with the containment environment. The Containment Pressure – High 2 setpoint has been specified as low as reasonable, without creating potential for spurious trips during normal operations, consistent with the TMI action item (NUREG-0933, Item II.E.4.2) guidance.

13. PRHR Heat Exchanger Actuation

The PRHR Heat Exchanger (HX) provides emergency core decay heat removal when the Startup Feedwater System is not available to provide a heat sink. PRHR is actuated when the discharge valves are opened in response to Steam Generator Narrow Range (NR) Level – Low coincident with Startup Feedwater Flow – Low, Steam Generator Wide Range (WR) Level – Low, ADS Stages 1, 2, and 3 Actuation, CMT Actuation, Pressurizer Water Level – High 3, or Manual Initiation.

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APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

13.a. Manual Initiation

Manual PRHR actuation is accomplished by either of two switches in the main control room. Either switch actuates all four ESFAC Divisions.

This Function is required to be OPERABLE in MODES 1, 2, 3, and 4, and MODE 5 with the RCS pressure boundary intact. This ensures that PRHR can be actuated in the event of a loss of the normal heat removal systems.

13.b. Steam Generator Narrow Range Level – Low Coincident with Startup Feedwater Flow – Low

PRHR is actuated when the Steam Generator Narrow Range Level reaches its low setpoint coincident with an indication of low Startup Feedwater Flow.

The LCO requires four channels per steam generator to be OPERABLE to satisfy the requirements with a two-out-of-four logic. Four channels are provided to permit one channel to be in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function. The Setpoint reflects both steady state and adverse environmental instrument uncertainties as the detectors provide protection for an event that results in a harsh environment.

Startup Feedwater Flow – Low uses a one-out-of-two logic on each of the two startup feedwater lines. This Function is required to be OPERABLE in MODES 1, 2, and 3 and in MODE 4 when the RCS is not being cooled by the Normal Residual Heat Removal System (RNS). This ensures that PRHR can be actuated in the event of a loss of the normal heat removal systems. In MODE 4 when the RCS is being cooled by the RNS, and in MODES 5 and 6, the SGs are not required to provide the normal RCS heat sink. Therefore, startup feedwater flow is not required, and PRHR actuation on low startup feedwater flow is not required.

13.c. Steam Generator Wide Range Level – Low

PRHR is also actuated when the SG Wide Range Level reaches its Low Setpoint. There are four wide range level channels for each steam generator and a two-out-of-four logic

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APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

is used. Four channels are provided to permit one channel to be in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function. This Function is required to be OPERABLE in MODES 1, 2, and 3 and in MODE 4 when the RCS is not being cooled by the RNS. This ensures that PRHR can be actuated in the event of a loss of the normal heat removal systems. In MODE 4 when the RCS is being cooled by the RNS, and in MODES 5 and 6, the SGs are not required to provide the normal RCS heat sink. Therefore, SG Wide Range Level is not required, and PRHR actuation on low wide range SG level is not required.

13.d. ADS Stages 1, 2, and 3 Actuation

PRHR is also actuated any time ADS Stages 1, 2, and 3 Actuation is initiated. The PRHR actuation Function requirements for the ADS Stages 1, 2, and 3 actuation are the same as the requirements for the ADS Stages 1, 2, and 3 Actuation Function, but only in MODES 2, 3, and 4, and in MODE 5 with the RCS pressure boundary intact.

13.e. CMT Actuation (Function 2)

PRHR is also actuated by all the Functions that actuate CMT injection. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead, Function 2 (CMT Actuation) is referenced for all initiating functions and requirements.

13.f. Pressurizer Water Level – High 3

PRHR is actuated when the pressurizer water level reaches its High 3 setpoint. This signal provides protection against a pressurizer overfill following an inadvertent core makeup tank actuation with consequential loss of offsite power. This Function is automatically unblocked when RCS pressure is above the P-19 setpoint. This Function is required to be OPERABLE in MODES 1, 2, and 3, and in MODE 4 when the RCS is not being cooled by the RNS and above the P-19 (RCS pressure) interlock. This Function is not required to be OPERABLE in MODES 5 and 6 because it is not required to mitigate DBA in these MODES.

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APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

14. Steam Generator Blowdown Isolation

The primary Function of the steam generator blowdown isolation is to ensure that sufficient water inventory is present in the steam generators to remove the excess heat being generated until the decay heat has decreased to within the PRHR HX capability.

This Function closes the isolation valves of the Steam Generator Blowdown System in both steam generators when a signal is generated from the PRHR HX Actuation or Steam Generator Narrow Range Water Level – Low. This Function is required to be OPERABLE in MODES 1, 2, and 3, and in MODE 4 when the RCS is not being cooled by the RNS. This Function is not required to be OPERABLE in MODE 4 if the steam generator blowdown line is isolated.

14.a. PRHR Heat Exchanger Actuation (Function 13)

Steam Generator Blowdown Isolation is also initiated by all Functions that initiate PRHR actuation. The Steam Generator Blowdown Isolation requirements for these Functions are the same as the requirements for the PRHR Actuation. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead, Function 13, PRHR HX Actuation, is referenced for all initiating Functions and requirements.

14.b. Steam Generator Narrow Range Level – Low

The Steam Generator Blowdown isolation is actuated when the Steam Generator Narrow Range Level reaches its Low Setpoint.

The LCO requires four channels per steam generator to be OPERABLE to satisfy the requirements with a two-out-of-four logic. Four channels are provided to permit one channel to be in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function. Setpoint reflects both steady state and adverse environmental instrument uncertainties as the detectors provide protection for an event that results in a harsh environment.

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APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

15. Boron Dilution Block

The block of boron dilution is accomplished by closing the CVS suction valves to demineralized water storage tanks, and aligning the boric acid tank to the CVS makeup pumps. This Function is actuated by Source Range Neutron Flux Doubling and Reactor Trip.

15.a. Source Range Neutron Flux Doubling

A signal to block boron dilution in MODES 2 or 3, when not critical or during an intentional approach to criticality, and MODES 4 or 5 is derived from source range neutron flow increasing at an excessive rate (source range flux doubling). This Function is not applicable in MODES 4 and 5 if the demineralized water makeup flowpath is isolated. The source range neutron detectors are used for this Function. The LCO requires four divisions to be OPERABLE. There are four divisions and two-out-of-four logic is used. On a coincidence of excessively increasing source range neutron flux in two of the four divisions, demineralized water is isolated from the makeup pumps and reactor coolant makeup is isolated from the reactor coolant system to preclude a boron dilution event. In MODE 6, a dilution event is precluded by the requirement in LCO 3.9.2 to close, lock and secure at least one valve in each unborated water source flow path.

15.b. Reactor Trip (Function 18.a)

Demineralized Water Makeup is also isolated by all the Functions that initiate a Reactor Trip. The isolation requirements for these Functions are the same as the requirements for the Reactor Trip Function. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead Function 18.a, (P-4 Reactor Trip Breakers), is referenced for all initiating Functions and requirements.

16. Chemical Volume and Control System Makeup Line Isolation

The CVS makeup line is isolated following certain events to prevent overfilling of the RCS. In addition, this line is isolated on High 2 containment radioactivity to provide containment isolation following an accident. This line is not isolated on a containment isolation signal, to allow the CVS makeup pumps to perform their defense-in-depth functions. However, if very high containment radioactivity exists (above the High 2 setpoint) this line is isolated.

BASES

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

A signal to isolate the CVS is derived from two-out-of-four high steam generator levels on either steam generator, two-out-of-four channels of pressurizer level indicating high or two-out-of-four channels of containment radioactivity indicating high. Four channels are provided to permit one channel to be in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function.

16.a. Steam Generator Narrow Range Water Level – High 2

Four channels of steam generator level are provided for each steam generator. Two-out-of-four channels on either steam generator indicating level greater than the setpoint will close the isolation valves for the CVS. This Function prevents adding makeup water to the RCS during a SGTR. This Function is required to be OPERABLE in MODES 1, 2, 3, and 4 with the RCS not being cooled by the RNS. This Function is not applicable in MODES 3 and 4 if the CVS makeup flowpath is isolated. This Function is not required to be OPERABLE in MODES 5 and 6 because the RCS pressure and temperature are reduced and a steam generator tube rupture event is not credible.

16.b. Pressurizer Water Level – High 1 Coincident with Safeguards Actuation

Four channels of pressurizer level are provided on the pressurizer. Two-out-of-four channels on indicating level greater than the High 1 setpoint coincident with a Safeguards Actuation signal (Function 1) will close the containment isolation valves for the CVS. This Function prevents the pressurizer level from reaching a level that could lead to water relief through the pressurizer safety valves during some DBAs. This Function is required to be OPERABLE in MODES 1, 2, and 3. This function is not required to be OPERABLE in MODES 4, 5, and 6, because it is not required to mitigate a DBA in these MODES. This Function is not applicable in MODE 3, if the CVS makeup flowpath is isolated.

BASES

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

16.c. Pressurizer Water Level – High 2

A signal to close the CVS isolation valves is generated on Pressurizer Water Level – High 2. This Function results from the coincidence of pressurizer level above the High 2 setpoint in any two of the four divisions. This Function is automatically blocked when the pressurizer pressure is below the P-11 permissive setpoint to permit pressurizer water solid conditions with the plant cold and to permit level makeup during plant cooldowns. This Function is automatically unblocked when RCS pressure is above the P-19 setpoint. This Function is required to be OPERABLE in MODES 1, 2, and 3 and in MODE 4 when the RCS is not being cooled by the RNS. This Function is not required to be OPERABLE in MODE 4 if the CVS makeup flowpath is isolated. This Function is not required to be OPERABLE in MODES 5 and 6 because it is not required to mitigate a DBA in these MODES.

16.d. Containment Radioactivity – High 2

Four channels of Containment Radioactivity – High 2 are required to be OPERABLE in MODES 1, 2, and 3 when the potential exists for a LOCA, to ensure that the radioactivity inside containment is not released to the atmosphere. This Function is not required to be OPERABLE in MODE 3 if the associated flowpath is isolated. This signal results from the coincidence of containment radioactivity above the High 2 Setpoint in any two of the four divisions. These Functions are not required to be OPERABLE in MODES 4, 5, and 6 because there is no credible release of radioactivity into the containment in these MODES that would result in a High 2 actuation.

16.e Manual Initiation

Manual Chemical Volume Control System Makeup Isolation is actuated by either of two switches in the main control room. Either switch closes Chemical Volume Control System Makeup valves. The LCO requires two switches to be OPERABLE.

BASES

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

16.f. Source Range Neutron Flux Doubling (Function 15.a)

Chemical Volume Control System Makeup Isolation is actuated by the Source Range Neutron Flux Doubling Function. The Source Range Neutron Flux Doubling Function requirements are the same as the requirements for Boron Dilution Block Function 15.a, Source Range Neutron Flux Doubling. Therefore, the requirements are not repeated in Table 3.3.2-1, and Function 15.a is referenced for all requirements.

17. Normal Residual Heat Removal System Isolation

The RNS suction line is isolated by closing the containment isolation valves on High 2 containment radioactivity to provide containment isolation following an accident. This line is isolated on a safeguards actuation signal. However, the valves may be reset to permit the RNS pumps to perform their defense-in-depth functions post-accident. Should a high containment radiation signal (above the High 2 setpoint) develop following the containment isolation signal, the RNS valves would re-close. A high containment radiation signal is indicative of a high RCS source term and the valves would re-close to assure offsite doses do not exceed regulatory limits.

17.a. Containment Radioactivity – High 2

A signal to isolate the normal residual heat removal system is generated from the coincidence of containment radioactivity above the High 2 setpoint in two-out-of-four channels. Four channels of Containment Radioactivity – High 2 are required to be OPERABLE in MODES 1, 2, and 3 when the potential exists for a LOCA, to ensure that the radioactivity inside containment is not released to the atmosphere. This Function is not required to be OPERABLE in MODE 3 if the RNS suction line is isolated. These Functions are not required to be OPERABLE in MODES 4, 5, and 6 because no DBA that could release radioactivity into the containment is considered credible in these MODES.

BASES

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

17.b. Safeguards Actuation (Function 1)

This Function is also initiated by all Functions that initiated the Safeguards Actuation signal. The requirements to isolate the normal residual heat removal system are the same as the requirements for the Safeguards Actuation Function. Therefore, the requirements are not repeated in Table 3.3.2.1. Instead, Function 1 is referenced for all initiating Functions and requirements.

17.c Manual Initiation

The operator can initiate RNS isolation at any time from the control room by simultaneously actuating two switches in the same actuation set. Because an inadvertent actuation of RNS isolation could have serious consequences, two switches must be actuated simultaneously to initiate isolation. There are two sets of two switches in the control room. Simultaneously actuating the two switches in either set will isolate the RNS in the same manner as the automatic actuation signal. Two Manual Initiation switches in each set are required to be OPERABLE to ensure no single failure disables the Manual Initiation Function.

18. ESFAS Interlocks

To allow some flexibility in unit operations, several interlocks are included as part of the ESFAS. These interlocks permit the operator to block some signals, automatically enable other signals, prevent some actions from occurring, and cause other actions to occur. The interlock Functions backup manual actions to ensure bypassable Functions are in operation under the conditions assumed in the safety analyses.

18.a. Reactor Trip, P-4

There are eight reactor trip breakers with two breakers in each division. The P-4 interlock is enabled when the breakers in two-out-of-four divisions are open. Additionally, the

BASES

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

P-4 interlock is enabled by all Automatic Reactor Trip Actuations. The Functions of the P-4 interlock are:

- Trip the main turbine
- Block boron dilution
- Isolate main feedwater coincident with low reactor coolant temperature (This function is not assumed in safety analysis therefore, it is not included in the technical specifications.)

The reactor trip breaker position switches that provide input to the P-4 interlock only function to energize or de-energize or open or close contacts. Therefore, this Function has no adjustable Trip Setpoint.

This Function must be OPERABLE in MODES 1, 2, and 3 when the reactor may be critical or approaching criticality. This Function does not have to be OPERABLE in MODE 4, 5, or 6 to trip the main turbine, because the main turbine is not in operation.

The P-4 Function does not have to be OPERABLE in MODE 4 or 5 to block boron dilution, because Function 15.a, Source Range Neutron Flux Doubling, provides the required block. In MODE 6, the P-4 interlock with the Boron Dilution Block Function is not required, since the unborated water source flow path isolation valves are locked closed in accordance with LCO 3.9.2.

18.b. Pressurizer Pressure, P-11

The P-11 interlock permits a normal unit cooldown and depressurization without Safeguards Actuation or main steam line and feedwater isolation. With pressurizer pressure channels less than the P-11 setpoint, the operator can manually block the Pressurizer pressure – Low, Steam Line Pressure – Low, and T_{cold} – Low Safeguards Actuation signals and the Steam Line Pressure – Low and T_{cold} – Low steam line isolation signals. When the Steam Line Pressure – Low is manually blocked, a main steam isolation signal on Steam Line Pressure-Negative Rate – High is enabled. This provides protection for an SLB by closure of the main steam isolation

BASES

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

valves. Manual block of feedwater isolation on T_{avg} – Low 1, Low 2, and T_{cold} – Low is also permitted below P-11. With pressurizer pressure channels ≥ P-11 setpoint, the Pressurizer Pressure – Low, Steam Line Pressure – Low, and T_{cold} – Low Safeguards Actuation signals and the Steam Line Pressure Low and T_{cold} – Low steam line isolation signals are automatically enabled. The feedwater isolation signals on T_{cold} – Low, T_{avg} – Low 1 and Low 2 are also automatically enabled above P-11. The operator can also enable these signals by use of the respective manual reset buttons. When the Steam Line Pressure – Low and T_{cold} – Low steam line isolation signals are enabled, the main steam isolation on Steam Line Pressure-Negative Rate – High is disabled. The Setpoint reflects only steady state instrument uncertainties.

This Function must be OPERABLE in MODES 1, 2, and 3 to allow an orderly cooldown and depressurization of the unit without the Safeguards Actuation or main steam or feedwater isolation. This Function does not have to be OPERABLE in MODE 4, 5, or 6, because plant pressure must already be below the P-11 setpoint for the requirements of the heatup and cooldown curves to be met.

18.c. Intermediate Range Neutron Flux, P-6

The Intermediate Range Neutron Flux, P-6 interlock is actuated when the respective NIS intermediate range channel goes approximately one decade above the minimum channel reading. Below the setpoint, the P-6 interlock automatically unblocks the flux doubling function, permitting the block of boron dilution. Normally, this Function is blocked by the main control room operator during reactor startup. This Function is required to be OPERABLE in MODE 2.

18.d. Pressurizer Level, P-12

The P-12 interlock is provided to permit midloop operation without core makeup tank actuation, reactor coolant pump trip, CVS letdown isolation, or purification line isolation. With pressurizer level channels less than the P-12 setpoint, the operator can manually block low pressurizer level signal used for these actuations. Concurrent with blocking CMT actuation

BASES

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

on low pressurizer level, ADS 4th Stage actuation on Low 2 RCS hot leg level is enabled. Also CVS letdown isolation on Low 1 RCS hot leg level is enabled. When the pressurizer level is above the P-12 setpoint, the pressurizer level signal is automatically enabled and a confirmatory open signal is issued to the isolation valves on the CMT cold leg balance lines. This Function is required to be OPERABLE in MODES 1, 2, 3, 4, 5, and 6.

18.e. RCS Pressure, P-19

The P-19 interlock is provided to permit water solid conditions (i.e., when the pressurizer water level is >92%) in lower MODES without automatic isolation of the CVS makeup pumps. With RCS pressure below the P-19 setpoint, the operator can manually block CVS isolation on High 2 pressurizer water level, and block Passive RHR actuation and Pressurizer Heater Trip on High 3 pressurizer water level. When RCS pressure is above the P-19 setpoint, these Functions are automatically unblocked. This Function is required to be OPERABLE IN MODES 1, 2, 3, and 4 with the RCS not being cooled by the RNS. When the RNS is cooled by the RNS, the RNS suction relief valve provides the required overpressure protection (LCO 3.4.14).

18.f. Reactor Trip Breaker Open, P-3

The P-3 interlock is provided to permit the block of automatic Safeguards Actuation after a predetermined time interval following automatic Safeguards Actuation.

The reactor trip breaker position switches that provide input to the P-3 interlock only function to energize or de-energize (open or close) contacts. Therefore, this Function does not have an adjustable Trip Setpoint.

19. Containment Air Filtration System Isolation

Some DBAs such as a LOCA may release radioactivity into the containment where the potential would exist for the radioactivity to be released to the atmosphere and exceed the acceptable site dose limits. Isolation of the Containment Air Filtration System provides protection to prevent radioactivity inside containment from being released to the atmosphere.

BASES

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

19.a. Containment Radioactivity – High 1

Three channels of Containment Radioactivity – High 1 are required to be OPERABLE in MODES 1, 2, 3, and 4 with the RCS not being cooled by the RNS, when the potential exists for a LOCA, to protect against radioactivity inside containment being released to the atmosphere. These Functions are not required to be OPERABLE in MODE 4 with the RCS being cooled by the RNS or MODES 5 and 6, because any DBA release of radioactivity into the containment in these MODES would not require containment isolation.

19.b. Containment Isolation (Function 3)

Containment Air Filtration System Isolation is also initiated by all Functions that initiate Containment Isolation. The Containment Air Filtration System Isolation requirements for these Functions are the same as the requirements for the Containment Isolation. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead, Function 3, Containment Isolation, is referenced for initiating Functions and requirements.

20. Main Control Room Isolation and Air Supply Initiation

Isolation of the main control room and initiation of the air supply provides a protected environment from which operators can control the plant following an uncontrolled release of radioactivity. This Function is required to be OPERABLE in MODES 1, 2, 3, and 4, and during movement of irradiated fuel because of the potential for a fission product release following a fuel handling accident, or other DBA.

20.a. Control Room Air Supply Radiation – High 2

Two radiation monitors are provided on the main control room air intake. If either monitor exceeds the High 2 setpoint, control room isolation is actuated.

20.b. Pressurizer Pressure – Low

This signal provides protection against a potential release of radioactivity due to a LOCA.

BASES

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

The transmitters are located inside containment, with the taps in the vapor space region of the pressurizer, and thus possibly experiencing adverse environmental conditions (LOCA, SLB inside containment). Therefore, the Trip Setpoint reflects the inclusion of both steady-state and adverse environmental instrument uncertainties.

The LCO requires four channels of Pressurizer Pressure – Low to be OPERABLE in MODES 1, 2, and 3 (above P-11, when the RCS boron concentration is below that necessary to meet the SDM requirements at an RCS temperature of 200°F), to mitigate the consequences of a high energy line rupture inside containment. Four channels are provided to permit one channel to be in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function. This signal may be manually blocked by the operator below the P-11 setpoint.

This Function is not required to be OPERABLE in MODE 3 below the P-11 setpoint.

21. Auxiliary Spray and Purification Line Isolation

The CVS maintains the RCS fluid purity and activity level within acceptable limits. The CVS purification line receives flow from the discharge of the RCPs. The CVS also provides auxiliary spray to the pressurizer. To preserve the reactor coolant pressure in the event of a break in the CVS loop piping, the purification line and the auxiliary spray line are isolated on a pressurizer water level Low 1 setpoint. This helps maintain reactor coolant system inventory.

21.a. Pressurizer Water Level – Low 1

A signal to isolate the purification line and the auxiliary spray line is generated upon the coincidence of pressurizer level below the Low 1 setpoint in any two-out-of-four divisions. This Function is required to be OPERABLE in MODES 1 and 2 to help maintain RCS inventory. In MODES 3, 4, 5, and 6, this Function is not needed for accident detection and mitigation.

21.b. Manual Chemical Volume Control System Makeup Isolation (Function 16.e)

The Auxiliary Spray and Purification Line Isolation is also initiated by the Manual Chemical Volume Control System Makeup Isolation Function. The requirements for this Function are the same as the requirements for Manual Chemical

BASES

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

Volume Control System Makeup Isolation (Function 16.e), but only apply in MODES 1 and 2. Therefore, the requirements are not repeated in Table 3.3.2-1, and Function 16.e is referenced for all requirements.

22. IRWST Injection Line Valve Actuation

The PXS provides core cooling by gravity injection and recirculation for decay heat removal following an accident. The IRWST has two injection flow paths. Each injection path includes a normally open motor operated isolation valve and two parallel lines, each isolated by one check valve and one squib valve in series. Manual initiation or automatic actuation on an ADS Stage 4 actuation signal or a coincident RCS Loops 1 and 2 Hot Leg Level-Low will generate a signal to open the IRWST injection line and actuate IRWST injection.

22.a. Manual Initiation

The operator can open IRWST injection line valves at any time from the main control room by actuating two IRWST injection actuation switches in the same actuation set. There are two sets of two switches each in the main control room. This Function is required to be OPERABLE in MODES 1, 2, 3, 4, 5, and 6.

22.b. ADS Stage 4 Actuation (Function 10)

An open signal will be issued to the IRWST injection isolation valves when an actuation signal is issued to the ADS Stage 4 valves. The requirements for this function are the same as the requirements for the ADS Stage 4 Actuation Function. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead, Function 10 is referenced for all initiating functions and requirements.

23. IRWST Containment Recirculation Valve Actuation

The PXS provides core cooling by gravity injection and recirculation for decay heat removal following an accident. The PXS has two containment recirculation flow paths. Each path contains two parallel flow paths, one path is isolated by a motor operated valve in series with a squib valve and one path is isolated by a check valve in series with a squib valve. Manual initiation or automatic actuation on a Safeguards Actuation signal coincident with a Low 3 level signal in the IRWST will open these valves.

BASES

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

23.a. Manual Initiation

The operator can open the containment recirculation valves at any time from the main control room by actuating two containment recirculation actuation switches in the same actuation set. There are two sets of two switches each in the main control room. This Function is required to be OPERABLE in MODES 1, 2, 3, 4, 5, and 6.

23.b. ADS Stage 4 Actuation Coincident with IRWST Level – Low 3

A low IRWST level coincident with a ADS Stage 4 Actuation signal will open the containment recirculation valves. Four channels are provided to permit one channel to be in trip or bypass indefinitely and still ensure that no single random failure will disable this trip Function. This Function is required to be OPERABLE in MODES 1, 2, 3, 4, 5, and 6, except when the ADS Stage 4 valves are open or an equivalent relief area is open. The required ADS valves or equivalent relief area are specified in LCO 3.4.12, ADS – Shutdown, RCS Intact and LCO 3.4.13, ADS – Shutdown, RCS Open.

24. Refueling Cavity Isolation

The containment isolation valves in the lines between the refueling cavity and the Spent Fuel Pool Cooling System are isolated on a Low spent fuel pool level.

24.a. Spent Fuel Pool Level – Low

In the event of a leak in the non-safety Spent Fuel Pool Cooling System, closure of the containment isolation valves on low spent fuel pool level in two of three channels will terminate draining of the refueling cavity. Since the transfer canal is open in MODE 6, the spent fuel pool level is the same as the refueling cavity.

Draining of the spent fuel pool, directly, through a leaking Spent Fuel Pool Cooling System is limited by the location of the suction piping, which is near the top of the pool. Therefore, closure of the containment isolation valves between the refueling cavity and the Spent Fuel Pool Cooling System is sufficient to terminate refueling cavity and spent fuel pool leakage through the Spent Fuel Pool Cooling System. This Function is required in MODE 6 to maintain water inventory in the refueling cavity.

BASES

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

25. ESF Logic

This LCO requires four sets of ESF coincidence logic, each set with one battery backed logic group OPERABLE to support automatic actuation. These logic groups are implemented as processor based actuation subsystems. The ESF coincidence logic provides the system level logic interfaces for the divisions.

25.a. Coincidence Logic

If one division of battery backed coincidence logic is OPERABLE, an additional single failure will not prevent ESF actuations because three divisions will still be available to provide redundant actuation for all ESF Functions. This Function is required to be OPERABLE in MODES 1, 2, 3, 4, 5, and 6.

26. ESF Actuation

This LCO requires that for each division of ESF actuation, one battery backed logic group be OPERABLE to support both automatic and manual actuation. The ESF actuation subsystems provide the logic and power interfaces for the actuated components.

26.a. Actuation Subsystem

If one battery backed logic group is OPERABLE for the ESF actuation subsystem in all four divisions, an additional single failure will not prevent ESF actuations because ESF actuation subsystems in the other three divisions are still available to provide redundant actuation for ESF Functions. The remaining cabinets in the division with a failed ESF actuation cabinet are still OPERABLE and will provide their ESF Functions. This Function is required to be OPERABLE in MODES 1, 2, 3, 4, 5, and 6.

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

27. Pressurizer Heater Trip

Pressurizer heaters are automatically tripped upon receipt of a core makeup tank operation signal or a Pressurizer Water Level – High 3 signal. This pressurizer heater trip reduces the potential for steam generator overfill and automatic ADS Stages 1, 2, and 3 actuation for a steam generator tube rupture event. Automatically tripping the pressurizer heaters reduces the pressurizer level swell for certain

BASES

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

non-LOCA events such as loss of normal feedwater, inadvertent CMT operation, and CVS malfunction resulting in an increase in RCS inventory. For small break LOCA analysis, tripping the pressurizer heaters supports depressurization of the RCS following actuation of the CMTs.

27.a. CMT Actuation (Function 2)

A signal to trip the pressurizer heaters is generated on a CMT actuation signal. The requirements for this function are the same as the requirements for the CMT Actuation Function, except this function is only required to be OPERABLE in MODES 1, 2, and 3, and in MODE 4 when the RCS is not being cooled by the RNS and above the P-19 (RCS pressure) interlock. Therefore, the requirements are not repeated in Table 3.3.2.1. Instead, Function 2 is referenced for initiating Functions and requirements and SR 3.3.2.9 also applies.

27.b. Pressurizer Water Level – High 3

A signal to trip the pressurizer heaters is generated when the pressurizer water level reaches its High 3 setpoint. This signal provides protection against a pressurizer overfill following an inadvertent core makeup tank actuation with consequential loss of offsite power. This Function is automatically unblocked when RCS pressure is above the P-19 setpoint. This Function is required to be OPERABLE in MODES 1, 2, and 3, and in MODE 4 when the RCS is not being cooled by the RNS and above the P-19 (RCS pressure) interlock. This Function is not required to be OPERABLE in MODES 5 and 6 because it is not required to mitigate DBA in these MODES.

28. Chemical and Volume Control System Letdown Isolation

The CVS provides letdown to the liquid radwaste system to maintain the pressurizer level. To help maintain RCS inventory in the event of a LOCA, the CVS letdown line is isolated on a Low 1 hot leg level signal in either of the RCS hot leg loops. This Function is required to be OPERABLE in MODE 4 with the RCS being cooled by the RNS. This Function is also required to be OPERABLE in MODE 5, and in MODE 6 with the water level < 23 feet above the top of the reactor vessel flange.

BASES

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

28.a. Hot Leg Level – Low 1

A signal to isolate the CVS letdown valves is generated upon the occurrence of a Low 1 hot leg level in either of the two RCS hot leg loops. This helps to maintain reactor system inventory in the event of a LOCA. This function can be blocked in Modes 1, 2 and 3 and is automatically reset when P-12 is first activated. This function may be manually reset as well. These letdown valves are also closed by all of the initiating Functions and requirements that generate the Containment Isolation Function in Function 3.

29. SG Power Operated Relief Valve and Block Valve Isolation

The Function of the SG Power Operated Relief Valve and Block Valve Isolation is to ensure that the SG PORV flow paths can be isolated during a SG tube rupture (SGTR) event. The PORV flow paths must be isolated following a SGTR to minimize radiological releases from the ruptured steam generator into the atmosphere. The PORV flow path is assumed to open due to high secondary side pressure, during the SGTR. Dose analyses take credit for subsequent isolation of the PORV flow path by the PORV and/or the block valve which receive a close signal on low steam line pressure. Additionally, the PORV flow path can be isolated manually.

This Function is required to be OPERABLE in MODES 1, 2, 3, and 4 with the RCS cooling not being provided by the Normal Residual Heat Removal System (RNS). In MODE 4 with the RCS cooling being provided by the RNS and in MODES 5 and 6, the steam generators are not being used for RCS cooling and the potential for a SGTR is minimized due to the reduced mass and energy in the RCS and steam generators.

29.a. Manual Initiation

Manual initiation of SG Power Operated Relief Valve and Block Valve Isolation can be accomplished from the control room. There are two switches in the control room and either switch can close the SG PORVs and PORV block valves. The LCO requires two switches to be OPERABLE.

BASES

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

29.b. Steam Line Pressure – Low

Steam Line Pressure – Low provides closure of the PORV flow paths in the event of SGTR in which the PORV(s) open, to limit the radiological releases from the ruptured steam generator into the atmosphere.

This Function is anticipatory in nature and has a typical leading/lag ratio of 50/5.

The LCO requires four channels of Steam Line Pressure – Low Function to be OPERABLE in MODES 1, 2, 3, and 4 with the RCS cooling not being provided by the RNS. Four channels are provided in each steam line to permit one channel to be in trip or bypass indefinitely and still ensure that no single random failure will disable this Function.

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

ACTIONS

A Note has been added in the ACTIONS to clarify the application of Completion Time rules. The Conditions of this specification may be entered independently for each Function listed on Table 3.3.2-1. The Completion Time(s) of the inoperable equipment of a Function will be tracked separately for each Function starting from the time the Condition was entered for that Function.

A second Note has been added to provide clarification that, more than one Condition is listed for each of the Functions in Table 3.3.2-1. If the Required Action and associated Completion Time of the first Condition listed in Table 3.3.2-1 is not met, the second Condition shall be entered.

In the event a channel's Nominal Trip Setpoint is not met, or the transmitter, or the Protection and Safety Monitoring System Division, associated with a specific Function is found inoperable, then all affected Functions provided by that channel must be declared inoperable and the LCO Condition(s) entered for the particular protection Function(s) affected. When the Required Channels are specified only on a per steam line, per loop, per SG, basis, then the Condition may be entered separately for each steam line, loop, SG, etc., as appropriate.

BASES

ACTIONS (continued)

When the number of inoperable channels in a trip function exceed those specified in one or other related Conditions associated with a trip function, then the plant is outside the safety analysis. Therefore, LCO 3.0.3 in MODES 1 through 4 and LCO 3.0.8 for MODE 5 and 6 should be immediately entered if applicable in the current MODE of operation.

A.1

Condition A is applicable to all ESFAS protection Functions. Condition A addresses the situation where one or more channels/divisions for one or more functions are inoperable at the same time. The Required Action is to refer to Table 3.3.2-1 and to take the Required Actions for the protection Functions affected. The Completion Times are those from the referenced Conditions and Required Actions.

B.1 and B.2

With one or two channels or divisions inoperable, one affected channel or division must be placed in a bypass or trip condition within 6 hours. If one channel or division is bypassed, the logic becomes two-out-of-three, while still meeting the single failure criterion. (A failure in one of the three remaining channels or divisions will not prevent the protective function.) If one channel or division is tripped, the logic becomes one-out-of-three, while still meeting the single failure criterion. (A failure in one of the three remaining channels or divisions will not prevent the protective function.) If one channel or division is bypassed and one channel or division is tripped, the logic becomes one-out-of-two, while still meeting the single failure criterion. The 6 hours allowed to place the inoperable channel(s) or division(s) in the bypassed or tripped condition is justified in Reference 6.

C.1

With one channel inoperable, the affected channel must be placed in a bypass condition within 6 hours. The 6 hours allowed to place the inoperable channel in the bypass condition is justified in Reference 6. If one CVS isolation channel is bypassed, the logic becomes one-out-of-one. A single failure in the remaining channel could cause a spurious CVS isolation. Spurious CVS isolation, while undesirable, would not cause an upset plant condition.

D.1

With one required division inoperable, the affected division must be restored to OPERABLE status within 6 hours.

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ACTIONS (continued)

Condition D applies to one inoperable required division of the P-3 & P-4 Interlocks (Functions 18.a and 18.f). With one required division inoperable, the 2 remaining OPERABLE divisions are capable of providing the required interlock function, but without a single failure. The P-3 & P-4 Interlocks are enabled when RTBs in two divisions are detected as open. The status of the other inoperable, non-required P-3 & P-4 division is not significant, since P-3 & P-4 divisions cannot be tripped or bypassed. In order to provide single failure tolerance, 3 required divisions must be OPERABLE.

Condition D also applies to one inoperable division of ESF coincidence logic or ESF actuation (Functions 25 and 26). The ESF coincidence logic and ESF actuation divisions are inoperable when their associated battery-backed subsystem is inoperable. With one inoperable division, the 3 remaining OPERABLE divisions are capable of mitigating all DBAs, but without a single failure.

The 6 hours allowed to restore the inoperable division is reasonable based on the capability of the remaining OPERABLE divisions to mitigate all DBAs and the low probability of an event occurring during this interval.

E.1

Condition E is applicable to manual initiation of:

- Safeguards Actuation;
- CMT Actuation;
- Containment Isolation;
- Steam Line Isolation;
- Main Feedwater Control Valve Isolation;
- Main Feedwater Pump Trip and Valve Isolation;
- ADS Stages 1, 2, & 3 Actuation;
- ADS Stage 4 Actuation;
- Passive Containment Cooling Actuation;
- PRHR Heat Exchanger Actuation;
- CVS Makeup Line Isolation;

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ACTIONS (continued)

- IRWST Injection Line Valve Actuation;
- IRWST Containment Recirculation Valve Actuation;
- Steam Generator PORV Flow Path Isolation.

This Action addresses the inoperability of the system level manual initiation capability for the ESF Functions listed above. With one switch or switch set inoperable for one or more Functions, the system level manual initiation capability is reduced below that required to meet single failure criterion. Required Action E.1 requires the switch or switch set for system level manual initiation to be restored to OPERABLE status within 48 hours. The specified Completion Time is reasonable considering that the remaining switch or switch set is capable of performing the safety function.

F.1, F.2.1, and F.2.2

Condition F is applicable to the Main Control Room (MCR) isolation and air supply initiation function which has only two channels of the initiating process variable. With one channel inoperable, the logic becomes one-out-of-one and is unable to meet single failure criterion. Restoring all channels to OPERABLE status ensures that a single failure will not prevent the protective Function.

Alternatively, radiation monitor(s) which provide equivalent information and control room isolation and air supply initiation manual controls may be verified to be OPERABLE. These provisions for operator action can replace one channel of radiation detection and system actuation. The 72 hour Completion Time is reasonable considering that there is one remaining channel OPERABLE and the low probability of an event occurring during this interval.

G.1

With one switch, switch set, channel, or division inoperable, the system level initiation capability is reduced below that required to meet single failure criterion. Therefore, the required switch, switch set, channel, and division must be returned to OPERABLE status within 72 hours. The specified Completion Time is reasonable considering the remaining switch, switch set, channel, or division is capable of performing manual initiation.

H.1

With one channel inoperable, the inoperable channel must be placed in a trip condition within 6 hours.

BASES

ACTIONS (continued)

Condition H is applicable to the PRHR heat exchangers actuation on SG Narrow Range Water Level Low coincident with Startup Feedwater Flow Low (Function 13.b). With one startup feedwater channel inoperable, the inoperable channel must be placed in a trip condition within 6 hours. If one channel is tripped, the interlock condition is satisfied. Condition H is also applicable to Refueling Cavity Isolation (Function 24.a). With one of the three spent fuel pool level channels inoperable, the inoperable channel must be placed in a trip condition within 6 hours. If one channel is tripped, the logic becomes one-out-of-two, while still meeting the single failure criterion. The specified Completion Time is reasonable considering the time required to complete this action.

I.1 and I.2

Condition I applies to IRWST containment recirculation valve actuation on safeguards actuation coincident with IRWST Level Low 3 (Function 23.b). With one or two channels inoperable, one affected channel must be placed in a bypass or trip condition within 6 hours. If one channel is bypassed, the logic becomes two-out-of-three, while still meeting the single failure criterion. (A failure in one of the three remaining channels will not prevent the protective function.) If one channel is tripped, the logic becomes one-out-of-three, while still meeting the single failure criterion. (A failure in one of the three remaining channels will not prevent the protective function.) If one channel is bypassed and one channel is tripped, the logic becomes one-out-of-two, while still meeting the single failure criterion. The 6 hours allowed to place the inoperable channel(s) in the bypassed or tripped condition is justified in Reference 6.

J.1 and J.2

Condition J applies to the P-6, P-11, P-12, and P-19 interlocks. With one or two required channel(s) inoperable, the associated interlock must be verified to be in its required state for the existing plant condition within 1 hour, or any Function channels associated with inoperable interlocks placed in a bypassed condition within 7 hours. Verifying the interlock state manually accomplishes the interlock role.

If one interlock channel is inoperable, the associated Function(s) must be placed in a bypass or trip condition within 7 hours. If one channel is bypassed, the logic becomes two-out-of-three, while still meeting the single failure criterion. (A failure in one of the three remaining channels will not prevent the protective function.) If one channel is tripped, the logic becomes one-out-of-three, while still meeting the single failure criterion. (A failure in one of the three remaining channels will not prevent the protective function.)

BASES

ACTIONS (continued)

If two interlock channels are inoperable, one channel of the associated Function(s) must be bypassed and one channel of the associated

Function(s) must be tripped. In this state, the logic becomes one-out-of-two, while still meeting the single failure criterion. The 7 hours allowed to place the inoperable channel(s) in the bypassed or tripped condition is justified in Reference 6.

K.1

LCO 3.0.8 is applicable while in MODE 5 or 6. Since irradiated fuel assembly movement can occur in MODE 5 or 6, the ACTIONS have been modified by a Note stating that LCO 3.0.8 is not applicable. If moving irradiated fuel assemblies while in MODE 5 or 6, the fuel movement is independent of shutdown reactor operations. Entering LCO 3.0.8 while in MODE 5 or 6 would require the optimization of plant safety, unnecessarily.

Condition K is applicable to the MCR Isolation and Air Supply Initiation (Function 20), during movement of irradiated fuel assemblies. If the Required Action and associated Completion Time of the first Condition listed in Table 3.3.2-1 is not met, the plant must suspend movement of the irradiated fuel assemblies immediately. The required action suspends activities with potential for releasing radioactivity that might enter the MCR. This action does not preclude the movement of fuel to a safe position.

L.1

If the required Action and associated Completion Time of the first Condition listed in Table 3.3.2-1 is not met, the plant must be placed in a MODE in which the LCO does not apply. This accomplished by placing the plant in MODE 3 within 6 hours. The allowed time is reasonable, based operating experience, to reach the required plant conditions from full power conditions in an orderly manner without challenging plant systems.

M.1 and M.2

If the Required Action and associated Completion Time of the first condition listed in Table 3.3.2-1 is not met, the plant must be placed in a MODE in which the LCO does not apply. This is accomplished by placing the plant in MODE 3 within 6 hours and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner without challenging plant systems.

BASES

ACTIONS (continued)

N.1 and N.2

If the Required Action and associated Completion Time of the first Condition listed in Table 3.3.2-1 is not met, the plant must be placed in a MODE in which the LCO does not apply. This is accomplished by placing the plant in MODE 3 within 6 hours and in MODE 4 with the RCS being cooled by the RNS within 24 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner without challenging plant systems.

O.1 and O.2

If the Required Action and associated Completion Time of the first Condition listed in Table 3.3.2-1 is not met, the plant must be placed in a MODE in which the LCO does not apply. This is accomplished by placing the plant in MODE 3 within 6 hours and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner without challenging plant systems.

P.1, P.2.1, and P.2.2

If the Required Action and associated Completion Time of the first Condition listed in Table 3.3.2-1 cannot be met, the plant must be placed in a condition where the instrumentation Function for valve isolation is no longer needed. This is accomplished by isolating the affected flow path(s) within 24 hours. By isolating the flow path from the demineralized water storage tank to the RCS, the need for automatic isolation is eliminated.

To assure that the flow path remains closed, the flow path shall be isolated by the use of one of the specified means (P.2.1) or the flow path shall be verified to be isolated (P.2.2). A means of isolating the affected flow path(s) includes at least one closed and deactivated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured within 7 days. If one of the P.2.1 specified isolation means is not used, the affected flow path shall be verified to be isolated once per 7 days.

This action is modified by a Note allowing the flow path(s) to be unisolated intermittently under administrative control. These administrative controls consist of stationing a dedicated operator at the

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ACTIONS (continued)

valve controls, who is in continuous communication with the control room. In this way the flow path can be rapidly isolated when a need for flow path isolation is indicated.

Q.1, Q.2.1, and Q.2.2

If the Required Action and associated Completion Time of the first Condition listed in Table 3.3.2-1 is not met, the plant must be placed in a condition where the instrumentation Function for valve isolation is no longer needed. This is accomplished by isolating the affected flow path by the use of at least one closed manual or closed and deactivated automatic valve within 6 hours.

If the flow path is not isolated within 6 hours the plant must be placed in a MODE in which the LCO does not apply. This is accomplished by placing the plant in MODE 3 within 12 hours and in MODE 4 within 18 hours.

This action is modified by a Note allowing the flow path(s) to be unisolated intermittently under administrative control. These administrative controls consist of stationing a dedicated operator at the valve controls, who is in continuous communication with the control room. In this way the flow path can be rapidly isolated when a need for flow path isolation is indicated.

R.1, R.2.1.1, R.2.1.2, and R.2.2

If the Required Action and associated Completion Time of the first Condition given in Table 3.3.2-1 is not met the plant must be placed in a condition in which the likelihood and consequences of an event are minimized. This is accomplished by placing the plant in MODE 3 within 6 hours and isolating the affected flow path(s) within 12 hours. To assure that the flow path remains closed, the affected flow path shall be verified to be isolated once per 7 days.

If the flow path is not isolated within 12 hours the plant must be placed in a MODE in which the LCO does not apply. This is accomplished by placing the plant in MODE 4 with the RCS cooling provided by the RNS within 30 hours. The allowed Completion Time is reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner without challenging plant systems.

This action is modified by a Note allowing the flow path(s) to be unisolated intermittently under administrative control. These administrative controls consist of stationing a dedicated operator at the

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ACTIONS (continued)

valve controls, who is in continuous communication with the control room. In this way the flow path can be rapidly isolated when a need for flow path isolation is indicated.

S.1, S.2.1.1, S.2.1.2, S.2.1.3, and S.2.2

If the Required Action and associated Completion Time of the first Condition listed in Table 3.3.2-1 is not met, the plant must be placed in a condition in which the likelihood and consequences of an event are minimized. This is accomplished by placing the plant in MODE 3 within 6 hours and in MODE 4 with the RCS cooling provided by the RNS within 24 hours. Once the plant has been placed in MODE 4 the affected flow path must be isolated within 30 hours. To assure that the flow path remains closed, the affected flow path shall be verified to be isolated once per 7 days.

If the flow path is not isolated within 12 hours, the plant must be placed in a MODE in which the LCO does not apply. This is accomplished by placing the plant in MODE 5 within 42 hours. The allowed Completion Time is reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner without challenging plant systems.

This action is modified by a Note allowing the flow path(s) to be unisolated intermittently under administrative control. These administrative controls consist of stationing a dedicated operator at the valve controls, who is in continuous communication with the control room. In this way the flow path can be rapidly isolated when a need for flow path isolation is indicated.

T.1.1, T.1.2.1, T.1.2.2, T.2.1, and T.2.2

If the Required Action and associated Completion Time of the first Condition listed in Table 3.3.2-1 is not met, the plant must be placed in a Condition in which the likelihood and consequences of an event are minimized. This is accomplished by isolating the affected flow path within 6 hours and isolating the affected flow path(s) by the use of at least one closed and deactivated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured within 7 days or verify the affected flow path is isolated once per 7 days.

If the flow path is not isolated within 6 hours the plant must be placed in a MODE in which the LCO does not apply. This is accomplished by placing the plant in MODE 3 within 12 hours and in MODE 5 within 42 hours.

BASES

ACTIONS (continued)

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

This action is modified by a Note allowing the flow path(s) to be unisolated intermittently under administrative control. These administrative controls consist of stationing a dedicated operator at the valve controls, who is in continuous communication with the control room. In this way the flow path can be rapidly isolated when a need for flow path isolation is indicated.

U.1 and U.2

If the Required Action and the associated Completion Time of the first Condition given in Table 3.3.2-1 is not met, and the required switch or switch set is not restored to OPERABLE status within 48 hours, the plant must be placed in a condition in which the likelihood and consequences of an event are minimized. This is accomplished by placing the plant in MODE 5 within 12 hours. Once in MODE 5, action shall be immediately initiated to open the RCS pressure boundary and establish ≥ 20% pressurizer level. The 12 hour Completion Time is a reasonable time to reach MODE 5 from MODE 4 with RCS cooling provided by the RNS (approximately 350°F) in an orderly manner without challenging plant systems. Opening the RCS pressure boundary assures that cooling water can be injected without ADS operation. Filling the RCS to provide ≥ 20% pressurizer level minimizes the consequences of a loss of decay heat removal event.

V.1, V.2.1, and V.2.2

If the Required Action and the associated Completion Time of the first Condition listed in Table 3.3.2-1 is not met and the required channel(s) is not bypassed within 6 hours, the inoperable channel(s) must be restored within 168 hours. The 168 hour Completion Time is based on the ability of the two remaining OPERABLE channels to provide the protective Function even with a single failure.

If the channel(s) is not restored within the 168 hour Completion Time, the plant shall be placed in a condition in which the likelihood and consequences of an event are minimized. This is accomplished by placing the plant in MODE 5 within 180 hours (the next 12 hours). Once in MODE 5, action shall be initiated to open the RCS pressure boundary and establish ≥ 20% pressurizer level. The 12 hours is a reasonable time

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ACTIONS (continued)

to reach MODE 5 from MODE 4 with RCS cooling provided by the RNS (approximately 350°F) in an orderly manner without challenging plant systems.

Opening the RCS pressure boundary assures that cooling water can be injected without ADS operation. Filling the RCS to provide ≥ 20% pressurizer level minimizes the consequences of a loss of decay heat removal event.

W.1, W.2, W.3, and W.4

If the Required Action and the associated Completion Time listed in Table 3.3.2-1 is not met while in MODES 5 and 6, the plant must be placed in a MODE in which the likelihood and consequences of an event are minimized. This is accomplished by immediately initiating action to be in MODE 5 with the RCS open and ≥ 20% pressurizer level or to be in MODE 6 with the upper internals removed. The flow path from the demineralized water storage tank to the RCS shall also be isolated by the used of at least one closed and de-activated automatic valve or closed manual valve. These requirements minimize the consequences of the loss of decay heat removal by maximizing RCS inventory and maintaining RCS temperature as low as practical. Additionally, the potential for a criticality event is minimized by isolation of the demineralized water storage tank and by suspension of positive reactivity additions.

X.1, X.2, and X.3

If the Required Action and the associated Completion Time of the first Condition listed in Table 3.3.2-1 is not met while in MODES 5 and 6, the plant must be placed in a MODE in which the likelihood and consequences of an event are minimized. This is accomplished by immediately initiating action to be in MODE 5 with the RCS open and ≥ 20% pressurizer level or to be in MODE 6 with the upper internals removed. These requirements minimize the consequences of the loss of decay heat removal by maximizing RCS inventory and maintaining RCS temperature as low as practical. Additionally, the potential for a criticality event is minimized by suspension of positive reactivity additions.

Y.1, Y.2, Y.3, and Y.4

If the Required Action and the associated Completion Time of the first Condition listed in Table 3.3.2-1 is not met while in MODE 4, with RCS cooling provided by the RNS, MODE 5, or MODE 6, the plant must be placed in a MODE in which the likelihood and consequences of an event

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ACTIONS (continued)

are minimized. If in MODE 4, this is accomplished by placing the plant in MODE 5 within 12 hours. The 12 hours is a reasonable time to reach MODE 5 from MODE 4 with RCS cooling provided by the RNS (approximately 350°F) in an orderly manner without challenging plant systems.

If in MODE 4 or 5, Required Action Y.3 requires initiation of action within 12 hours to close the RCS pressure boundary and establish ≥ 20% pressurizer level. The 12 hour Completion Time allows transition to MODE 5 in accordance with Y.2, if needed, prior to initiating action to open the RCS pressure boundary.

If in MODE 6, Required Action Y.4 requires the plant to be maintained in MODE 6 and initiation of action to establish the reactor cavity water level ≥ 23 feet above the top of the reactor vessel flange.

Required Actions Y.2, Y.3, and Y.4 minimize the consequences of a loss of decay heat removal event by optimizing conditions for RCS cooling in MODE 5 using the PRHR HX or in MODE 6 using IRWST injection. Additionally, maximizing RCS inventory and maintaining RCS temperature as low as practical further minimize the consequences of a loss of decay heat removal event. Closing the RCS pressure boundary in MODE 5 assures that PRHR HX cooling is available. Additionally, the potential for a criticality event is minimized by suspension of positive reactivity additions.

Z.1, Z.2.1, and Z.2.2

If the Required Action and associated Completion Time of the first Condition listed in Table 3.3.2-1 is not met, the plant must be placed in a condition where the instrumentation Function for valve isolation is no longer needed. This is accomplished by isolating the affected flow path by the use of at least one closed manual or closed and deactivated automatic valve within 6 hours.

If the flow path is not isolated within 6 hours, the plant must be placed in a MODE in which the LCO does not apply. This is accomplished by placing the plant in MODE 3 within 12 hours and in MODE 4 with RCS cooling provided by the RNS within 30 hours.

This Action is modified by a Note allowing the flow path(s) to be unisolated intermittently under administrative control. These administrative controls consist of stationing a dedicated operator at the

BASES

ACTIONS (continued)

valve controls, who is in continuous communication with the control room. In this way the flow path can be rapidly isolated when a need for flow path isolation is indicated.

AA.1.1, AA.1.2.1, AA1.2.2, AA.2.1, AA.2.2, and AA.2.3

If the Required Action and associated Completion Time of the first condition listed in Table 3.3.2-1 is not met, the plant must be placed in a condition where the instrumentation Function for valve isolation is no longer needed. This is accomplished by isolating the affected flow path within 24 hours. By isolating the CVS letdown flow path from the RCS, the need for automatic isolation is eliminated.

To assure that the flow path remains closed, the flow path shall be isolated by the use of one of the specified means (AA.1.2.1) or the flow path shall be verified to be isolated (AA.1.2.2). A means of isolating the affected flow path includes at least one closed and deactivated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured, within 7 days. If one of the P.2.1 specified isolation means is not used, the affected flow path shall be verified to be isolated once per 7 days.

This action is modified by a Note allowing the flow path to be unisolated intermittently under administrative control. These administrative controls consist of stationing a dedicated operator at the valve controls, who is in continuous communication with the control room. In this way the flow path can be rapidly isolated when a need for flow path isolation is indicated.

If the flow path cannot be isolated in accordance with Required Actions AA.1.1, AA.1.2.1 and AA.1.2.2, the plant must be placed in a MODE in which the likelihood and consequences of an event are minimized. If in MODE 4, this is accomplished by placing the plant in MODE 5 within 12 hours. The 12 hours is a reasonable time to reach MODE 5 from MODE 4 with RCS cooling provided by the RNS (approximately 350°F) in an orderly manner without challenging plant systems.

If in MODE 4 or 5, Required Action AA.2.2 requires initiation of action, within 12 hours, to establish > 20% pressurizer level. The 12 hour Completion Time allows transition to MODE 5 in accordance with AA.2.1, if needed, prior to initiating action to establish the pressurizer level.

BASES

ACTIONS (continued)

If in MODE 6, Required Action AA.2.3 requires the plant to be maintained in MODE 6 and initiation of action to establish the reactor cavity water level ≥ 23 feet above the top of the reactor vessel flange.

Required Actions AA.2.2 and AA.2.3 minimize the consequences of an event by optimizing conditions for RCS cooling in MODE 5 using the PRHR HX or in MODE 6 using IRWST injection.

BB.1 and BB.2

With one channel inoperable, the inoperable channel must be placed in bypass and the hot leg level continuously monitored.

If one channel is placed in bypass, automatic actuation will not occur. Continuous monitoring of the hot leg level provides sufficient information to permit timely operator action to ensure that ADS Stage 4 actuation can occur, if needed to mitigate events requiring RCS makeup, boration, or core cooling. Operator action to manually initiate ADS Stage 4 actuation is assumed in the analysis of shutdown events (Reference 11). It is also credited in the shutdown PRA (Reference 12) when automatic actuation is not available.

SURVEILLANCE REQUIREMENTS

The Surveillance Requirements for each ESF Function are identified by the Surveillance Requirements column of Table 3.3.2-1. A Note has been added to the Surveillance Requirement table to clarify that Table 3.3.2-1 determines which Surveillance Requirements apply to which ESF Functions.

SR 3.3.2.1

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

BASES

SURVEILLANCE REQUIREMENTS (continued)

Agreement criteria are determined by the plant staff, based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the match criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

The Surveillance Frequency is based on operating experience that demonstrates channel failure is rare. Automated operator aids may be used to facilitate performance of the CHANNEL CHECK.

SR 3.3.2.2

SR 3.3.2.2 is the performance of an ACTUATION LOGIC TEST. This test, in conjunction with the ACTUATION DEVICE TEST, demonstrates that the actuated device responds to a simulated actuation signal. The ESF coincidence logic and ESF actuation subsystems within a division are tested every 92 days on a STAGGERED TEST BASIS.

A test subsystem is provided with the protection and safety monitoring system to aid the plant staff in performing the ACTUATION LOGIC TEST. The test subsystem is designed to allow for complete functional testing by using a combination of system self-checking features, functional testing features, and other testing features. Successful functional testing consists of verifying that the capability of the system to perform the safety function has not failed or degraded.

For hardware functions this would involve verifying that the hardware components and connections have not failed or degraded. Generally this verification includes a comparison of the outputs from two or more redundant subsystems or channels.

Since software does not degrade, software functional testing involves verifying that the software code has not changed and that the software code is executing.

To the extent possible, protection and safety monitoring system functional testing is accomplished with continuous system self-checking features and the continuous functional testing features. The ACTUATION LOGIC TEST shall include a review of the operation of the test subsystem to verify the completeness and adequacy of the results.

If the ACTUATION LOGIC TEST can not be completed using the built-in test subsystem, either because of failures in the test subsystem or failures in redundant channel hardware used for functional testing, the ACTUATION LOGIC TEST can be performed using portable test equipment.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The Frequency of every 92 days on a STAGGERED TEST BASIS provides a complete test of all four divisions once per year. This frequency is adequate based on the inherent high reliability of the solid state devices which comprise this equipment; the additional reliability provided by the redundant subsystems; and the use of continuous diagnostic test features, such as deadman timers, memory checks, numeric coprocessor checks, cross-check of redundant subsystems, and tests of timers, counters, and crystal time basis, which will report a failure within these cabinets to the operator.

SR 3.3.2.3

SR 3.3.2.3 is the performance of a TADOT of the manual actuations, initiations, and blocks for various ESF Functions, the reactor trip breaker open (P-3), and the reactor trip (P-4) input from the IPCs. This TADOT is performed every 24 months.

The Frequency is based on the known reliability of the ESF Functions and the multichannel redundancy available, and has been shown to be acceptable through operating experience.

The SR is modified by a Note that excludes verification of setpoints from the TADOT. The functions have no setpoints associated with them.

SR 3.3.2.4

SR 3.3.2.4 is the performance of a CHANNEL CALIBRATION every 24 months or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor and the IPC.

The Frequency is based on operating experience and consistency with the refueling cycle.

This Surveillance Requirement is modified by a Note. The Note states that this test should include verification that the time constants are adjusted to the prescribed values where applicable.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.2.5

SR 3.3.2.5 is the performance of a CHANNEL OPERATIONAL TEST (COT) every 92 days.

A COT is performed on each required channel to provide reasonable assurance that the entire channel will perform the intended ESF Function.

A test subsystem is provided with the protection and safety monitoring system to aid the plant staff in performing the COT. The test subsystem is designed to allow for complete functional testing by using a combination of system self-checking features, functional testing features, and other testing features. Successful functional testing consists of verifying that the capability of the system to perform the safety function has not failed or degraded.

For hardware functions this would involve verifying that the hardware components and connections have not failed or degraded. Generally this verification includes a comparison of the outputs from two or more redundant subsystems or channels.

Since software does not degrade, software functional testing involves verifying that the software code has not changed and that the software code is executing.

To the extent possible, protection and safety monitoring system functional testing is accomplished with continuous system self-checking features and the continuous functional testing features. The COT shall include a review of the operation of the test subsystem to verify the completeness and adequacy of the results.

If the COT can not be completed using the built-in test subsystem, either because of failures in the test subsystem or failures in redundant channel hardware used for functional testing, the COT can be performed using portable test equipment.

The 92 day Frequency is based on Reference 6 and the use of continuous diagnostic test features, such as deadman timers, cross-check of redundant channels, memory checks, numeric coprocessor checks, and tests of timers, counters and crystal time bases, which will report a failure within the integrated protection cabinets to the operator.

During the COT, the protection and safety monitoring system cabinets in the division under test may be placed in bypass.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.2.6

This SR ensures the individual channel ESF 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 Trip Setpoint value at the sensor, to the point at which the equipment reaches the required functional state (e.g., valves in full open or closed position).

For channels that include dynamic transfer functions (e.g., lag, lead/lag, rate/lag, etc.), the response time test may be performed with the transfer functions set to one with the resulting measured response time compared to the appropriate Chapter 7 (Ref. 2) response time. Alternately, the response time test can be performed with the time constants set to their nominal value provided the required response time is analytically calculated assuming the time constants are set at their nominal values. The response time may be measured by a series of overlapping tests such that the entire response time is measured.

Response time may be verified by actual response time tests in any series of sequential, overlapping or total channel measurements, or by the summation of allocated sensor, signal processing and actuation logic response times with actual response time tests on the remainder of the channel. Allocations for sensor response times may be obtained from: (1) historical records based on acceptable response time tests (hydraulic, noise, or power interrupt tests), (2) in place, onsite, or offsite (e.g. vendor) test measurements, or (3) utilizing vendor engineering specifications. WCAP-13632-P-A, Revision 2, “Elimination of Pressure Sensor Response Time Testing Requirements” (Ref. 11), provides the basis and methodology for using allocated sensor response times in the overall verification of the channel response time for specific sensors identified in the WCAP. Response time verification for other sensor types must be demonstrated by test.

ESF RESPONSE TIME tests are conducted on an 24 month STAGGERED TEST BASIS. Testing of the devices, which make up the bulk of the response time, is included in the testing of each channel. The final actuation device in one train is tested with each channel. Therefore, staggered testing results in response time verification of these devices every 24 months. The 24 month Frequency is consistent with the typical refueling cycle and 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.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.2.7

SR 3.3.2.7 is the performance of an ACTUATION DEVICE TEST. This test, in conjunction with the ACTUATION LOGIC TEST, demonstrates that the actuated device responds to a simulated actuation signal. This Surveillance Requirement is applicable to the equipment which is actuated by the Protection Logic Cabinets except squib valves. The OPERABILITY of the actuated equipment is checked by exercising the equipment on an individual basis.

The Frequency of 24 months is based on the need to perform this surveillance during periods in which the plant is shutdown for refueling to prevent any upsets of plant operation.

This Surveillance Requirement is modified by a Note that states that actuated equipment, that is included in the Inservice Test (IST) Program, is exempt from this surveillance. The IST Program provides for exercising of the safety related valves on a more frequent basis. The results from the IST Program can therefore be used to verify OPERABILITY of the final actuated equipment.

SR 3.3.2.8

SR 3.3.2.8 is the performance of an ACTUATION DEVICE TEST, similar to that performed in SR 3.3.2.7, except this Surveillance Requirement is specifically applicable to squib valves. This test, in conjunction with the ACTUATION LOGIC TEST, demonstrates that the actuated device responds to a simulated actuation signal. The OPERABILITY of the squib valves is checked by performing a continuity check of the circuit from the Protection Logic Cabinets to the squib valve.

The Frequency of 24 months is based on the need to perform this surveillance during periods in which the plant is shutdown for refueling to prevent any additional risks associated with inadvertent operation of the squib valves.

SR 3.3.2.9

SR 3.3.2.9 is the performance of an ACTUATION DEVICE TEST. This test, in conjunction with the ACTUATION LOGIC TEST, demonstrates that the actuated device responds to a simulated actuation signal. This Surveillance Requirement is applicable to the circuit breakers which de-energize the power to the pressurizer heaters upon a pressurizer heater trip. The OPERABILITY of these breakers is checked by opening these breakers using the Plant Control System.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The Frequency of 24 months is based on the need to perform this surveillance during periods in which the plant is shutdown for refueling to prevent any upsets of plant operation. This Frequency is adequate based on the use of multiple circuit breakers to prevent the failure of any single circuit breaker from disabling the function and that all circuit breakers are tested.

REFERENCES

1. Chapter 6, “Engineered Safety Features.”
2. Chapter 7, “Instrumentation and Controls.”
3. Chapter 15, “Accident Analysis.”
4. Institute of Electrical and Electronic Engineers, IEEE-603-1991, “IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations,” June 27, 1991.
5. 10 CFR 50.49, “Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants.”
6. APP-GW-GSC-020, “Technical Specification Completion Time and Surveillance Frequency Justification.”
7. 10 CFR 50, Appendix A, “General Design Criteria for Nuclear Power Plants.”
8. NUREG-1218, “Regulatory Analysis for Resolution of USI A-47,” 4/88.
9. WCAP-16361-P, “Westinghouse Setpoint Methodology for Protection Systems – AP1000,” May 2006 (proprietary).
10. ESBU-TB-97-01, Westinghouse Technical Bulletin, “Digital Process Rack Operability Determination Criteria,” May 1, 1997.
11. WCAP-13632-P-A (Proprietary) and WCAP-13787-A (Non-Proprietary), Revision 2, “Elimination of Pressure Sensor Response Time Testing Requirements,” January 1996.

B 3.3 INSTRUMENTATION

B 3.3.3 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 main control room operators during accident situations. These plant variables provide the necessary information to assess the process of accomplishing or maintaining critical safety functions. The instruments which monitor these variables are designated in accordance with Reference 1.

The OPERABILITY of the PAM Instrumentation ensures that there is sufficient information available on selected plant parameters to monitor and assess plant status and behavior following an accident. This capability is consistent with the recommendations of Reference 1.

A PAM CHANNEL shall extend from the sensor up to the display device, and shall include the sensor (or sensors), the signal conditioning, any associated datalinks, the display device, any signal gathering or processing subsystems, and any data processing subsystems. Note that for digital PAM CHANNELs, the information may be displayed on multiple display devices. For this case, the PAM CHANNEL shall extend to any available qualified display device.

The instrument channels required to be OPERABLE by this LCO include two classes of parameters identified during unit specific implementation of Regulatory Guide 1.97 as Type A and Category 1 variables. The unit specific implementation of Regulatory Guide 1.97 has not identified any Type A variables, therefore, only Category 1 variables are specified.

APPLICABLE SAFETY ANALYSES

The PAM Instrumentation ensures that the main control room operating staff can:

- Determine whether systems important to safety are performing their intended functions;
- Determine the likelihood of a gross breach of the barriers to radioactivity release;
- Determine if a gross breach of a barrier has occurred; and
- Initiate action necessary to protect the public and to estimate the magnitude of any impending threat.

BASES

APPLICABLE SAFETY ANALYSES (continued)

PAM Instrumentation that is required in accordance with Regulatory Guide 1.97 satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The PAM instrumentation LCO provides OPERABILITY requirements for those monitors which provide information required by the control room operators to assess the process of accomplishing or maintaining critical safety functions. This LCO addresses those Regulatory Guide 1.97 instruments which are listed in Table 3.3.3-1.

The OPERABILITY of the PAM Instrumentation ensures there is sufficient information available on selected plant parameters to monitor and assess plant status following an accident. This capability is consistent with the recommendations of Reference 1.

Category 1 non-type A variables are required to meet Regulatory Guide 1.97 Category 1 (Ref. 1) design and qualification requirements for seismic and environmental qualification, single-failure criterion, utilization of emergency standby power, immediately accessible display, continuous readout, and recording of display.

Listed below are discussions of the specified instrument functions listed in Table 3.3.3-1. Each of these is a Category 1 variable.

1. Intermediate Range Neutron Flux

   Neutron Flux indication is provided to verify reactor shutdown. The neutron flux intermediate range is sufficient to cover the full range of flux that may occur post accident.

   Neutron flux is used for accident diagnosis, verification of subcriticality, and diagnosis of positive reactivity insertion.

2, 3. Reactor Coolant System (RCS) Wide Range Hot and Cold Leg Temperature

   RCS Hot and Cold Leg Temperatures are provided for verification of core cooling and long-term surveillance. The channels provide indication over a range of 50°F to 700°F.

   In addition to this, RCS cold leg temperature is used in conjunction with RCS hot leg temperature to verify the plant conditions necessary to establish natural circulation in the RCS.

BASES

LCO (continued)

4. RCS Pressure

RCS wide range pressure is provided for verification of core cooling and RCS integrity long term surveillance.

5. Pressurizer Pressure and RCS Subcooling Monitor

Pressurizer Pressure is used to determine RCS Subcooling. The RCS Subcooling Monitor is provided for verification of core cooling. Subcooling margin is available when the RCS pressure is greater than the saturation pressure corresponding to the core exit temperature. Inputs to the Subcooling Monitor are pressurizer pressure and RCS hot leg temperature.

6. Containment Water Level

Containment Water Level is used to monitor the containment environment during accident conditions. The containment water level can also provide information to the operators that the various stages of safety injection along with system depressurization are progressing.

7. Containment Pressure

The containment pressure transmitters monitor the containment pressure over the range of -5 to 10 psig. This provides information on post accident containment pressure and containment integrity.

8. Containment Pressure (Extended Range)

The extended range containment pressure transmitters are instruments that operators use for monitoring the potential for breach of containment, a fission product barrier. The extended range sensors monitor containment pressure over the range of 0 to 240 psig.

9. Containment Area Radiation (High Range)

Containment Area Radiation is provided to monitor for the potential of significant radiation releases and to provide release assessment for use by operators in determining the need to invoke site emergency plans.

BASES

LCO (continued)

10. Pressurizer Level and Associated Reference Leg Temperature

Pressurizer level is provided to monitor the RCS coolant inventory. During an accident, operation of the safeguards systems can be verified based on coolant inventory indicators.

The reference leg temperature is included in the Technical Specification since it is used to compensate the level signal.

11. In-Containment Refueling Water Storage Tank (IRWST) Water Level

The IRWST provides a long term heat sink for non-LOCA events and is a source of injection flow for LOCA events. When the IRWST is a heat sink, the level will change due to increased volume associated with the temperature increase. When saturation temperature is reached, the IRWST will begin steaming and initially lose mass to the containment atmosphere until condensation occurs on the steel containment shell which is cooled by the passive containment cooling system. The condensate is returned to the IRWST via a gutter.

During a LOCA, the IRWST is available for injection. Depending on the severity of the event, when a fully depressurized RCS has been achieved, the IRWST will inject by gravity flow.

12. Passive Residual Heat Removal (PRHR) Flow and PRHR Outlet Temperature

PRHR Flow is provided to monitor primary system heat removal during accident conditions when the steam generators are not available. PRHR provides primary protection for non-LOCA events when the normal heat sink is lost.

PRHR outlet temperature is provided to monitor primary system heat removal during accident conditions when the steam generators are not available. PRHR provides primary protection for non-LOCA events when the normal heat sink is lost.

13, 14, 15, 16. Core Exit Temperature

Core Exit Temperature is provided for verification and long term surveillance of core cooling.

BASES

LCO (continued)

An evaluation was made of the minimum number of valid core exit thermocouples necessary for In-Core Cooling (ICC) detection. The evaluation determined the reduced complement of core exit thermocouples necessary to detect initial core recovery and trend the ensuing core heatup. The evaluations account for core nonuniformities including incore effects of the radial decay power distribution and excore effects of condensate runback in the hot legs and nonuniform inlet temperatures. Based on these evaluations, adequate ICC detection is assured with two valid core exit thermocouples per quadrant. Core Exit Temperature is also used for plant stabilization and cooldown monitoring.

Two OPERABLE channels of Core Exit Temperature are required in each quadrant to provide indication of radial distribution of the coolant temperature rise across representative regions of the core. Power distribution symmetry was considered in determining the specific number and locations provided for diagnosis of local core problems. Two thermocouples in each of the two divisions ensure a single failure will not disable the ability to determine the temperature at two locations within a quadrant.

17. Passive Containment Cooling System (PCS) Storage Tank Level and PCS Flow

The PCS must be capable of removing the heat from the containment following a postulated LOCA or steam line break (SLB). The tank level instruments provide indication that sufficient water is available to meet this requirement. The PCS flow instrument provides a diverse indication of the PCS heat removal capability.

18. Remotely Operated Containment Isolation Valve Position

The Remotely Operated Containment Isolation Valve Position is provided for verification of containment OPERABILITY.

19. IRWST to RNS Suction Valve Status

The position of the motor-operated valve in the line from the IRWST to the pump suction header is monitored to verify that the valve is closed following postulated events. The valve must be closed to prevent loss of IRWST inventory into the RNS.

BASES

APPLICABILITY

The PAM instrumentation LCO is applicable in MODES 1, 2, and 3. These variables provide the information necessary to assess the process of accomplishing or maintaining critical safety functions following Design Basis Accidents (DBAs). The applicable DBAs are assumed to occur in MODES 1, 2, and 3. In MODES 4, 5, and 6, plant conditions are such that the likelihood of an event that would require PAM instrumentation is low; therefore, the PAM instrumentation is not required to be OPERABLE in these MODES.

ACTIONS

The ACTIONS Table has been modified by two Notes.

The first Note 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 plant shutdown. This exception is acceptable due to the passive function of the instruments, the operator's ability to respond to an accident using alternate instruments and methods, and low probability of an event requiring these instruments.

The second Note in the ACTIONS clarifies the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed on Table 3.3.3-1. The Completion Time(s) of the inoperable channel(s) 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 which is inoperable, the required inoperable channel must be restored to OPERABLE status within 30 days. The 30 day Completion Time is based on operating experience and 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

The Required Action directs actions to be taken in accordance with Specification 5.6.7 immediately. Each time an inoperable channel has not met Required Action A.1, and the associated Completion Time has expired, Condition B is entered.

BASES

ACTIONS (continued)

C.1

When one or more Functions have two required channels which are inoperable, (two channels inoperable in the same Function), one channel in the Function should be restored to OPERABLE status within 7 days. The Completion Time of 7 days is based on the relatively low probability of an event requiring PAM instrument operation and the availability of alternate means to obtain the required information.

Continuous operation with two required channels inoperable in a Function is not acceptable because the alternate indications may not fully meet all performance 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

This Required Action directs entry into the appropriate Condition referenced in Table 3.3.3-1. The applicable Condition referenced in the Table is Function dependent.

Each time an inoperable channel has not met any Required Action of Condition C, and the associated Completion Time has expired, Condition D is entered for that channel and provides for transfer to the appropriate subsequent Condition.

E.1 and E.2

If the Required Action and associated Completion Time of Condition C are not met for the Functions in Table 3.3.3-1, the plant must be placed in a MODE in which the LCO does not apply. This is done by placing the plant in at least MODE 3 within 6 hours and MODE 4 within 12 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

The following SRs apply to each PAM instrumentation function in Table 3.3.3-1:

SR 3.3.3.1

Performance of the CHANNEL CHECK once every 31 days verifies that a gross instrumentation failure has not occurred. A CHANNEL CHECK is a

BASES

SURVEILLANCE REQUIREMENTS (continued)

comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION. The high radiation instrumentation should be compared to similar plant instruments located throughout the plant.

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 match 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 match criteria, it is an indication that the channels are OPERABLE.

As specified in the SR, a CHANNEL CHECK is only required for those channels that are normally energized.

The Frequency of 31 days is based on operating experience with regard to channel OPERABILITY and drift, which demonstrates that failure of more than one channel of a given function in any 31 day interval is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of those displays associated with the required channels of this LCO.

SR 3.3.3.2

A CHANNEL CALIBRATION is performed every 24 months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop including the sensor. The test verifies that the channel responds to the measured parameter with the necessary range and accuracy. This SR is modified by a Note that excludes neutron detectors. The calibration method for neutron detectors is specified in the Bases of LCO 3.3.1, “Reactor Trip System (RTS) Instrumentation.” RTD and Thermocouple channels are to be calibrated in place using cross-calibration techniques. The Frequency is based on operating experience and consistency with the typical industry refueling cycle.

BASES

REFERENCES

1. Regulatory Guide 1.97, Rev. 3, “Instrumentation for Light-Water Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident,” U.S. Nuclear Regulatory Commission.

B 3.3 INSTRUMENTATION

B 3.3.4 Remote Shutdown Workstation (RSW)

BASES

BACKGROUND

The RSW provides the control room operator with sufficient displays and controls to place and maintain the unit in a safe shutdown condition from a location other than the control room. This capability is necessary to protect against the possibility that the control room becomes inaccessible. Passive residual heat removal (PRHR), the core makeup tanks (CMTs), and the in-containment refueling water storage tank (IRWST) can be used to remove core decay heat. The use of passive safety systems allows extended operation in MODE 4.

If the control room becomes inaccessible, the operators can establish control at the RSW and place and maintain the unit in MODE 4 with T_{avg} < 350°F. The unit can be maintained safely in MODE 4 with T_{avg} < 350°F for an extended period of time.

The OPERABILITY of the remote shutdown control and display functions ensures there is sufficient information available on selected unit parameters to place and maintain the unit in MODE 4 with T_{avg} < 350°F should the control room become inaccessible.

APPLICABLE SAFETY ANALYSES

The RSW 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 4 with T_{avg} < 350°F.

The criteria governing the design and the specific system requirements of the RSW are located in 10 CFR 50, Appendix A, GDC 19 (Ref. 1).

Since the passive safety systems alone can establish and maintain safe shutdown conditions for the unit, nonsafety systems are not required for safe shutdown of the unit. Therefore, no credit is taken in the safety analysis for nonsafety systems.

The RSW satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

LCO

The RSW LCO provides the OPERABILITY requirements of the displays and controls necessary to place and maintain the unit in MODE 4 from a location other than the control room.

The RSW is OPERABLE if the display instrument and control functions needed to support the RSW are OPERABLE.

BASES

LCO (continued)

The RSW covered by this LCO does not need to be energized to be considered OPERABLE. This LCO is intended to ensure the RSW will be OPERABLE if unit conditions require that the RSW be placed in operation.

APPLICABILITY

The RSW LCO is applicable in MODES 1, 2, and 3 and in MODE 4 with $T_{avg} \geq 350°F$. This is required so that the facility can be placed and maintained in MODE 4 for an extended period of time from a location other than the control room.

This LCO is not applicable in MODE 4 with $T_{avg} < 350°F$ or in MODE 5 or 6. In these MODES, the unit is already subcritical and in a condition of reduced Reactor Coolant System (RCS) energy. Under these conditions, considerable time is available to restore necessary instrument control functions if control room instruments or controls become unavailable.

ACTIONS

The Note 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 the RSW and because the equipment can generally be repaired during operation without significant risk of a spurious trip.

A.1

Condition A addresses the situation where the RSW is inoperable. The Required Action is to restore the RSW to OPERABLE status within 30 days. The Completion Time is based on operating experience and the low probability of an event that would require evacuation of the control room.

B.1 and B.2

If the Required Action and associated Completion Time of Condition A is not met, 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 with $T_{avg} < 350°F$ 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.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.4.1

SR 3.3.4.1 verifies that each required RSW transfer switch performs the required functions. This ensures that if the control room becomes inaccessible, the unit can be placed and maintained in MODE 4 with T_{avg} < 350°F from the RSW. The 24 month Frequency was developed considering it is prudent that these types of surveillances be performed during a unit outage. However, this surveillance is not required to be performed only during a unit outage. This is due to the unit conditions needed to perform the surveillance and the potential for unplanned transients if the surveillance is performed with the reactor at power. Operating experience demonstrates that RSW transfer switches usually pass the surveillance test when performed on the 24 month Frequency.

SR 3.3.4.2

This Surveillance verifies that the RSW communicates controls and indications with Divisions A, B, C, and D of the PMS. Communication is accomplished by use of separate multiplexers for each division. The operator can select the controls and indications available through each PMS division.

The Frequency is based on the known reliability of the Functions and the redundancy available, and has been shown to be acceptable through operating experience.

SR 3.3.4.3

SR 3.3.4.3 verifies the OPERABILITY of the RSW hardware and software by performing diagnostics to show that operator displays are capable of being called up and displayed to an operator at the RSW. The RSW has several video display units which can be used by the operator. The video display units are identical to that provided in the control room and the operator can display information on the video display units in a manner which is identical to the way the information is displayed in the control room. The operator normally selects an appropriate set of displays based on the particular operational goals being controlled by the operator at the time. Each display consists of static graphical and legend information which is contained within the display processor associated with each video display unit and dynamic data which is updated by the data display system.

The Frequency of 24 months is based on the use of the data display capability in the control room as part of the normal unit operation and the

BASES

SURVEILLANCE REQUIREMENTS (continued)

availability of multiple video display units at the RSW. The Frequency of 24 months is based upon operating experience and consistency with control room hardware and software.

SR 3.3.4.4

SR 3.3.4.4 is the performance of a TRIP ACTUATING DEVICE OPERATIONAL TEST (TADOT) every 24 months. This test should verify the OPERABILITY of the reactor trip breakers (RTBs) open and closed indication on the RSW by actuating the RTBs. The Frequency of 24 months was chosen because the RTBs may not be exercised while the facility is at power and is based on operating experience and consistency with the refueling outage.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 19.

2. Section 7.4.1, “Safe Shutdown.”

B 3.3 INSTRUMENTATION

B 3.3.5 Diverse Actuation System (DAS) Manual Controls

BASES

BACKGROUND

The Diverse Actuation System (DAS) manual controls provide non-Class 1E backup controls in case of common-mode failure of the Protection and Safety Monitoring System (PMS) automatic and manual actuations evaluated in the AP1000 PRA. These DAS manual controls are not credited for mitigating accidents in the DCD Chapter 15 analyses.

The specific DAS controls were selected based on PRA risk importance as discussed in Reference 1. As noted in Reference 1, electrical power for these controls and instrument indications need not be covered by Technical Specifications. The rational is that these controls use the same nonsafety-related power supply used by the plant control system. This power is required to be available to support normal operation of the plant. With offsite power available, there are several sources to provide this power including AC power to non-Class 1E battery chargers, AC power to rectifiers, and non-Class 1E batteries. As a result, with offsite power available it is very likely that power will be available for these DAS controls. If offsite power is not available, then there is still the likelihood that the non-1E batteries or the non-1E diesel generators will be available. Even if these sources are unavailable, the desired actions will occur without operator action for the more probable events. The rods will insert automatically on loss of offsite power. The passive residual heat removal heat exchanger (PRHR HX), core makeup tanks (CMT), passive containment cooling system (PCS), and containment isolation features are initiated by operation of fail-safe, air-operated valves. If all offsite and onsite AC power is lost, the instrument air system will depressurize by the time these functions are needed in the 1-hour time frame.

Instrument readouts are expected to be available even in case of complete failure of the PMS due to common cause failure. These instruments include both DAS and PLS instruments. They are powered by DC sources for 24 or 72 hours following a loss of AC power, as described in DCD Section 8.3.2. As discussed above, it is expected that AC power will be available to power the instruments. Even if the operators have no instrument indications, they are expected to actuate the controls most likely to be needed (PRHR HX, CMT, PCS, and containment isolation). If all AC power fails, then the rods will drop and the air-operated valves will go to their fail-safe positions.

The DAS uses equipment from sensor output to the final actuated device that is diverse from the PMS to automatically initiate a reactor trip, or to manually actuate the identified safety-related equipment. DCD Section 7.7.1.11 (Ref. 2) provides a description of the DAS.

BASES

APPLICABLE SAFETY ANALYSES

The DAS manual controls are required to provide a diverse capability to manually trip the reactor and actuate the specified safety-related equipment, based on risk importance in the AP1000 PRA.

The DAS manual controls are not credited for mitigating accidents in the DCD Chapter 15 safety analyses.

The AP1000 PRA, Appendix A, provides additional information, including the thermal and hydraulic analyses of success sequences used in the PRA.

The DAS manual controls satisfy Criterion 4 of 10 CFR 50.36(c)(2)(ii).

LCO

The DAS LCO provides the requirements for the OPERABILITY of the DAS manual trip and actuation controls necessary to place the reactor in a shutdown condition and to remove decay heat in the event that the PMS automatic actuation and manual controls are inoperable.

APPLICABILITY

The DAS manual controls are required to be OPERABLE in the MODES specified in Table 3.3.5-1.

The manual DAS reactor trip control is required to be OPERABLE in MODES 1 and 2 to mitigate the effects of an ATWS event occurring during power operation.

The other manual DAS actuation controls are required to be available in the plant MODES specified, based on the need for operator action to actuate the specified components during events that may occur in these various plant conditions, as identified in the AP1000 PRA.

ACTIONS

A.1

Condition A applies when one or more DAS manual controls are inoperable.

The Required Action A.1 to restore the inoperable DAS manual control(s) to OPERABLE status within 30 days is reasonable because the DAS is a separate and diverse non-safety backup system for the manual reactor trip and manual safety-related equipment actuation controls. The 30-day Completion Time allows sufficient time to repair an inoperable manual DAS control but ensures the control is repaired to provide backup protection.

BASES

ACTIONS (continued)

B.1 and B.2

Condition B applies when Required Action A cannot be completed for the DAS manual reactor trip control within the required completion time of 30 days.

Required Action B.1 requires SR 3.3.1.6, “Perform TADOT” for the reactor trip breakers, to be performed once per 31 days, instead of once every 92 days. Condition A of Example 1.3-6 illustrates the use of the Completion Time for Required Action B.1. The initial performance of SR 3.3.1.6 on the first division (since it is performed on a STAGGERED TEST BASIS) must be completed within 31 days of entering Condition B. The normal surveillance test frequency requirements for SR 3.3.1.6 must still be satisfied while performing SR 3.3.1.6 for Required Action B.1. The predominant failure requiring the DAS manual reactor trip control is common-mode failure of the reactor trip breakers. This change in surveillance frequency for testing the reactor trip breakers increases the likelihood that a common-mode failure of the reactor trip breakers would be detected while the DAS manual reactor trip control is inoperable. This reduces the likelihood that a diverse manual reactor trip is required. It is not required to perform a TADOT for the manual actuation control. The manual reactor trip control is very simple, highly reliable, and does not use software in the circuitry. Although the DAS manual controls are non-Class 1E, they have been shown to be PRA risk important as discussed in Reference 1. The impact of an inoperable DAS manual control is compensated for by increasing the reactor trip breaker surveillance frequency from once every 92 days to once every 31 days.

Action B.2 requires that the inoperable DAS manual reactor trip control be restored to OPERABLE status prior to entering MODE 2 following any plant shutdown to MODE 5 while the control is inoperable. This ACTION is provided to ensure that all DAS manual controls are restored to OPERABLE status following the next plant shutdown.

C.1 and C.2

Condition C applies when Required Action A cannot be completed for any DAS manual actuation control (other than reactor trip) within the required completion time of 30 days.

Required Action C.1 requires SR 3.3.2.2, “Perform ACTUATION LOGIC TEST,” to be performed once per 31 days, instead of once every 92 days. Condition A of Example 1.3-6 illustrates the use of the Completion Time for Required Action C.1. The initial performance of SR 3.3.2.2 on the first

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ACTIONS (continued)

division (since it is performed on a STAGGERED TEST BASIS) must be completed within 31 days of entering Condition C. The normal surveillance test frequency requirements for SR 3.3.2.2 must still be satisfied while performing SR 3.3.2.2 for Required Action C.1. The predominant failure requiring the DAS manual actuation control is common-mode failure of the PMS actuation logic software or hardware. This change in surveillance frequency for actuation logic testing increases the likelihood that a common-mode failure of the PMS actuation logic from either cause would be detected while any DAS manual actuation control is inoperable. This reduces the likelihood that a diverse component actuation is required. It is not required to perform a TADOT for the manual actuation control device since the manual actuation control devices are very simple and highly reliable. Although the DAS manual controls are non-Class 1E, they have been shown to be PRA risk important as discussed in Reference 1. The impact of an inoperable DAS manual control is compensated for by increasing the automatic actuation surveillance frequency from once every 92 days to once every 31 days.

Action C.2 requires that the inoperable DAS manual actuation control(s) be restored to OPERABLE status prior to entering MODE 2 following any plant shutdown to MODE 5 while the control is inoperable. This ACTION is provided to ensure that all DAS manual controls are restored to OPERABLE status following the next plant shutdown.

D.1 and D.2

Condition D is entered if the Required Action associated with Condition B or C is not met within the required Completion Time.

Required Actions D.1 and D.2 ensure that the plant is placed in a condition where the probability and consequences of an event are minimized. The allowed Completion Times are reasonable based on plant operating experience, for reaching the required plant conditions from full power conditions in an orderly manner, without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.3.5.1

SR 3.3.5.1 is the performance of a TADOT of the DAS manual trip and actuation controls for the specified safety-related equipment. This TADOT is performed every 24 months.

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SURVEILLANCE REQUIREMENTS (continued)

The Frequency is based on the known reliability of the DAS functions and has been shown to be acceptable through operating experience.

The SR is modified by a Note that excludes verification of the setpoints from the TADOT. The functions have no setpoints associated with them.

REFERENCES

1. WCAP-15985, "AP1000 Implementation of the Regulatory Treatment of Nonsafety-Related Systems Process," Revision 2, dated August 2003.

2. DCD, Section 7.7.1.11.