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# COVER SHEET FOR CORRESPONDENCE

## USE THIS COVER SHEET TO PROTECT ORIGINALS OF MULTI-PAGE CORRESPONDENCE

OPERATING LICENSE NOS. DPR-38, 47, 55

BASES TO TECHNICAL SPECIFICATIONS

FOR THE

OCONEE NUCLEAR STATION UNITS 1, 2, 3

DUKE ENERGY CORPORATION

DOCKET NOS. 50-269, 270, 287

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# B 2.0 SAFETY LIMITS (SLs)

## B 2.1.1 Reactor Core SLs

BASES

BACKGROUND

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

DNB is not a directly measurable parameter during operation, but neutron power and Reactor Coolant System (RCS) temperature, flow and pressure can be related to DNB using a critical heat flux (CHF) correlation. The BWC (Ref. 2) CHF correlation has been developed to predict DNB for axially uniform and non-uniform heat flux distributions. The BWC correlation applies to Mark-BZ fuel. The local DNB heat flux ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the actual local heat flux, is indicative of the margin to DNB. The minimum value of the DNBR, during steady-state operation, normal operational transients, and anticipated transients is limited to 1.18 (BWC).

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

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

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of DNB and the resultant sharp reduction in heat transfer coefficient. Inside the steam film,

BASES

BACKGROUND (continued)

high cladding temperatures are reached, and a cladding-water (zirconium-water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

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

APPLICABLE SAFETY ANALYSES

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

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

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

The RPS setpoints (Ref. 3), in combination with all the LCOs, are designed to prevent any analyzed combination of transient conditions for RCS temperature, flow and pressure, and THERMAL POWER level that would result in a DNB ratio (DNBR) of less than the DNBR limit and preclude the existence of flow instabilities.

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

a. RCS High Pressure trip;

b. RCS Low Pressure trip;

c. Nuclear Overpower trip;

d. RCS Variable Low Pressure trip;

e. Reactor Coolant Pump to Power trip;

f. Flux/Flow Imbalance trip;

BASES

APPLICABLE SAFETY ANALYSES (continued)

g. High Core Outlet Temperature trip; and

h. MSRVs.

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

SAFETY LIMITS

SL 2.1.1.1 and SL 2.1.1.2 ensure that fuel centerline temperature stays below the melting point and that the minimum DNBR is not less than the safety analyses limit.

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

Operation within these limits is ensured by compliance with the AXIAL POWER IMBALANCE and Variable Low RCS Pressure protective limits preserved by their corresponding RPS setpoints in LCO 3.3.1, "Reactor Protection System (RPS) Instrumentation," as specified in the COLR. The AXIAL POWER IMBALANCE protective limits are separate and distinct from the AXIAL POWER IMBALANCE operating limits defined by LCO 3.2.2, "AXIAL POWER IMBALANCE Operating Limits." The AXIAL POWER IMBALANCE operating limits in LCO 3.2.2, also specified in the COLR, preserve initial conditions of the safety analyses but are not reactor core SLs.

APPLICABILITY

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

In MODES 3, 4, 5, and 6, Applicability is not required, since the reactor is not generating significant THERMAL POWER.

BASES (continued)

SAFETY LIMIT VIOLATIONS

The following SL violation responses are applicable to the reactor core SLs.

2.2.1

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

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

REFERENCES

1. UFSAR, Section 3.1.

2. BAW-10143P-A, "BWC Correlation of Critical Heat Flux," April 1995.

3. UFSAR, Chapter 15.

# B 2.0 SAFETY LIMITS (SLs)

## B 2.1.2 Reactor Coolant System (RCS) Pressure SL

BASES

**BACKGROUND**

According to ONS Design Criteria (Ref. 1), the reactor coolant pressure boundary (RCPB) design conditions are not to be exceeded during normal operation nor during anticipated transients. ONS Design Criteria (Ref. 1), specifies that reactivity accidents including rod ejection do not result in damage to the RCPB greater than limited local yielding.

The design pressure of the RCS is 2500 psig. During normal operation and anticipated transients, the RCS pressure is kept from exceeding the design pressure by more than 10% in order to remain in accordance with Section III of the ASME Code (Ref. 2). Hence, the safety limit is 2750 psig. To ensure system integrity, all RCS components are hydrostatically tested at 125% of design pressure prior to initial operation, according to the ASME Code requirements. Following inception of unit operation, RCS components shall be pressure tested, in accordance with the requirements of ASME Code, Section XI (Ref. 3).

**APPLICABLE SAFETY ANALYSES**

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

The RCS pressurizer safety valves are sized to prevent system pressure from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME Code for Nuclear Power Plant Components (Ref. 2).

The limiting peak pressure transient, as determined by the safety analyses (Ref. 5), is performed using conservative assumptions relative to pressure control devices.

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

a. Pressurizer power operated relief valve (PORV);

b. Steam line turbine bypass valves;

BASES

APPLICABLE SAFETY ANALYSES (continued)

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

d. Pressurizer spray valve.

SAFETY LIMITS

The maximum transient pressure allowed in the RCS pressure vessel under the ASME Code, Section III (Ref. 2), is 110% of design pressure. The maximum transient pressure allowed in the RCS piping, valves, and fittings under USAS, Section B31.7 (Ref. 4), is 110% of design pressure. Therefore, the SL on maximum allowable RCS pressure is 2750 psig.

Overpressurization of the RCS can result in a breach of the RCPB. If such a breach occurs in conjunction with a fuel cladding failure, fission products could enter the containment atmosphere and steam generators, raising concerns relative to limits on radioactive releases specified in 10 CFR 100, "Reactor Site Criteria" (Ref. 6).

APPLICABILITY

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

SAFETY LIMIT VIOLATIONS

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

2.2.2

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

Exceeding the RCS pressure SL may cause immediate RCS failure and create a potential for radioactive releases in excess of 10 CFR 100, "Reactor Site Criteria," limits (Ref. 6).

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

BASES

SAFETY LIMIT VIOLATIONS (continued)

2.2.3

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

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

REFERENCES

1. UFSAR, Section 3.1.
2. ASME Boiler and Pressure Vessel Code, Section III, Article NB-7000.
3. ASME Boiler and Pressure Vessel Code, Section XI, Article IW-5000.
4. ASME USAS B31.7, Nuclear Power Piping, dated February 1968 with June 1968 Errata.
5. UFSAR, Chapters 5 and 15.
6. 10 CFR 100.

# B 3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

## BASES

| | |
|---|---|
| LCOs | LCO 3.0.1 through LCO 3.0.7 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated. |
| LCO 3.0.1 | LCO 3.0.1 establishes the Applicability statement within each individual Specification as the requirement for when the LCO is required to be met (i.e., when the unit is in the MODES or other specified conditions of the Applicability statement of each Specification). |
| LCO 3.0.2 | LCO 3.0.2 establishes that upon discovery of a failure to meet an LCO, the associated ACTIONS shall be met. The Completion Time of each Required Action for an ACTIONS Condition is applicable from the point in time that an ACTIONS Condition is entered. The Required Actions establish those remedial measures that must be taken within specified Completion Times when the requirements of an LCO are not met. This Specification establishes that:<br><br>a. Completion of the Required Actions within the specified Completion Times constitutes compliance with a Specification; and<br><br>b. Completion of the Required Actions is not required when an LCO is met within the specified Completion Time, unless otherwise specified.<br><br>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 Specification.

The nature of some Required Actions of some Conditions necessitates that, once the Condition is entered, the Required Actions must be completed even though the associated Conditions no longer exist. The individual LCO's ACTIONS specify the Required Actions where this is the case. An example of this is in LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits."

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

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

LCO 3.0.3

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

a. An associated Required Action and Completion Time is not met and no other Condition applies; or

b. The condition of the unit is not specifically addressed by the associated ACTIONS. This means that no combination of Conditions stated in the ACTIONS can be made that exactly

BASES

LCO 3.0.3 (continued)

corresponds to the actual condition of the unit. Sometimes, possible combinations of Conditions are such that entering LCO 3.0.3 is warranted; in such cases, the ACTIONS specifically state a Condition corresponding to such combinations and also that LCO 3.0.3 be entered immediately.

This Specification delineates the time limits for placing the unit in a safe MODE or other specified condition when operation cannot be maintained within the limits for safe operation as defined by the LCO and its ACTIONS. It is not intended to be used as an operational convenience that permits routine voluntary removal of redundant systems or components from service in lieu of other alternatives that would not result in redundant systems or components being inoperable.

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

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

a. The LCO is now met.

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

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

The time limits of LCO 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

BASES

LCO 3.0.3 (continued)

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 16 hours, because the total time for reaching MODE 4 is not reduced from the allowable limit of 18 hours. Therefore, if remedial measures are completed that would permit a return to MODE 1, a penalty is not incurred by having to reach a lower MODE of operation in less than the total time allowed.

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

Exceptions to LCO 3.0.3 are provided in instances where requiring a unit shutdown, in accordance with LCO 3.0.3, would not provide appropriate remedial measures for the associated condition of the unit. An example of this is in LCO 3.7.14, "Spent Fuel Pool Water Level." LCO 3.7.14 has an Applicability of "During movement of irradiated fuel assemblies in the spent fuel pool." Therefore, this LCO can be applicable in any or all MODES. If the LCO and the Required Actions of LCO 3.7.14 are not met while in MODE 1, 2, 3, or 4, there is no safety benefit to be gained by placing the unit in a shutdown condition. The Required Action of LCO 3.7.14 of "Suspend movement of irradiated fuel assemblies in 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 in that Applicability (e.g., Applicability desired to be entered) when the following exist:

a. Unit conditions are such that the requirements of the LCO would not be met in the Applicability desired to be entered; and

b. Continued noncompliance with the LCO requirements, if the Applicability were entered, would result in the unit being required to exit the Applicability desired to be entered to comply with the Required Actions.

BASES

LCO 3.0.4 (continued)

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 result from any unit shutdown.

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

LCO 3.0.4 is only applicable when entering MODE 4 from MODE 5, MODE 3 from MODE 4, MODE 2 from MODE 3, or MODE 1 from MODE 2. Furthermore, LCO 3.0.4 is applicable when entering any other specified condition in the Applicability associated with operating in MODES 1, 2, 3, or 4. The requirements of LCO 3.0.4 do not apply in MODES 5 and 6, or in other specified conditions of the Applicability (unless in MODES 1, 2, 3, or 4) because the ACTIONS 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 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.

BASES (continued)

LCO 3.0.5

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

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

b. The OPERABILITY of other equipment.

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

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

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

LCO 3.0.6

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

BASES

LCO 3.0.6 (continued)

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

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

Specification 5.5.16, "Safety Function Determination Program (SFDP)," ensures loss of safety function is detected and appropriate actions are taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other limitations, remedial actions, or compensatory actions may be identified as a result of the support system inoperability and corresponding exception to entering supported system Conditions and Required Actions. The SFDP implements the requirements of LCO 3.0.6.

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

a. A required system redundant to system(s) supported by the inoperable support system is also inoperable; or (EXAMPLE B3.06-1)

b. A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable; or (EXAMPLE B3.06-2)

c. A required system redundant to support system(s)for the supported systems (a) and (b) above is also inoperable. (EXAMPLE B3.06-3)

BASES

LCO 3.0.6
(continued)

EXAMPLE B3.06-1

If System 2 of Train A is inoperable, and System 5 of Train B is inoperable, a loss of safety function exists in supported System 5.

EXAMPLE B3.06-2

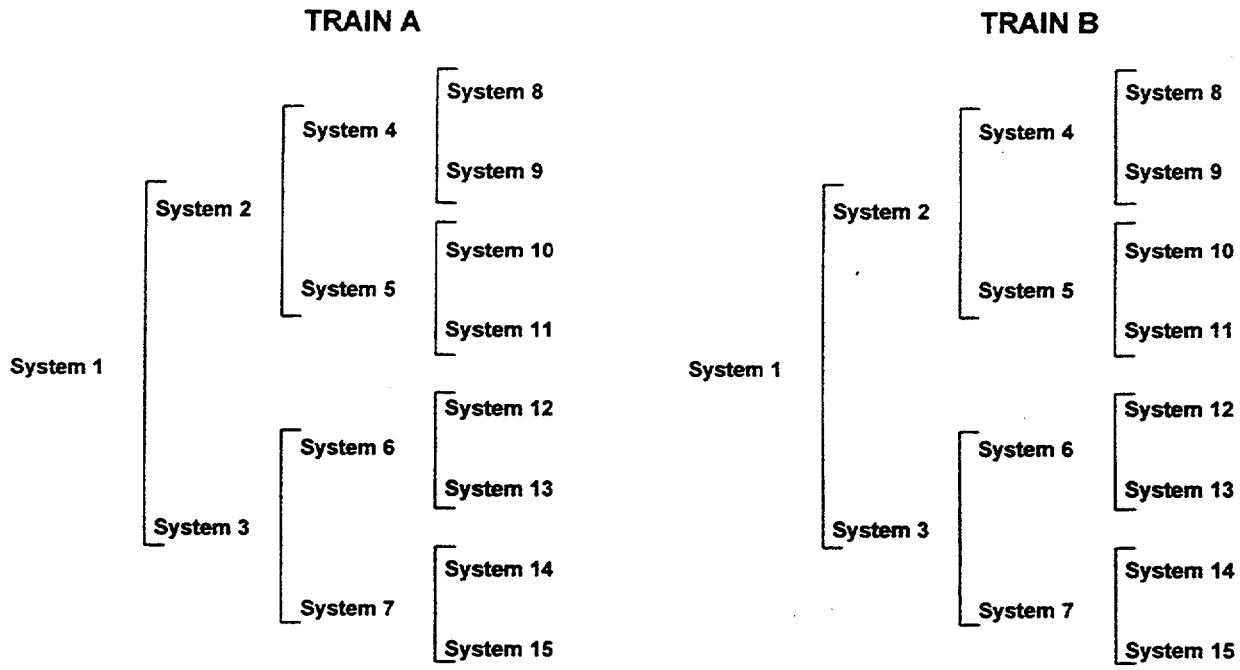
If System 2 of Train A is inoperable, and System 11 of Train B is inoperable, a loss of safety function exists in System 11 which is in turn supported by System 5.

EXAMPLE B3.06-3

If System 2 of Train A is inoperable, and System 1 of Train B is inoperable, a loss of safety function exists in Systems 2, 4, 5, 8, 9, 10 and 11.

If this evaluation determines that a loss of safety function exists, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

**EXAMPLES**



BASES (continued)

LCO 3.0.7

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

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

## B 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

BASES

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

SR 3.0.1

SR 3.0.1 establishes the requirement that SRs must be met during the MODES or other specified conditions in the Applicability for which the requirements of the LCO apply, unless otherwise specified in the individual SRs. This Specification is to ensure that Surveillances are performed to verify the OPERABILITY of systems and components, and that variables are within specified limits. Failure to meet a Surveillance within the specified Frequency, in accordance with SR 3.0.2, constitutes a failure to meet an LCO.

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

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

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

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

BASES

SR 3.0.1 (continued)

define the remedial measures that apply. Surveillances have to be met and performed in accordance with SR 3.0.2, prior to returning equipment to OPERABLE status.

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

Some example of this process are:

a. Emergency feedwater (EFW) pump turbine maintenance during refueling that requires testing at steam pressures > 300 psi. However, if other appropriate testing is satisfactorily completed, the EFW System can be considered OPERABLE. This allows startup and other necessary testing to proceed while the plant reaches the steam pressure required to perform the EFW pump testing.

b. High Pressure Injection (HPI) maintenance during shutdown that requires system functional tests at a specified pressure. Provided other appropriate testing is satisfactorily completed, startup can proceed with HPI considered OPERABLE. This allows operation to reach the specified pressure to complete the necessary post maintenance testing.

SR 3.0.2

SR 3.0.2 establishes the requirements for meeting the specified Frequency for Surveillances and any Required Action with a Completion Time that requires the periodic performance of the Required Action on a "once per..." interval.

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

BASES

SR 3.0.2 (continued)

The 25% extension does not significantly degrade the reliability that results from performing the Surveillance at its specified Frequency. This is based on the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the SRs. The exceptions to SR 3.0.2 are those Surveillances for which the 25% extension of the interval specified in the Frequency does not apply. These exceptions are stated in the individual Specifications. An example of where SR 3.0.2 does not apply is a Surveillance with a Frequency of "in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions." The requirements of regulations take precedence over the TS. The TS cannot in and of themselves extend a test interval specified in the regulations.

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

As stated in SR 3.0.2, the 25% extension also does not apply to the initial portion of a periodic Completion Time that requires performance on a "once per..." basis. The 25% extension applies to each performance after the initial performance. The initial performance of the Required Action, whether it is a particular Surveillance or some other remedial action, is considered a single action with a single Completion Time. One reason for not allowing the 25% extension to this Completion Time is that such an action usually verifies that no loss of function has occurred by checking the status of redundant or diverse components or accomplishes the function of the inoperable equipment in an alternative manner.

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

SR 3.0.3

SR 3.0.3 establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a Surveillance has not been completed within the specified Frequency. A delay period of up to 24 hours or up to the limit of the specified Frequency, whichever is less, applies from the point in time that it is discovered that the Surveillance has not been performed in accordance with SR 3.0.2, and not at the time that the specified Frequency was not met.

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

BASES

SR 3.0.3 (continued)

The basis for this delay period includes consideration of unit conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, the safety significance of the delay in completing the required Surveillance, and the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the requirements.

When a Surveillance with a Frequency based not on time intervals, but upon specified unit conditions or operational situations, is discovered not to have been performed when specified, SR 3.0.3 allows the full delay period of 24 hours to perform the Surveillance. SR 3.0.3 also provides a time limit for completion of Surveillances that become applicable as a consequence of MODE changes imposed by Required Actions.

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

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

Satisfactory completion of the Surveillance within the delay period allowed by this Specification, or within the Completion Time of the ACTIONS, restores compliance with SR 3.0.1.

SR 3.0.4

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

This Specification ensures that system and component OPERABILITY requirements and variable limits are met before entry into MODES or other specified conditions in the Applicability for which these systems and components ensure safe operation of the unit. The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to OPERABLE status before entering an associated MODE or other specified condition in the Applicability.

BASES

SR 3.0.4 (continued)

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

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

The precise requirements for performance of SRs are specified such that exceptions to SR 3.0.4 are not necessary. The specific time frames and conditions necessary for meeting the SRs are specified in the Frequency, in the Surveillance, or both. This allows performance of Surveillances when the prerequisite condition(s) specified in a Surveillance procedure require entry into the MODE or other specified condition in the Applicability of the associated LCO prior to the performance or completion of a Surveillance. A Surveillance that could not be performed until after entering the LCO Applicability would have its Frequency specified such that it is not "due" until the specific conditions needed are met. Alternately, the Surveillance may be stated in the form of a Note, as not required (to be met or performed) until a particular event, condition, or time has been reached. Further discussion of the specific formats of SRs' annotation is found in Section 1.4, Frequency.

SR 3.0.4 is only applicable when entering MODE 4 from MODE 5, MODE 3 from MODE 4, MODE 2 from MODE 3, or MODE 1 from MODE 2. Furthermore, SR 3.0.4 is applicable when entering any other specified condition in the Applicability associated with operation in MODES 1, 2, 3, or 4. The requirements of SR 3.0.4 do not apply in MODES 5 and 6, or in other specified conditions of the Applicability (unless in MODES 1, 2, 3, or 4) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken.

# B 3.1 REACTIVITY CONTROL SYSTEMS

## B 3.1.1 SHUTDOWN MARGIN (SDM)

BASES

BACKGROUND

According to the ONS Design Criteria (Ref. 1), the reactivity control systems must be redundant and capable of holding the reactor core subcritical when shut down under cold conditions. SDM requirements provide sufficient reactivity margin to ensure that acceptable fuel design limits will not be exceeded for normal shutdown and anticipated transients. In MODES 3, 4, and 5, the SDM defines the degree of subcriticality that would be obtained immediately following the insertion of all CONTROL RODS, assuming the single CONTROL ROD of highest reactivity worth is fully withdrawn. The Axial Power Shaping Rods (APSRs) do not trip and are not credited in the SDM.

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 CONTROL RODS and soluble boric acid in the Reactor Coolant System (RCS). The CONTROL RODS can compensate for the reactivity effects of the fuel and RCS temperature changes accompanying power level changes over the range from full load to no load. In addition, the CONTROL RODS provide SDM during low power operation and during accidents and anticipated transients and are capable of making the core subcritical rapidly enough to prevent exceeding acceptable fuel damage limits, assuming that the CONTROL ROD of highest reactivity worth remains fully withdrawn.

The control of boron concentration in the RCS can compensate for fuel depletion, during operation and all xenon burnout reactivity changes, and maintain the reactor subcritical under cold conditions.

While in MODES 1 and 2, SDM control is ensured by operating with the safety rods fully withdrawn (LCO 3.1.5, "Safety Rod Position Limits") and the regulating rods within the limits of LCO 3.2.1, "Regulating Rod Position Limits." When the unit is in MODE 3, 4, 5 or 6, the SDM requirements are met by means of adjustments to the RCS boron concentration. Adjusted SDM limits preclude recriticality in the event of a main steam line break (MSLB) in MODE 3, 4, or 5 when high steam generator levels exist.

BASES (continued)

APPLICABLE SAFETY ANALYSES

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

The criteria for SDM requirements are that specified acceptable fuel design limits are maintained. The SDM requirements must ensure that:

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

b. The reactivity transients associated with postulated accident conditions are controllable with acceptable limits (departure from nucleate boiling ratio (DNBR), fuel centerline temperature limits for anticipated transients, and ≤ 280 cal/gm energy deposition for the rod ejection accident (Ref. 3)); and

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

The most limiting accident for the SDM requirements is based on an MSLB, as described in the accident analyses (Ref. 2). In addition to the limiting MSLB accident, the SDM requirement must also protect against other accidents described in UFSAR Chapter 15 (Ref. 2).

The basis for the shutdown requirement when high steam generator levels exist is the heat removal potential in the secondary system fluid and the positive reactivity added via MTC. At any given initial primary system temperature and its associated secondary system pressure, the secondary system liquid levels can be equated to a final primary system temperature assuming the entire secondary system mass is boiled away or reaches the thermal equilibrium with the primary system. A SDM at 60°F with the highest worth rod stuck out will bound all resulting SDM's in the event that the entire secondary system is boiled away. However, a 200°F SDM is adequately conservative since the RCS will not cool down below this temperature in the event of a MSLB with raised SG levels.

SDM satisfies Criterion 2 of 10 CFR 50.36 (Ref. 4).

LCO

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

BASES

LCO (continued)

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

The MSLB (Ref. 2) accident is the most limiting analysis that establishes the SDM value of the LCO.

For MSLB accidents, if the LCO is violated, there is a potential to exceed the DNBR limit and to exceed 10 CFR 100 limits (Ref. 5).

To compensate for the potential heat removal associated with an MSLB accident when high steam generator levels exist, such as during secondary system chemistry control and steam generator cleaning, the initial SDM in the core must be adjusted. The operating procedures provide adjusted SDM limits that ensure the core will remain subcritical following an MSLB accident from those conditions.

APPLICABILITY

In MODES 3, 4, and 5, the SDM requirements are applicable to provide sufficient negative reactivity to meet the assumptions of the safety analysis discussed above. The operating procedures are used to define the SDM when high steam generator levels exist, such as during secondary system chemistry control and steam generator cleaning in MODES 3, 4, and 5. In MODES 1 and 2, SDM is ensured by complying with LCO 3.1.5 and LCO 3.2.1. In MODE 6, the shutdown reactivity requirements are given in LCO 3.9.1, "Boron Concentration."

ACTIONS

A.1

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

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

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1

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

a. RCS boron concentration;

b. CONTROL ROD position;

c. RCS average temperature;

d. Fuel burnup;

e. Xenon concentration;

f. Samarium concentration; and

g. Overall temperature coefficient.

Operators verify that the existing boron concentration in the RCS meets SDM requirements using a SDM curve.

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

REFERENCES

1. UFSAR, Section 3.1.

2. UFSAR, Chapter 15.

3. UFSAR, Section 15.12.

4. 10 CFR 50.36.

5. 10 CFR 100, "Reactor Site Criteria."

# B 3.1 REACTIVITY CONTROL SYSTEMS

## B 3.1.2 Reactivity Balance

BASES

BACKGROUND

According to ONS Design Criteria (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 transients. Therefore, the reactivity balance is used as a measure of the agreement between the predicted core reactivity and the actual core reactivity during power operation. The periodic confirmation of the predicted core reactivity is necessary to ensure that safety analyses of design basis transients and accidents remain valid. A large reactivity difference could be the result of unanticipated changes in fuel, CONTROL ROD, burnable poison worth, or operation at conditions not consistent with those assumed in the predictions of core reactivity. These could potentially result in a loss of SDM or violation of acceptable fuel design limits. Comparing predicted core reactivity with the actual core reactivity validates the nuclear methods used in the safety analysis and supports the SDM demonstrations in ensuring the reactor can be brought safely to cold, subcritical conditions. The difference between the actual and predicted core reactivity is commonly referred to as a reactivity anomaly.

When the reactor is critical in MODE 1 or 2, a reactivity balance exists where the net reactivity is zero (referred to as the actual core reactivity state). A comparison of predicted core reactivity and the actual core reactivity is convenient under such a balance, since parameters are being maintained relatively stable under steady state power conditions and the net reactivity is known to be zero. 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 soluble boron and burnable absorbers, producing zero net reactivity.

In order to achieve the required fuel cycle energy output, the uranium enrichment in the new fuel loading and the fuel 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 and above the point of adding heat, the excess positive reactivity of the fuel is compensated by burnable absorbers, CONTROL RODS, APSRs, thermal feedback from the fuel and moderator, fission product poisons (mainly xenon and samarium), and the reactor coolant system (RCS) boron concentration.

BASES

BACKGROUND (continued)

During cycle operation, the fuel is being depleted and excess reactivity is decreasing. As the fuel depletes, the primary method of compensating for the reduction in excess reactivity is through a reduction in the RCS boron concentration.

APPLICABLE SAFETY ANALYSES

The criteria for core reactivity are the establishment of the reactivity balance limits to ensure that unit operation is maintained within the assumptions of the safety analyses.

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

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

The comparison between the actual reactivity condition of the critical reactor and the predicted initial core reactivity provides an opportunity for the normalization of the calculational models used to predict core reactivity. If the predicted core reactivity and the actual core reactivity at reference core conditions at beginning of cycle (BOC) do not agree, then the assumptions used in the reload cycle design analysis or the calculational models used to predict reactivity requirements may not be accurate. If reasonable agreement between the actual and predicted core reactivity exists at BOC, then the prediction may be normalized to the measured boron concentration. Thereafter, any significant deviations in the predicted reactivity condition from the actual reactivity condition during the operating cycle may be an indication that the calculational model is not adequate for the operating cycle or that an unexpected change in core conditions has occurred.

The normalization of predicted reactivity parameters to the actual reactivity value is typically performed after reaching RTP following startup from a refueling outage, with the RCS temperature, CONTROL RODS,

BASES

APPLICABLE SAFETY ANALYSES (continued)

and APSRs in their reference positions and fission product poisons at their expected equilibrium concentrations. The normalization is performed at BOC conditions, so that core reactivity relative to predicted values can be continually monitored and evaluated, as core conditions change during the cycle.

Reactivity balance satisfies Criterion 2 of 10 CFR 50.36 (Ref. 3).

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 accident analyses are no longer valid, or that the uncertainties in the nuclear design methodology are larger than expected. A limit on the reactivity of ± 1% $\Delta k/k$ has been established, based on engineering judgment. A ± 1% $\Delta k/k$ deviation in the predicted reactivity from the actual reactivity condition of the reactor is larger than expected for normal operation and should therefore be evaluated.

When the predicted core reactivity is within 1% $\Delta k/k$ of the actual reactivity value at steady state thermal conditions, the core is considered to be operating within acceptable design limits.

APPLICABILITY

In MODES 1 and 2, the limits on the core reactivity must be maintained to ensure an acceptable SDM and continued adherence to the assumptions used in the accident analyses. 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 2 because enough operating margin exists to limit the effects of a reactivity anomaly, and THERMAL POWER is low enough (≤ 5% RTP) such that reactivity anomalies are unlikely to occur.

This Specification does not apply in MODES 3, 4, and 5, because the reactor is shutdown and the net reactivity condition of the reactor cannot be determined.

In MODE 6, boron concentration requirements (LCO 3.9.1, "Refueling Boron Concentration") ensure that fuel movements are performed within acceptable bounds.

BASES (continued)

ACTIONS

A.1 and A.2

Should an anomaly outside the tolerance band develop between the actual core reactivity and the predicted core reactivity, an evaluation of the core design and safety analysis must be performed. Core conditions are evaluated to determine their consistency with the input assumptions used in the core 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 an abnormality or accident occurring during this period, and allows sufficient time to assess the physical condition of the core 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 reference conditions at the time of the reactivity balance, then a recalculation of the reactivity balance 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 appropriate reactivity parameter may be renormalized, and operation in MODES 1 or 2 may continue. If operational restrictions or additional surveillance requirements are necessary to ensure the reactor core is acceptable for continued operation, then they must be defined.

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

B.1

If the core reactivity balance cannot be restored to within the 1% $\Delta k/k$ limit, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 2 within 12 hours. The allowed Completion Time of 12 hours is reasonable, based on operating experience to reach the required unit conditions from RTP conditions in an orderly manner and without challenging unit systems.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1

Core reactivity is verified by periodic reactivity balance calculations that compares the predicted core reactivity to the actual core reactivity condition (net reactivity of zero condition). The comparison is made considering that other core conditions are fixed or stable, including CONTROL ROD and APSR positions, moderator temperature, fuel temperature, fuel depletion, xenon concentration, and samarium concentration. The Surveillance is performed once prior to entering MODE 1, after each fuel loading as an initial check on core reactivity conditions and design calculations at BOC. A Note is included in the SR to indicate that the normalization of predicted core reactivity to the measured value may take place within the first 60 effective full power days (EFPD) after each fuel loading. The required Frequency of 31 EFPD, following the initial 60 EFPD after entering MODE 1 is acceptable, based on the slow rate of core reactivity changes due to fuel depletion and the presence of other indicators (QPT, etc.) for prompt indication of an anomaly. The 60 EFPD after entering MODE 1 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.

REFERENCES

1. UFSAR, Section 3.1.
2. UFSAR, Chapter 15.
3. 10 CFR 50.36.

## B 3.1 REACTIVITY CONTROL SYSTEMS

### B 3.1.3 Moderator Temperature Coefficient (MTC)

BASES

BACKGROUND

According to ONS Design Criteria (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). Therefore, a coolant temperature increase will cause a reactivity decrease with a negative MTC. Reactivity increases that cause a coolant temperature increase will thus be self limiting. The same characteristic is true when the MTC is positive and coolant temperature decreases occur.

MTC values are predicted at selected burnups during the safety evaluation analysis and are confirmed to be acceptable by measurements. Reactor cores are designed so that the beginning of cycle (BOC) MTC is less than or equal to zero when THERMAL POWER is 95% RTP or greater. The actual value of the MTC is dependent on core characteristics, such as fuel loading and reactor coolant soluble boron concentration. The core design may require additional burnable absorbers to yield an MTC at BOC within the range analyzed in the accident analysis. The end of cycle (EOC) MTC is also limited by the requirements of the accident analysis. Fuel cycles that are designed to achieve high burnups or that have changes to other characteristics are evaluated to ensure the MTC does not become more negative than the value assumed in the safety analyses.

APPLICABLE SAFETY ANALYSES

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

BASES

APPLICABLE SAFETY ANALYSES (continued)

The criteria for the specified MTC are:

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

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

Accidents that cause core overheating (either decreased heat removal or increased power production) must be evaluated for results when the MTC is positive. Reactivity accidents that cause increased power production include the uncontrolled CONTROL ROD group withdrawal transient from either zero or full THERMAL POWER. The limiting overheating event relative to unit response is based on the maximum difference between core power and steam generator heat removal during a transient.

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

MTC values are bounded in reload safety evaluations, assuming steady state conditions at BOC and EOC.

MTC satisfies Criterion 2 of 10 CFR 50.36 (Ref. 3).

LCO

LCO 3.1.3 requires the MTC to be within specified limits in the COLR to ensure the core operates within the assumptions of the accident analysis. During the reload core safety evaluation, the MTC is analyzed to determine that its values remain within the bounds of the accident analysis during operation. The LCO establishes a maximum positive value that can not be exceeded. The limit in the COLR on positive MTC ensures that core overheating accidents will not violate the accident analysis assumptions. The limit on MTC when THERMAL POWER is ≥ 95% RTP, ensures that steady state core operation will be stable.

BASES

LCO (continued)

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

APPLICABILITY

In MODE 1, the limits on MTC must be maintained to ensure that any accident initiated from THERMAL POWER operation will not violate the design assumptions of the accident analysis. In MODE 2, the limits must also be maintained to ensure that startup and subcritical accidents, such as the uncontrolled CONTROL ROD group withdrawal, will not violate the assumptions of the accident analysis. In MODES 3, 4, 5, and 6, this LCO is not applicable, since no accidents using the MTC as an analysis assumption are initiated from these MODES. However, the variation of MTC with temperature in MODES 3, 4, and 5 for accidents initiated in MODES 1 and 2 is accounted for in the subject accident analysis. The Surveillance check at BOC on MTC provides confirmation that MTC is behaving as anticipated.

ACTIONS

A.1

MTC is a core physics parameter determined by the fuel and fuel cycle designs, and cannot be controlled directly once the designs have been implemented in the core. If MTC exceeds its limits, the reactor must be placed in MODE 3. This eliminates the potential for violation of the accident analysis assumptions. The associated Completion Time of 12 hours is reasonable, considering the probability of an accident occurring during the time period that would require an MTC value within the LCO limits, for reaching MODE 3 conditions from RTP conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.3.1

The SR for measurement of the MTC at the beginning of each fuel cycle provides for confirmation of the limiting MTC values. The MTC changes slowly in the negative direction from most positive value during fuel cycle operation, as the RCS boron concentration is reduced with fuel depletion.

The requirement for measurement, prior to initial operation in MODE 1, satisfies the confirmatory check on the most positive (least negative) MTC value.

BASES (continued)

REFERENCES

1. UFSAR, Section 3.1.
2. UFSAR, Chapter 15.
3. 10 CFR 50.36.

## B 3.1 REACTIVITY CONTROL SYSTEMS

## B 3.1.4 CONTROL ROD Group Alignment Limits

BASES

BACKGROUND

The OPERABILITY (e.g., trippability) of the CONTROL RODS is an initial condition assumption in all safety analyses that assume rod insertion upon reactor trip. Maximum rod misalignment is an initial condition assumption in the safety analysis that directly affects core power distributions and assumptions of SDM. An inoperable CONTROL ROD that is unable to respond to positioning signals from the Rod Drive Control System may still meet its SDM capabilities if it is capable of responding to a valid trip signal (i.e., inoperable but trippable). It would, however, have the potential to adversely affect core power distribution due to its inability to maintain itself within the group average. An inoperable CONTROL ROD which is not "trippable" would satisfy neither the capacity to supply SDM requirements nor the ability to maintain itself in alignment with the group to assure acceptable core power distribution.

The applicable criteria for these design requirements are ONS Design Criteria (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 CONTROL ROD worth for reactor shutdown. Therefore, CONTROL ROD alignment and OPERABILITY are related to core operation within design power peaking limits and the core design requirement of a minimum SDM.

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

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

BASES

BACKGROUND (continued)

The CONTROL RODS are arranged into rod groups that are radially symmetric. Therefore, movement of the CONTROL RODS does not introduce radial asymmetries in the core power distribution. The CONTROL RODS provide required negative reactivity worth for immediate reactor shutdown upon a reactor trip. The regulating rods provide reactivity control during normal operation and transients, and their movement is normally governed by the Integrated Control System.

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

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

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

APPLICABLE SAFETY ANALYSES

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

a. There shall be no violations of:

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

b. The core must remain subcritical after accident transients, except for a main steam line break (MSLB). The analysis results for a MSLB with a coincident failure of the most reactive rod to insert results in a return to criticality.

BASES

APPLICABLE SAFETY ANALYSES (continued)

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 when one CONTROL ROD drops partially or fully into the reactor core. With ICS in manual, this event causes an initial power reduction followed by a return towards the original power due to positive reactivity feedback from the negative moderator temperature coefficient. Increased peaking during the power increase may result in excessive local linear heat rates (LHRs).

The accident analysis and reload safety evaluations define regulating rod position limits that ensure the required SDM can always be achieved if the maximum worth CONTROL ROD is stuck fully withdrawn (Ref. 3). If a CONTROL ROD is stuck in or dropped in, continued operation is permitted. The Required Action statements in the LCOs provide conservative reductions in THERMAL POWER and verification of SDM to ensure continued operation remains within the bounds of the safety analysis (Ref. 3).

The CONTROL ROD group alignment limits satisfy Criterion 2 of 10 CFR 50.36 (Ref. 4).

LCO

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

The limit for individual CONTROL ROD misalignment is 6.5% (9 inches) deviation from the group average position. This value is established, based on the distance between reed switches, with additional allowances for uncertainty in the equipment used to determine this value. For the purpose of complying with this LCO, the position of a misaligned rod is not included in the calculation of the rod group average position.

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

BASES (continued)

APPLICABILITY

The requirements on CONTROL ROD OPERABILITY and alignment are applicable in MODES 1 and 2 because these are the only MODES in which significant 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 unit. In MODES 3, 4, 5, and 6, the alignment limits do not apply because the reactor is shut down and resultant local power peaking would not exceed fuel design limits. In MODES 3, 4, 5 and 6, the OPERABILITY of the 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 MODE 6.

ACTIONS

A.1

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

A.2.1.1

Compliance with Required Actions of Condition A allows for continued power operation with one CONTROL ROD declared inoperable due to inoperable position indication but trippable, or misaligned from its group average position. These Required Actions comprise the final alternate for Condition A.

If realignment of the CONTROL ROD to the group average or alignment of the group to the misaligned CONTROL ROD is not completed within 1 hour (Required Action A.1 not met), the rod shall be considered inoperable. Since the rod may be inserted farther than the group average position for a

BASES

ACTIONS

A.2.1.1 (continued)

long time, SDM must be evaluated. Ensuring the SDM meets the minimum requirement specified in the COLR within 1 hour and once per 12 hours thereafter is adequate to determine that SDM requirements are met.

A.2.1.2

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

A.2.2

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

A.2.3

Reduction of the nuclear overpower trip setpoints, based on flux and flux/flow imbalance, to ≤ 65.5% ALLOWABLE THERMAL POWER, after THERMAL POWER has been reduced to 60% ALLOWABLE THERMAL POWER, maintains both core protection and an operating margin at reduced power similar to that at RTP. The required Completion Time of 10 hours allows the operator 8 additional hours after completion of the THERMAL POWER reduction in Required Action A.2.2.1 to adjust the trip setpoints.

BASES

ACTIONS (continued)

A.2.4

The existing CONTROL ROD configuration must not cause an ejected rod to exceed the limit of 0.2% Δk/k at RPT, 0.4% Δk/k at 80% RPT, or 0.8% Δk/k at zero power. This evaluation may require a computer calculation of the maximum ejected rod worth based on nonstandard configurations of the CONTROL ROD groups. The evaluation must determine the ejected rod worth for the duration of time that operation is expected to continue with a misaligned rod. Should fuel cycle conditions at some later time become more bounding than those at the time of the rod misalignment, additional evaluation will be required to verify the continued acceptability of operation. The required Completion Time of 72 hours is acceptable because LHRs are limited by the THERMAL POWER reduction and sufficient time is provided to perform the required evaluation.

B.1

If the Required Actions and associated Completion Times for Condition A are 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 12 hours. The allowed Completion Time of 12 hours is reasonable, based on operating experience, for reaching MODE 3 from RTP in an orderly manner and without challenging unit systems.

C.1.1

More than one trippable CONTROL ROD becoming inoperable or misaligned, or both inoperable but trippable and misaligned from their group average position, is not expected and may violate the minimum SDM requirement. Therefore, SDM must be evaluated. Ensuring the SDM meets the minimum requirement within 1 hour allows the operator adequate time to determine the SDM.

C.1.2

If the SDM is less than the limit, then the restoration of the required SDM requires increasing the RCS boron concentration to provide negative reactivity. RCS boration must occur as described in Bases Section 3.1.1. The required Completion Time of 1 hour for initiating boration is reasonable, based on the time required for potential xenon redistribution, the low probability of an accident occurring, and the steps required to

BASES

ACTIONS

C.1.2 (continued)

complete the action. This allows the operator sufficient time for aligning the required valves and starting the boric acid pumps. Boration will continue until the required SDM is restored.

C.2

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

Therefore, it is prudent to place the reactor in MODE 3. LCO 3.1.4 does not apply in MODE 3 since excessive power peaking cannot occur. The allowed Completion Time of 12 hours is reasonable, based on operating experience, for reaching MODE 3 from RTP in an orderly manner and without challenging unit systems.

D.1.1 and D.1.2

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

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

D.2

If the untrippable rod(s) cannot be restored to OPERABLE status, the 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 12 hours.

BASES

ACTIONS

D.2 (continued)

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 unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.4.1

Verification that individual CONTROL RODS are aligned within 6.5% of their group average height limits at a 12 hour Frequency allows the operator to detect a rod that is beginning to deviate from its expected position. The specified Frequency takes into account other CONTROL ROD position information that is continuously available to the operator in the control room, so that during actual CONTROL 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 could result in radial tilts. Exercising each individual CONTROL ROD every 92 days provides increased confidence that all rods continue to be OPERABLE without exceeding the alignment limit, even if they are not regularly tripped. Moving each CONTROL ROD by an amount in any direction sufficient to demonstrate the absence of thermal binding will not cause radial or axial power tilts, or oscillations, to occur. The 92 day Frequency takes into consideration other information available to the operator in the control room and SR 3.1.4.1, which is performed more frequently and adds to the determination of OPERABILITY of the rods. Between required performances of SR 3.1.4.2 (determination of CONTROL ROD OPERABILITY by movement), if a CONTROL ROD(S) is discovered to be immovable, but is determined to be trippable and aligned, the CONTROL ROD(S) is considered to be OPERABLE. At any time, if a CONTROL ROD(S) is immovable, a determination of the trippability (OPERABILITY) of the CONTROL ROD(S) must be made, and appropriate action taken.

SR 3.1.4.3

Verification of CONTROL ROD drop time allows the operator to determine that the maximum CONTROL ROD drop time permitted is consistent with the assumed CONTROL ROD drop time used in the safety analysis. The

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.4.3 (continued)

rod drop time given in the safety analysis is 1.66 seconds at reactor coolant full flow conditions and ≤ 1.40 seconds at no flow conditions to ¾ insertion (Ref. 5). The zone reference lights will activate at ¾ insertion to give an indication of the CONTROL ROD drop time and CONTROL ROD location. Measuring CONTROL ROD drop times, prior to reactor criticality after reactor vessel head removal, ensures that the reactor internals and CRDM will not interfere with CONTROL ROD motion or CONTROL ROD drop time. This Surveillance is performed during a unit outage, due to the unit conditions needed to perform the SR and the potential for an unplanned unit transient if the Surveillance were performed with the reactor at power.

This testing is normally performed with all reactor coolant pumps operating to simulate a reactor trip under actual conditions.

REFERENCES

1. UFSAR, Section 3.1.
2. 10 CFR 50.46.
3. UFSAR, Chapter 15.
4. 10 CFR 50.36.
5. UFSAR, Section 15.7.3.

## B 3.1 REACTIVITY CONTROL SYSTEMS

### B 3.1.5 Safety Rod Position Limit

BASES

BACKGROUND

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

The applicable criteria for the reactivity and power distribution design requirements are ONS Design Criteria (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 2).

Limits on safety rod position have been established, and all CONTROL ROD positions are monitored and controlled during operation in MODES 1 and 2 to ensure that the reactivity limits, ejected rod worth, and SDM limit are preserved.

The regulating groups are used for precise reactivity control of the reactor. The positions of the regulating groups are normally automatically controlled by the Integrated Control System, but they can also be manually controlled. They are capable of adding negative reactivity very quickly (compared to borating). In MODES 1 and 2 regulating groups must be maintained above specified position 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 and fuel burnup.

The safety groups can be fully withdrawn without the core going critical. This provides available negative reactivity in the event of boration errors. The safety groups are controlled manually by the control room operator. During entry into MODE 2 from MODE 3, the safety groups must be fully withdrawn. The safety groups must be completely withdrawn from the core prior to any other reduction in SDM by deboration or by withdrawing any regulating groups during an approach to criticality. The safety groups remain in the fully withdrawn position until the reactor is shut down. They add negative reactivity to shut down the reactor and maintain the required SDM upon receipt of a reactor trip signal.

BASES (continued)

APPLICABLE SAFETY ANALYSES

On a reactor trip, all CONTROL RODS, except the most reactive rod, are assumed to insert into the core. The safety groups are at their fully withdrawn limits and available to insert the maximum amount of negative reactivity on a reactor trip signal. The regulating groups may be partially inserted in the core as allowed by LCO 3.2.1, "Regulating Rod Position Limits." The safety group and regulating rod group position 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 RTP. The combination of regulating groups and safety groups (less the most reactive rod, which is assumed to be fully withdrawn) is sufficient to take the reactor from full power conditions at rated temperature to zero power and to maintain the required SDM at rated no load temperature (Ref. 3).

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

a. There shall be no violations of:

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

b. The core must remain subcritical after accident transients, except for a main steam line break (MSLB). The analysis results for a MSLB with a coincident failure of the most reactive rod to insert results in a return to criticality.

The safety rod position limits satisfy Criteria 2 and 3 of 10 CFR 50.36 (Ref. 4).

LCO

The safety groups must be fully withdrawn any time the reactor is in MODE 1 or 2. This ensures that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM following a reactor trip.

This LCO has been modified by a Note indicating the LCO requirement is suspended for those safety rods which are inserted solely due to testing in accordance with SR 3.1.4.2. This SR verifies the freedom of the rods to move, and requires the safety group to move below the LCO limits, which would normally violate the LCO.

BASES (continued)

APPLICABILITY

The safety groups must be within their position limits with the reactor in MODES 1 and 2. This ensures that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM following a reactor trip. Refer to LCO 3.1.1 for SDM requirements in MODES 3, 4, and 5. LCO 3.9.1, "Boron Concentration," ensures adequate SDM in MODE 6.

ACTIONS

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

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

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

Restoration of the required SDM, if necessary, requires increasing the boron concentration, since the safety rod may remain misaligned and not be providing its normal negative reactivity on tripping. The required Completion Time of 1 hour for initiating boration is reasonable, based on the time required for potential xenon redistribution, the low probability of an accident occurring, and the steps required to complete the action. This allows the operator sufficient time for aligning the required valves and starting the boric acid pumps. Boration will continue until the required SDM is restored.

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

B.1.1 and B.1.2

When more than one safety rod is not fully withdrawn, there is a possibility that the required SDM may be adversely affected. Under these conditions, it is important to determine the SDM, and if it is less than the required value, initiate boration until the required SDM is recovered. The Completion Time of 1 hour is adequate for determining SDM and, if necessary, for initiating emergency boration to restore SDM.

BASES

ACTIONS

B.1.1 and B.1.2 (continued)

In this situation, SDM verification must include the worth of the untrippable rod(s) as well as the CONTROL ROD of maximum worth.

B.2

If more than one safety rod is not fully withdrawn the unit must be brought to a MODE where the LCO is not applicable. The allowed Completion Time of 12 hours is reasonable, based on operating experience, for reaching the required MODE from RTP in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.5.1

Verification that each safety rod is fully withdrawn ensures the safety rods are available to provide reactor shutdown capability.

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

REFERENCES

1. UFSAR, Section 3.1.
2. 10 CFR 50.46.
3. UFSAR, Chapter 3.
4. 10 CFR 50.36.

## B 3.1 REACTIVITY CONTROL SYSTEMS

## B 3.1.6 AXIAL POWER SHAPING ROD (APSR) Alignment Limits

BASES

BACKGROUND

The OPERABILITY of the APSRs and APSR alignment are initial condition assumptions in the safety analysis that directly affect core power distributions. The applicable criteria for these power distribution design requirements are ONS Design Criteria (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 2).

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

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

APSRs are moved by their control rod drive mechanisms (CRDMs). Each CRDM moves its rod ¾ inch for one revolution of the leadscrew, but at different rates (jog and run) depending on the signal output from the Rod Control Drive System.

The APSRs are arranged into a group that is radially symmetric. Therefore, movement of the APSRs does not introduce radial asymmetries in the core power distribution. The APSRs, which are used to assist in control of the axial power distribution, are positioned manually and do not trip.

APPLICABLE SAFETY ANALYSES

There are no explicit safety analyses associated with mis-aligned APSRs. However, alignment of the APSRs is required to prevent inducing a QUADRANT POWER TILT. The LCOs governing APSR misalignment are provided because the power distribution analysis supporting LCO 3.2.1, LCO 3.2.2, and LCO 3.2.3 assumes the rods are aligned.

During movement of an APSR group, one rod may stop moving while the other rods in the group continue. This condition may cause excessive power peaking. The reload safety evaluations define APSR alignment

BASES

APPLICABLE SAFETY ANALYSES (continued)

limits that allow APSRs to be positioned anywhere within the operating band and the increase in local LHR is within the design limits. The Required Actions provide a conservative approach to ensure that continued operation remains within the bounds of the safety analysis. No safety analyses take credit for movement of the APSRs.

The APSR alignment limits satisfy Criterion 2 of 10 CFR 50.36 (Ref. 3).

LCO

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

The limit for individual APSR misalignment is 6.5% (9 inches) deviation from the group average position. This value is established based on the distance between reed switches, with additional allowances for uncertainty in the equipment used to determine this value.

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

APPLICABILITY

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

ACTIONS

A.1

The ACTIONS described below are required if one APSR is declared inoperable due to inoperable position indication or is misaligned. The unit is not allowed to operate with more than one inoperable or misaligned APSR. This would require the reactor to be placed in MODE 3, in accordance with LCO 3.0.3.

BASES

ACTIONS

A.1 (continued)

An alternative to realigning a single inoperable or misaligned APSR to the group average position is to align the remainder of the APSR group to the position of the inoperable or misaligned APSR. This restores the alignment requirements. Deviations up to 2 hours will not cause significant xenon redistribution to occur.

The reactor may continue in operation with the APSR inoperable or misaligned if the limits on AXIAL POWER IMBALANCE are surveilled within 2 hours to determine if the AXIAL POWER IMBALANCE is still within limits. Also, since any additional movement of the APSRs may result in additional imbalance, Required Action A.1 also requires the AXIAL POWER IMBALANCE surveillance to be performed again within 2 hours after each APSR movement. The required Completion Time of up to 2 hours will not cause significant xenon redistribution to occur.

B.1

The unit must be brought to a MODE in which the LCO does not apply if the Required Actions and associated Completion Times cannot be met. To achieve this status, the unit must be brought to at least MODE 3 within 12 hours. The Completion Time of 12 hours is reasonable, based on operating experience, for reaching MODE 3 from RTP in an orderly manner and without challenging unit systems. In MODE 3, APSR alignment limits are not required because the reactor is not generating significant THERMAL POWER and excessive local LHRs cannot occur from APSR misalignment.

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1

Verification at a 12 hour Frequency that individual APSR positions are within 6.5% of the group average height limits allows the operator to detect an APSR beginning to deviate from its expected position. In addition, APSR position is continuously available to the operator in the control room so that during actual APSR motion, deviations can immediately be detected.

BASES (continued)

REFERENCES

1. UFSAR, Section 3.1.
2. 10 CFR 50.46.
3. 10 CFR 50.36.

## B 3.1 REACTIVITY CONTROL

## B 3.1.7 Position Indicator Channels

BASES

BACKGROUND

According to ONS Design Criteria (Ref. 1), instrumentation to monitor variables and systems over their operating ranges during normal operation, anticipated operational occurrences, and accident conditions must be OPERABLE. LCO 3.1.7 is required to ensure OPERABILITY of the CONTROL ROD and APSR position indicators, and thereby ensure compliance with the CONTROL ROD alignment and position limits and APSR alignment limits.

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

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

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

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

BASES

BACKGROUND (continued)

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

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

CONTROL ROD and APSR position indicating readout devices located in the control room consist of single rod position meters on a position indication panel. A selector switch permits either relative or absolute position indication to be displayed. Indicator lights are provided on the position indication panel to indicate when each CONTROL ROD or APSR is fully withdrawn, fully inserted, or enabled, and whether a rod position asymmetry alarm condition is present. Alternate indicators show full insertion, full withdrawal, and under control for each CONTROL ROD and APSR group.

BASES (continued)

APPLICABLE SAFETY ANALYSES

CONTROL ROD and APSR position accuracy is essential during power operation. Power peaking, ejected rod worth, or SDM limits may be violated in the event of a Design Basis Accident (Ref. 2) with CONTROL RODS or APSRs operating outside their limits undetected. CONTROL ROD and APSR positions must be known in order to verify the core is operating within the group sequence, overlap, design peaking limits, ejected rod worth, and with minimum SDM (LCO 3.1.5, "Safety Rod Position Limits" and LCO 3.2.1, "Regulating Rod Position Limits"). CONTROL ROD and APSR positions must be known in order to verify the alignment limits are preserved (LCO 3.1.4, "CONTROL ROD Group Alignment Limits" and LCO 3.1.6, "AXIAL POWER SHAPING ROD (APSR) Alignment Limits"). CONTROL ROD and APSR positions are continuously monitored to provide operators with information that ensures the unit is operating within the bounds of the accident analysis assumptions.

The CONTROL ROD and APSR position indicator channels satisfy Criterion 2 of 10 CFR 50.36 (Ref. 3).

LCO

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

This requirement ensures that CONTROL ROD and APSR position indication during MODES 1 and 2 and PHYSICS TESTS is accurate, and that design assumptions are not challenged. OPERABILITY of the position indicator channel ensures that inoperable, misaligned, or mispositioned CONTROL RODS or APSRs can be detected. Therefore, power peaking and SDM can be controlled within acceptable limits.

APPLICABILITY

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

ACTIONS

A.1

If the required position indicator channel is inoperable for one or more rods, the position of the CONTROL ROD or APSR is not known with certainty. Therefore, each affected CONTROL ROD or APSR must be declared inoperable, and the limits of LCO 3.1.4 or LCO 3.1.6 apply. The required

BASES

ACTIONS

A.1 (continued)

Completion Time for declaring the rod(s) inoperable is immediately. Therefore, LCO 3.1.4 or LCO 3.1.6 is entered immediately, and the required Completion Times for the appropriate Required Actions in those LCOs apply without delay.

SURVEILLANCE REQUIREMENTS

SR 3.1.7.1

A CHANNEL CHECK of the required position indication channel ensures that position indication for each CONTROL ROD and APSR remains OPERABLE and accurate. This CHANNEL CHECK will detect gross failures. The required Frequency of 12 hours is adequate for verifying that no degradation in system OPERABILITY has occurred.

REFERENCES

1. UFSAR, Section 3.1.

2. UFSAR, Chapter 15.

3 10 CFR 50.36.

## B 3.1 REACTIVITY CONTROL

## B 3.1.8 PHYSICS TESTS Exceptions – MODE 2

BASES

BACKGROUND

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

The key objectives of a test program are to:

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

To accomplish these objectives, testing is performed prior to initial criticality; during startup, low power operations, and power ascension; at high powers; and after each refueling. The PHYSICS TESTS requirements for reload fuel cycles ensure that the operating characteristics of the core are consistent with the design predictions (Ref. 3), and therefore the core can be operated as designed.

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

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

BASES (continued)

APPLICABLE SAFETY ANALYSES

Reference 4 describes the initial testing of the facility, including PHYSICS TESTS. Table 14-2 (Ref. 5) summarizes the post-criticality tests. Requirements for reload fuel cycle PHYSICS TESTS are given in Reference 3. Although these PHYSICS TESTS are generally accomplished within the limits of all LCOs, conditions may occur when one or more of the LCOs must be suspended to make completion of PHYSICS TESTS possible or practical.

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

LCO 3.1.3, "Moderator Temperature Coefficient (MTC)";
LCO 3.1.5, "Safety Rod Position Limits";
LCO 3.2.1, "Regulating Rod Position Limits"; and
LCO 3.4.2, "RCS Minimum Temperature for Criticality."

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

Shutdown capability is preserved by limiting THERMAL POWER and maintaining adequate SDM, when in MODE 2 PHYSICS TESTS. In MODE 2, the Reactor Coolant System (RCS) temperature must be within the narrow range instrumentation for unit control. The narrow range temperature instrumentation goes on scale at 520°F. Therefore, it is considered safe to allow the minimum RCS temperature to decrease to 520°F during MODE 2 PHYSICS TESTS, based on the low probability of an accident occurring and on prior operating experience.

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

PHYSICS TESTS Exceptions – MODE 2 satisfy Criteria 1, 2, and 3 of 10 CFR 50.36 (Ref. 6).

LCO

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

BASES

LCO (continued)

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

a. THERMAL POWER is ≤ 5% RTP;

b. Nuclear overpower trip setpoints on the OPERABLE nuclear power range channels are set to ≤ 5% RTP;

c. Nuclear instrumentation wide range high startup rate CONTROL ROD withdrawal inhibit is OPERABLE;

d. SDM is maintained within the limit specified in the COLR; and

The limits of LCO 3.2.2 and LCO 3.2.3 are not exempted by this specification because they do not apply in MODE 2. Inhibiting CONTROL ROD withdrawal, based on startup rate, also limits local linear heat rate (LHR), departure from nucleate boiling ratio (DNBR), and peak RCS pressure during accidents initiated from low power.

APPLICABILITY

This LCO is applicable during PHYSICS TESTS in MODE 2 when the reactor is either subcritical or critical with THERMAL POWER ≤ 5% RTP. This LCO is applicable for low power testing, as defined by Regulatory Guide 1.68 (Ref. 3). In MODE 1, a test exception LCO is not required or allowed. In MODES 3, 4, 5, and 6, a test exception LCO is not required because the excepted LCOs do not apply in these MODES.

ACTIONS

A.1

If THERMAL POWER exceeds 5% RTP, a positive reactivity addition could be occurring, and a nuclear excursion could result. To ensure that local LHR, DNBR, and RCS pressure limits are not violated, the reactor is automatically tripped on a Reactor Protection System (RPS) high flux signal. However, if the RPS trip does not function when required, the necessary prompt action requires manual operator action to open the CONTROL ROD drive trip breakers without attempts to reduce THERMAL POWER by actuating the control system (i.e., CONTROL ROD insertion or RCS boration).

BASES

ACTIONS (continued)

B.1 and B.2

If the SDM requirements are not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. The operator should begin boration with the best source available for the unit conditions. Boration will be continued until SDM is within limit. In the determination of the required combination of boration flow rate and boron concentration, there is no unique requirement that must be satisfied.

Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable LCOs to within specification. A Completion Time of one hour is provided for the operator to restore compliance with the excepted LCOs.

C.1

If the nuclear overpower trip setpoint is > 5% RTP, then 1 hour is allowed for the operator to restore the nuclear overpower trip setpoint within limits or to complete an orderly suspension of PHYSICS TESTS exceptions. Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable individual LCOs to within specification, in order to ensure that continuity of reactor operation is within initial condition limits. This required Completion Time is consistent with, or more conservative than, those specified for the individual LCOs addressed by PHYSICS TESTS exceptions.

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

SURVEILLANCE REQUIREMENTS

SR 3.1.8.1

Verification that the nuclear overpower trip setpoint is within the limit specified for PHYSICS TESTS ensures that core protection at the reduced power level is established during PHYSICS TESTS. Performing the verification once within 8 hours prior to the performance of PHYSICS TESTS allows the operator adequate time for verifying the established trip setpoint margin before PHYSICS TESTS.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.8.2

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

a. RCS boron concentration;

b. CONTROL ROD position;

c. RCS average temperature;

d. Fuel burnup;

e. Xenon concentration; and

f. Moderator temperature coefficient (MTC).

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

REFERENCES

1. 10 CFR 50, Appendix B, Section XI.

2. 10 CFR 50.59.

3. UFSAR, Section 4.3.4.

4. UFSAR, Sections 14.3, 14.4 and 14.6.

5. UFSAR, Section 14.4, Table 14-2.

6. 10 CFR 50.36.

## B 3.2 POWER DISTRIBUTION LIMITS

### B 3.2.1 Regulating Rod Position Limits

BASES

BACKGROUND

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

The applicable criteria for these reactivity and power distribution design requirements are described in ONS Design Criteria (Ref. 1), and in 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 2).

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

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

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

The power density at any point in the core must be limited to maintain specified acceptable fuel design limits, including limits that ensure that the criteria specified in 10 CFR 50.46 (Ref. 2) are not violated. Together, LCO 3.2.1, "Regulating Rod Position Limits," LCO 3.2.2, "AXIAL POWER IMBALANCE Operating Limits," and LCO 3.2.3, "QUADRANT POWER TILT (QPT)," provide limits on control component operation and on monitored process variables to ensure that the core operates within the $F_Q(Z)$ and $F^N_{\Delta H}$ limits. $F_Q(Z)$ is the maximum local linear power density in

BASES

BACKGROUND (continued)

the core divided by the core average fuel rod linear power density, assuming nominal fuel pellet and fuel rod dimensions. Operation within the $F_Q(Z)$ limits prevents power peaks that would exceed the loss of coolant accident (LOCA) limits. $F^N_{\Delta H}$ is the ratio of the integral of linear power along the fuel rod on which minimum departure from nucleate boiling ratio occurs, to the average fuel rod power. Operation within the $F^N_{\Delta H}$ limits prevents departure from nucleate boiling (DNB) during an anticipated transient. In addition to the $F_Q(Z)$ and $F^N_{\Delta H}$ limits, certain reactivity limits are met by regulating rod position limits. The regulating rod position limits also restrict the ejected CONTROL ROD worth to the values assumed in the safety analysis and support the minimum required SDM in MODES 1 and 2.

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

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation or anticipated transients. The LCOs governing regulating rod position, AXIAL POWER IMBALANCE, and QPT preclude core power distributions that violate the following fuel design criteria:

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

b. During anticipated transients, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB condition.

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

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

Fuel cladding damage could result if an anticipated transient occurs with the simultaneous violation of one or more of the LCOs limiting the regulating rod position, the AXIAL POWER IMBALANCE, and the QPT. This potential for fuel cladding damage exists because changes in the power distribution can cause increased power peaking and correspondingly increased local linear heat rates (LHRs).

BASES

APPLICABLE SAFETY ANALYSES (continued)

The SDM requirement is met by limiting the regulating and safety rod position limits such that sufficient insertable reactivity is available in the rods to shut down the reactor to hot zero power with a reactivity margin that assumes that the maximum worth rod remains fully withdrawn upon trip (Ref. 4). Operation at the SDM based regulating rod position limit ensures that the maximum ejected rod worth is less than that assumed in the analyses.

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

The regulating rod and safety rod position limits ensure that the safety analysis assumptions for SDM, ejected rod worth, and power distribution peaking factors remain valid (Refs. 3 and 4).

The regulating rod position limits LCO satisfies Criterion 2 of 10 CFR 50.36 (Ref. 5).

LCO

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

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

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

LCO 3.1.1 has been modified by a Note that suspends the LCO requirement for those regulating rods not within the limits of the COLR solely due to testing in accordance with SR 3.1.4.2, which verifies the freedom of the rods to move. This SR may require the regulating rods to move below the LCO limit, which would otherwise violate the LCO.

APPLICABILITY

The regulating rod sequence, overlap, and physical position limits shall be maintained with the reactor in MODES 1 and 2. These limits maintain the validity of the assumed power distribution, ejected rod worth, SDM, and

BASES

APPLICABILITY (continued)

reactivity rate insertion assumptions used in the safety analyses. Applicability in MODES 3, 4, and 5 is not required, because neither the power distribution nor ejected rod worth assumptions are exceeded in these MODES. SDM in MODES 3, 4, and 5 is governed by LCO 3.1.1, "SHUTDOWN MARGIN (SDM)."

ACTIONS

The regulating rod position limits provided in the COLR are based on both the initial conditions assumed in the accident analyses and on the SDM. Specifically, separate position limits are specified to determine whether the unit is operating in violation of the initial conditions (e.g., the range of power distributions) assumed in the accident analyses or whether the unit is in violation of the SDM or ejected rod worth limits. Separate position limits are provided because different Required Actions and Completion Times apply, depending on which position limit has been violated. The area between the boundaries of acceptable operation and unacceptable operation, illustrated on the regulating rod position limit figures in the COLR, is the restricted region. The actions required when operation occurs with the regulating rod group sequence or overlap requirements not met are described under Condition A. The actions required when operation occurs in the restricted region and unacceptable region are described under Conditions B and C.

A.1

Operation with the regulating rod groups out of sequence or with the group overlap limits exceeded may represent a condition beyond the assumptions used in the safety analyses. The design calculations assume no deviation in nominal overlap between regulating rod groups. However, small deviations in group overlap, as allowed by the COLR, may occur and would not cause significant differences in core reactivity, in power distribution, or rod worth, relative to the design calculations. Group sequence must be maintained because design calculations assume the regulating rods withdraw and insert in a predetermined order. The Completion Time of 2 hours is intended to restrict operation in this condition because of the potential severity associated with gross violations of group sequence or overlap requirements. The 2 hour Completion Time is based on operating experience which supports the restoration time without unnecessarily challenging unit operation and the low probability of an event occurring simultaneously with the limit out of specification.

BASES

ACTIONS (continued)

B.1, B.2.1 and B.2.2

Operation in the unacceptable region shown on the figures in the COLR corresponds to power operation with an SDM less than the minimum required value or with the ejected rod worth greater than the allowable value. The regulating rods may be positioned too far to provide sufficient negative reactivity insertion following a reactor trip and the ejected rod worth may exceed its initial condition limit. Therefore, Required Action B.1 requires the RCS boron concentration be increased to restore the regulating rod position to a value that preserves the SDM and ejected rod worth limits. The RCS boration must occur as described in Section B 3.1.1. The required Completion Time of 15 minutes to initiate boration is reasonable, based on limiting the potential xenon redistribution, the low probability of an accident occurring in this relatively short time period, and the number of steps required to complete this Action. This period allows the operator sufficient time for aligning the required valves and for starting the boric acid pumps. Boration continues until the regulating rod group positions are restored to at least within the restricted region, which restores the minimum SDM and reduces the potential ejected rod worth to within its limit. Required Action B.1 is modified by a Note indicating that it is not applicable to regulating rod groups positioned in the restricted region.

Indefinite operation with the regulating rods inserted in the restricted region or unacceptable region is not prudent. Reactivity limits may not be met and the abnormal regulating rod position or group configuration may cause an adverse xenon redistribution, may cause the limits on AXIAL POWER IMBALANCE to be exceeded, or may adversely affect the long term fuel depletion pattern. Therefore, Required Action B.2.1 requires restoration of regulating rod groups to within their limits within 2 hours. This required Completion Time is reasonable based on the low probability of an event occurring simultaneously with the limit out of specification in this relatively short time period. In addition, it precludes long term depletion with abnormal group positions or configurations, thereby limiting the potential for an adverse xenon redistribution.

The regulating rods can also be restored within the acceptable position limits by reducing the THERMAL POWER to a value allowed by the regulating rod position limits in the COLR as allowed by Required Action B.2.2. The required Completion Time of 2 hours is sufficient to allow the operator to complete the power reduction in an orderly manner and without challenging the unit systems.

BASES

ACTIONS (continued)

C.1

If the Required Action and associated Completion Time of Condition A or B are not met, then the reactor must be placed in MODE 3, a MODE in which this LCO does not apply. This Action ensures that the reactor does not continue operating in violation of the peaking limits, the ejected rod worth, the reactivity insertion rate assumed as initial conditions in the accident analyses, or the required minimum SDM assumed in the accident analyses. The required Completion Time of 12 hours is reasonable, based on operating experience regarding the amount of time required to reach MODE 3 from RTP without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1

This Surveillance ensures that the sequence and overlap limits are not violated. A Surveillance Frequency of 12 hours is acceptable because little rod motion due to fuel burnup occurs in 12 hours. Also, the Frequency takes into account other information available in the control room for monitoring the status of the regulating rods.

SR 3.2.1.2

Verification of the regulating rod position limits as specified in the COLR at a Frequency of 12 hours is sufficient to detect whether the regulating rod groups may be approaching or exceeding their group position limits, because little rod motion occurs due to fuel burnup occurs in 12 hours. Also, the Frequency takes into account other information available in the control room for monitoring the status of the regulating rods.

SR 3.2.1.3

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

BASES (continued)

REFERENCES

1. UFSAR, Section 3.1.
2. 10 CFR 50.46.
3. UFSAR, Section 15.2.
4. UFSAR, Chapter 15.
5. 10 CFR 50.36.

## B 3.2 POWER DISTRIBUTION LIMITS

## B 3.2.2 AXIAL POWER IMBALANCE Operating Limits

BASES

BACKGROUND

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

The power density at any point in the core must be limited to maintain specified acceptable fuel design limits, including limits that satisfy the criteria specified in 10 CFR 50.46 (Ref. 1). This LCO provides limits on AXIAL POWER IMBALANCE to ensure that the core operates within the $F_Q(Z)$ and $F^N_{\Delta H}$ limits. $F_Q(Z)$ is the maximum local linear power density in the core divided by the core average fuel rod linear power density, assuming nominal fuel pellet and fuel rod dimensions. Operation within the $F_Q(Z)$ limits prevents power peaks that exceed the loss of coolant accident (LOCA) limits. $F^N_{\Delta H}$ is the ratio of the integral of linear power along the fuel rod on which minimum departure from nucleate boiling ratio occurs, to the average fuel rod power. Operation within the $F^N_{\Delta H}$ limits prevents departure from nucleate boiling (DNB) during an anticipated transient.

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

Fuel cladding failure during a postulated LOCA is limited by restricting the maximum linear heat rate (LHR) so that the peak cladding temperature does not exceed 2200°F (Ref. 1). Peak cladding temperatures > 2200°F cause severe cladding failure by oxidation due to a Zircaloy water reaction. Other criteria must also be met (e.g., maximum cladding oxidation, maximum hydrogen generation, coolable geometry, and long term cooling). However, peak cladding temperature is usually most limiting.

Proximity to the DNB condition is expressed by the departure from nucleate boiling ratio (DNBR), defined as the ratio of the cladding surface heat flux required to cause DNB to the actual cladding surface heat flux. The minimum DNBR value during both normal operation and anticipated transients is limited to the DNBR correlation limit for the particular fuel

BASES

BACKGROUND (continued)

design in use and is accepted as an appropriate margin to DNB. The DNB correlation limit ensures that there is at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB.

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

APPLICABLE SAFETY ANALYSES

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

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

b. During anticipated transients, there must be at least a 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB condition.

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

Fuel cladding damage could result should an anticipated transient occur with simultaneous violation of one or more of the LCOs governing the three process variables cited above. This potential for fuel cladding damage exists because changes in the power distribution can cause increased power peaking and corresponding increased local LHRs.

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

BASES

APPLICABLE SAFETY ANALYSES (continued)

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

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

AXIAL POWER IMBALANCE satisfies Criterion 2 of 10 CFR 50.36 (Ref. 2).

LCO

The power distribution LCO limits have been established based on correlations between power peaking and easily measured process variables: regulating rod position, AXIAL POWER IMBALANCE, and QPT. The AXIAL POWER IMBALANCE envelope contained in the COLR represents the setpoints for which the core power distribution would either exceed the LOCA LHR limits or cause a reduction in the DNBR below the Safety Limit during anticipated transients with the allowable QPT present and with the regulating rod positions consistent with the limitations on regulating rod positions determined by the fuel cycle design and specified by LCO 3.2.1.

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

APPLICABILITY

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

BASES (continued)

ACTIONS

A.1

The AXIAL POWER IMBALANCE operating limits that maintain the validity of the assumptions regarding the power distributions in the accident analyses of the LOCA and anticipated transients are provided in the COLR. Indefinite operation with the AXIAL POWER IMBALANCE outside the limits specified in the COLR is not prudent. Excessive AXIAL POWER IMBALANCE over an extended period of time may cause a potentially adverse xenon redistribution to occur. Therefore, operation is only allowed for a maximum of 2 hours. This required Completion Time is reasonable based on the low probability of a limiting event occurring simultaneously with the AXIAL POWER IMBALANCE outside the limits of this LCO. In addition, this limited Completion Time precludes long term depletion of the reactor fuel with excessive AXIAL POWER IMBALANCE and gives the operator sufficient time to reposition the APSRs or regulating rods to reduce the AXIAL POWER IMBALANCE because adverse effects of xenon redistribution and fuel depletion are limited.

B.1

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

SURVEILLANCE REQUIREMENTS

The AXIAL POWER IMBALANCE can be monitored by both the Incore and Excore Detector Systems. The AXIAL POWER IMBALANCE maximum allowable setpoints are derived from their corresponding measurement system independent limits by adjusting for both the system observability errors and instrumentation errors. Although they may be based on the same measurement system independent limits, the setpoints for the different systems are not identical because of differences in the errors applicable for each of these systems. The uncertainty analysis that defines the required error adjustment to convert the measurement system

BASES

SURVEILLANCE REQUIREMENTS (continued)

independent limits to alarm setpoints assumes that 75% of the detectors are OPERABLE. Detectors located on the core major axes are assumed to contribute one half of their output to each quadrant; detectors in the center assembly are assumed to contribute one quarter of their output to each quadrant. For AXIAL POWER IMBALANCE measurements using the Incore Detector System, the Backup Incore Detector System consists of OPERABLE detectors configured as follows:

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

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

c. The detector strings shall not have radial symmetry.

Figure B 3.2.2-1 (Backup Incore Detector System for AXIAL POWER IMBALANCE Measurement) depicts an example of this configuration. This arrangement is chosen to reduce the uncertainty in the measurement of the AXIAL POWER IMBALANCE by the Backup Incore Detector System. For example, the requirement for placing one detector of each of the three strings at the core midplane puts three detectors in the central region of the core where the neutron flux tends to be higher. It also helps prevent measuring an AXIAL POWER IMBALANCE that is excessively large when the reactor is operating at low THERMAL POWER levels. The third requirement for placement of detectors (i.e., radial asymmetry) reduces uncertainty by measuring the neutron flux at core locations that are not radially symmetric.

The Excore Detector System consists of four detectors (one located outside each quadrant of the core). Each detector consists functionally of two six-foot uncompensated ion chambers adjacent to the top and bottom halves of the core. Comparison of the signals from the two detectors gives an indication of the core axial offset or imbalance.

SR 3.2.2.1

Verification of the AXIAL POWER IMBALANCE indication every 12 hours ensures that the AXIAL POWER IMBALANCE limits are not violated and takes into account other information and alarms available to the operator in

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1 (continued)

the control room. This Surveillance Frequency is acceptable because the mechanisms that can cause AXIAL POWER IMBALANCE, such as xenon redistribution or CONTROL ROD drive mechanism malfunctions that cause slow AXIAL POWER IMBALANCE increases, can be discovered by the operator before the specified limits are violated.

REFERENCES

1. 10 CFR 50.46.

2. 10 CFR 50.36.

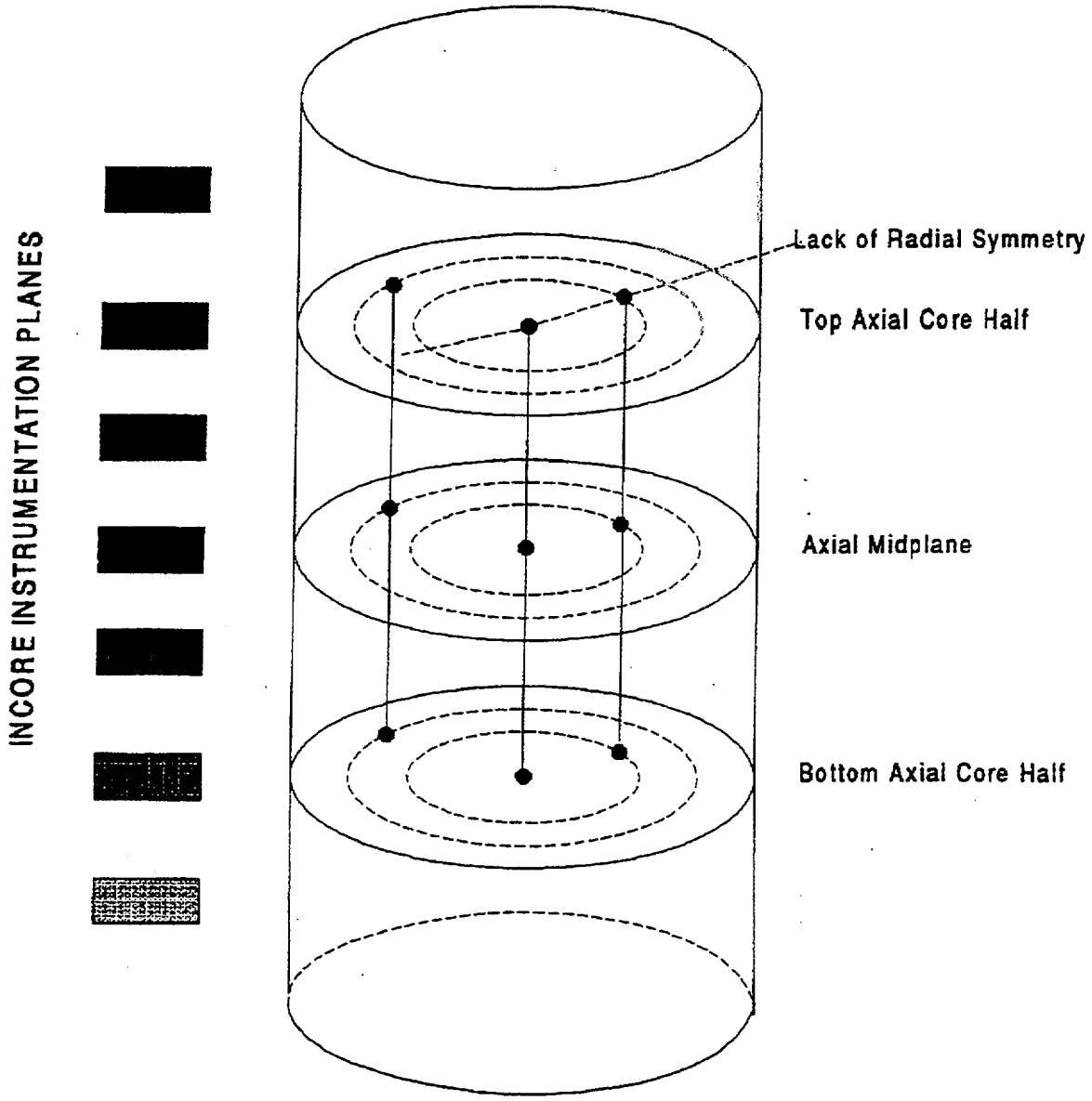



Figure B 3.2.2-1 (page 1 of 1)
Backup Incore Detector System for AXIAL POWER IMBALANCE Measurement

## B 3.2 POWER DISTRIBUTION LIMITS

## B 3.2.3 QUADRANT POWER TILT (QPT)

### BASES

BACKGROUND

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

The power density at any point in the core must be limited to maintain specified acceptable fuel design limits, including limits that preserve the criteria specified in 10 CFR 50.46 (Ref. 1). Together, LCO 3.2.1, "Regulating Rod Position Limits," LCO 3.2.2, "AXIAL POWER IMBALANCE Operating Limits," and LCO 3.2.3, "QUADRANT POWER TILT (QPT)," provide limits on control component operation and on monitored process variables to ensure that the core operates within the $F_Q(Z)$ and $F^N_{\Delta H}$ limits. $F_Q(Z)$ is the maximum local linear power density in the core divided by the core average fuel rod linear power density, assuming nominal fuel pellet and fuel rod dimensions. Operation within the $F_Q(Z)$ limits prevents power peaks that exceed the loss of coolant accident (LOCA) limits. $F^N_{\Delta H}$ is the ratio of the integral of linear power along the fuel rod on which minimum departure from nucleate boiling ratio occurs, to the average fuel rod power. Operation within the $F^N_{\Delta H}$ limits prevents departure from nucleate boiling (DNB) during an anticipated transient.

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

Fuel cladding failure during a postulated LOCA is limited by restricting the maximum linear heat rate (LHR) so that the peak cladding temperature does not exceed 2200°F (Ref. 1). Peak cladding temperatures > 2200°F cause severe cladding failure by oxidation due to a Zircaloy water reaction. Other criteria must also be met (e.g., maximum cladding oxidation, maximum hydrogen generation, coolable geometry, and long term cooling). However, peak cladding temperature is usually most limiting.

BASES

BACKGROUND (continued)

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

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

APPLICABLE SAFETY ANALYSES

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

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

b. During anticipated transients, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB condition.

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

Fuel cladding damage could result if an anticipated transient occurs with simultaneous violation of one or more of the LCOs governing the core power distribution. Changes in the power distribution can cause increased power peaking and correspondingly increased local LHRs.

The dependence of the core power distribution on burnup, regulating rod insertion, and spatial xenon distribution is taken into account during the reload safety evaluation analysis. An allowance for QPT is accommodated in the analysis and resultant LCO limits. The increase in peaking taken for QPT is developed from a database of full core power distribution

BASES

APPLICABLE SAFETY ANALYSES (continued)

calculations (Ref. 2). The calculations consist of simulations of many power distributions with tilt causing mechanisms (e.g., dropped or misaligned CONTROL RODS, misloaded assemblies, and burnup gradients). An increase of < 2% peak power per 1% QPT is supported by the analysis, therefore a value of 2% peak power increase per 1% QPT is used to bound peak power increases due to QPT.

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

QPT satisfies Criterion 2 of 10 CFR 50.36 (Ref. 3).

LCO

The power distribution LCO limits have been established based on correlations between power peaking and easily measured process variables: regulating rod position, AXIAL POWER IMBALANCE, and QPT. The regulating rod position limits and the AXIAL POWER IMBALANCE boundaries contained in the COLR represent the measurement system independent limits. These are the limits at which the core power distribution either exceeds the LOCA LHR limits or causes a reduction in DNBR below the safety limit during anticipated transients with the allowable QPT present and with regulating rod position consistent with the limitations on regulating rod positions determined by the fuel cycle design and specified by LCO 3.2.1.

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

APPLICABILITY

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

BASES

APPLICABILITY (continued)

In MODE 2, the combination of QPT with maximum ALLOWABLE THERMAL POWER level does not result in LHRs sufficiently large to violate the fuel design limits, and therefore, applicability in this MODE is not required. Although not specifically addressed in the LCO, QPTs greater than the maximum limit specified in the COLR in MODE 1 with THERMAL POWER < 20% RTP are allowed for the same reason.

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

ACTIONS

A.1

The steady state limit specified in the COLR provides an allowance for QPT that may occur during normal operation. A peaking increase to accommodate QPTs up to the steady state limit is allowed by the regulating rod position limits of LCO 3.2.1 and the AXIAL POWER IMBALANCE limits of LCO 3.2.2.

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

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

A.2

Power operation is allowed to continue if THERMAL POWER is reduced in accordance with Required Action A.1. The same reduction (i.e., 2% RTP or more) is also applicable to the nuclear overpower trip setpoints (flux and flux/flow imbalance), for each 1% of QPT in excess of the steady state limit. This reduction maintains both core protection and thermal margins at the

BASES

ACTIONS

A.2 (continued)

reduced THERMAL POWER level similar to that at RTP. The required Completion Time of 10 hours is reasonable based on the need to limit the potentially adverse xenon redistribution, the low probability of an accident occurring while operating out of specification, and the number of steps required to complete the Required Action.

A.3

Although the actions directed by Required Action A.1 restore margins, if the source of the QPT is not determined and corrected, it is prudent to establish increased margins. A required Completion Time of 24 hours to reduce QPT to less than the steady state limit is a reasonable time for investigation and corrective measures.

B.1

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

B.2

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

BASES

ACTIONS (continued)

C.1

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

C.2

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

D.1

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

D.2

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

BASES

ACTIONS (continued)

E.1

If the Required Action and associated Completion Time for Condition C or D are not met, then the reactor will continue in power operation with significant QPT. Either the power level has not been reduced to comply with the Required Action or the nuclear overpower trip setpoints (flux and flux/flow imbalance) have not been reduced within the required Completion Time. To preclude risk of fuel damage in any of these conditions, THERMAL POWER is reduced further. Operation at 20% RTP allows the operator to investigate the cause of the QPT and to correct it. Local LHRs with a large QPT do not violate the fuel design limits at or below 20% RTP. The required Completion Time of 4 hours is acceptable based on limiting the potential increase in local LHRs that could occur due to xenon redistribution with the QPT out of specification.

F.1

QPT in excess of the maximum limit specified in the COLR can be an indication of a severe power distribution anomaly, and a power reduction to at most 20% RTP ensures local LHRs do not exceed allowable limits while the cause is being determined and corrected.

The required Completion Time of 4 hours is reasonable to allow the operator to reduce THERMAL POWER to ≤ 20% RTP without challenging unit systems.

SURVEILLANCE REQUIREMENTS

QPT can be monitored by both the Incore and Excore Detector Systems. The QPT limits are derived from their corresponding measurement system independent limits by adjustment for system observability errors and instrumentation errors. Although they may be based on the same measurement system independent limit, the limits for the different systems are not identical because of differences in the errors applicable for these systems. For QPT measurements using the Incore Detector System, the Backup Incore Detector System consists of OPERABLE detectors configured as follows:

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

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

BASES

SURVEILLANCE REQUIREMENTS (continued)

Figure B 3.2.3-1 (Backup Incore Detector System for QPT Measurement) depicts an example of this configuration.

The Excore Detector System consists of four detectors (one located outside each quadrant of the core). Each detector consists functionally of two six-foot uncompensated ion chambers adjacent to the top and bottom halves of the core.

SR 3.2.3.1

Checking the QPT indication every 7 days ensures that the operator can determine whether the plant computer software and Incore Detector System inputs for monitoring QPT are functioning properly, and takes into account other information and alarms available to the operator in the Control Room. This procedure allows the QPT mechanisms, such as xenon redistribution, burnup gradients, and CONTROL ROD drive mechanism malfunctions, which can cause slow development of a QPT, to be detected. Operating experience has confirmed the acceptability of a Surveillance Frequency of 7 days.

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

REFERENCES

1. 10 CFR 50.46.

2. BAW 10122A, "Normal Operating Controls," Rev. 1, May 1984.

3. 10 CFR 50.36.

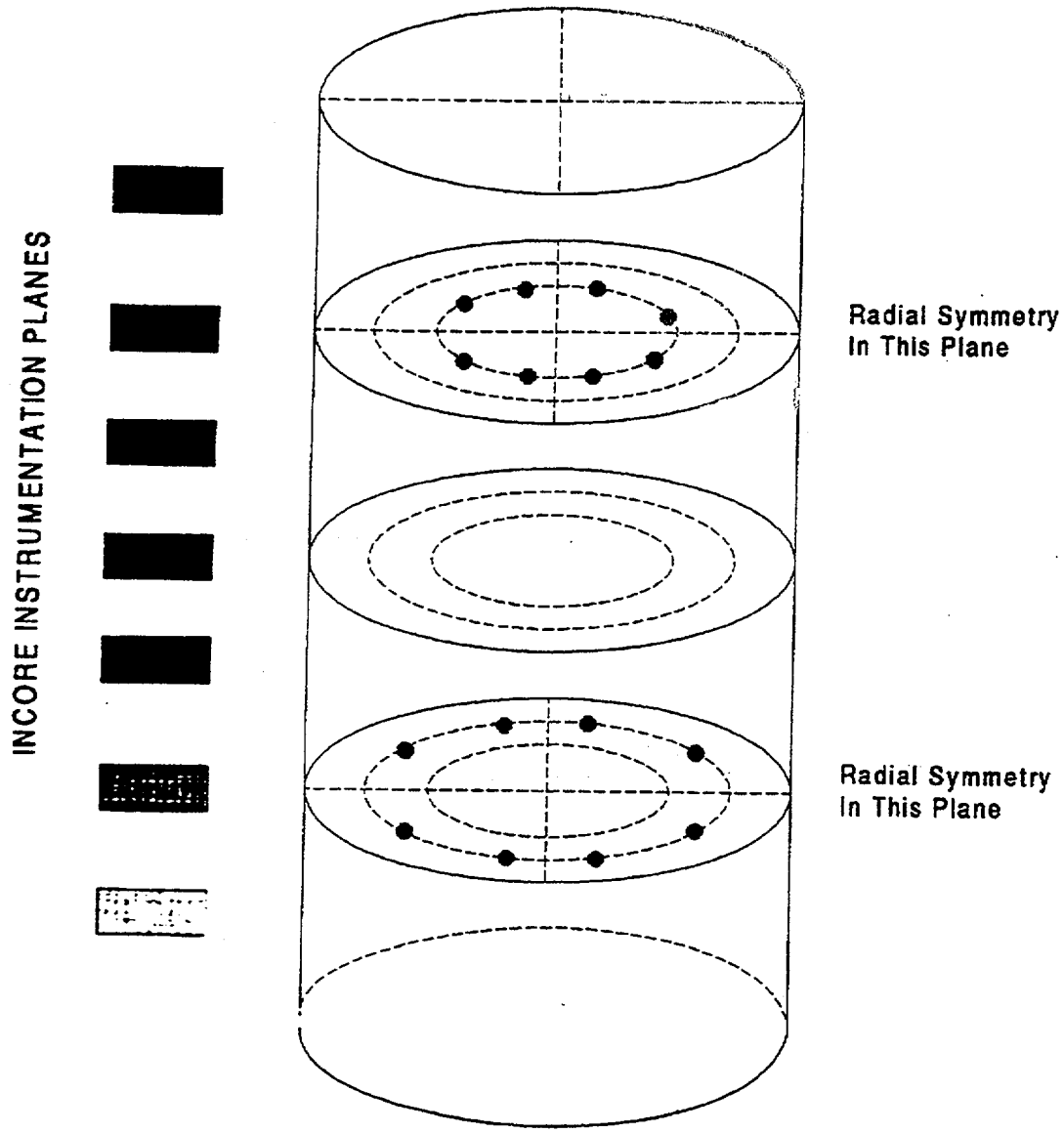



Figure B 3.2.3-1 (page 1 of 1)
Backup Incore Detector System for QUADRANT POWER TILT Measurement

# B 3.3 INSTRUMENTATION

## B 3.3.1 Reactor Protective System (RPS) Instrumentation

BASES

BACKGROUND

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

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

The LSSS, defined in this Specification as the Allowable Value, in conjunction with the LCOs, establishes the threshold for protective system action to prevent exceeding acceptable limits during accidents or transients.

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

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

b. Fuel centerline melt shall not occur; and

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

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

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

BASES

BACKGROUND (continued)

RPS Overview

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

Figure 7.1, UFSAR, Chapter 7 (Ref. 1), shows the arrangement of a typical RPS protective channel. A protective channel is composed of measurement channels, a manual trip channel, a reactor trip module (RTM), and control rod drive (CRD) trip devices. LCO 3.3.1 provides requirements for the individual measurement channels. These channels encompass all equipment and electronics from the point at which the measured parameter is sensed through the bistable relay contacts in the trip string. LCO 3.3.2, "Reactor Protective System (RPS) Manual Reactor Trip," LCO 3.3.3, "Reactor Protective System (RPS) – Reactor Trip Module (RTM)," and LCO 3.3.4, "control rod Drive (CRD) Trip Devices," discuss the remaining RPS elements.

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

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

The RPS consists of four independent protective channels, each containing an RTM. The RTM receives signals from its own measurement channels that indicate a protective channel trip is required. The RTM transmits this signal to its own two-out-of-four trip logic and to the two-out-of-four logic of the RTMs in the other three RPS channels. Whenever any two RPS channels transmit channel trip signals, the RTM logic in each channel actuates to remove 120 VAC power from its associated CRD trip device.

BASES

BACKGROUND

RPS Overview (continued)

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

The RPS has three bypasses: a shutdown bypass, a dummy bistable and an RPS channel bypass. Shutdown bypass allows the withdrawal of safety rods for SDM availability and rapid negative reactivity insertion during unit cooldowns or heatups. The dummy bistable is used to bypass one or more functions (bistable trips) associated with one RPS Channel. The RPS Channel bypass allows one entire RPS channel to be taken out of service for maintenance and testing. Test circuits in the trip strings allow complete testing of all RPS trip Functions.

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

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

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

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

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

Power Range Nuclear Instrumentation

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

1. Nuclear Overpower

BASES

BACKGROUND

Power Range Nuclear Instrumentation (continued)

a. Nuclear Overpower – High Setpoint;

b. Nuclear Overpower – Low Setpoint;

7. Reactor Coolant Pump to Power;

8. Nuclear Overpower Flux/Flow Imbalance;

9. Main Turbine Trip (Hydraulic Fluid Pressure); and

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

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

Reactor Coolant System Outlet Temperature

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

2. RCS High Outlet Temperature; and

5. RCS Variable Low Pressure.

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

BASES

BACKGROUND (continued)

Reactor Coolant System Pressure

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

3. RCS High Pressure;

4. RCS Low Pressure;

5. RCS Variable Low Pressure; and

11. Shutdown Bypass RCS High Pressure.

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

Reactor Building Pressure

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

Reactor Coolant Pump Power Monitoring

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

Reactor Coolant System Flow

The Reactor Coolant System Flow measurements are an input to the Nuclear Overpower Flux/Flow Imbalance trip, Function 8. The reactor coolant flow inputs to the RPS are provided by eight high accuracy differential pressure transmitters, four on each loop, which measure flow through calibrated flow tubes. One flow input in each loop is associated with each protective channel.

BASES

BACKGROUND (continued)

Main Turbine Automatic Stop Oil Pressure

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

Feedwater Pump Hydraulic Oil Pressure

Feedwater Pump Hydraulic Oil Pressure is an input to the Loss of Main Feedwater Pumps (Hydraulic Oil Pressure) trip, Function 10. Hydraulic Oil pressure is measured by four switches on each feedwater pump. One switch on each pump, connected in series with a switch on the other MFW pump, is associated with each protective channel.

RPS Bypasses

The RPS is designed with three types of bypasses: dummy bistable, channel bypass and shutdown bypass.

The dummy bistable provides a method of placing one or more functions in a RPS protective channel in a bypassed condition, the channel bypass provides a method of placing all Functions in one RPS protective channel in a bypassed condition, and shutdown bypass provides a method of leaving the safety rods withdrawn during cooldown and depressurization of the RCS. Each bypass is discussed next.

Dummy Bistable

The dummy bistable is used to bypass one or more functions (bistable trips) associated with one RPS Channel. A dummy bistable is used if a parameter in an RPS channel fails and causes that channel to trip. Dummy bistables may be used in only one RPS channel at a time. Also, if an RPS channel is bypassed, no other RPS channel may contain a dummy bistable. Inserting a dummy bistable in the place of a failed (tripped) bistable allows the RPS channels to be reset, thus allowing the remainder of the functions in that RPS channel to be returned to service. This is more

BASES

BACKGROUND

Dummy Bistable (continued)

conservative than manually bypassing the entire RPS channel. For an RPS channel with a dummy bistable installed, only the affected function(s) is inoperable. The installation of the STAR hardware in the nuclear overpower flux/flow imbalance trip string requires the use of jumpers to bypass the trip string. The installation of these jumpers does not require the removal of the STAR processor module, therefore, the protective channel is not forced into a tripped condition.

Channel Bypass

A channel bypass provision is provided to allow for maintenance and testing of the RPS. The use of channel bypass keeps the protective channel trip relay energized regardless of the status of the instrumentation channel of the bistable relay contacts. To place a protective channel in channel bypass, the other three channels must not be in channel bypass or otherwise inoperable (e.g., a dummy bistable installed). This can be verified by observing alarms/indicator lights. This is administratively controlled by having only one manual bypass key available for each unit. All RPS trips are reduced to a two-out-of-three logic in channel bypass.

Shutdown Bypass

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

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

In shutdown bypass, a normally closed contact opens when the operator closes the shutdown bypass key switch (status shall be indicated by a light). This action bypasses the RCS Low Pressure trip, Nuclear Overpower Flux/Flow Imbalance trip, Reactor Coolant Pump to Power trip,

BASES

BACKGROUND

Shutdown Bypass (continued)

and the RCS Variable Low Pressure trip, and inserts a new RCS High Pressure, 1720 psig trip. The operator can now withdraw the safety rods for additional rapidly insertable negative reactivity.

The insertion of the new high pressure trip performs two functions. First, with a trip setpoint of 1720 psig, the bistable prevents operation at normal system pressure, 2155 psig, with a portion of the RPS bypassed. The second function is to ensure that the bypass is removed prior to normal operation. When the RCS pressure is increased during a unit heatup, the safety rods are inserted prior to reaching 1720 psig. The shutdown bypass is removed, which returns the RPS to normal, and system pressure is increased to greater than 1800 psig. The safety rods are then withdrawn and remain at the full out condition for the rest of the heatup.

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

Module Interlock and Test Trip Relay

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

Trip Setpoints/Allowable Value

The Allowable Value and trip setpoint are based on the analytical limits stated in UFSAR, Chapter 15 (Ref. 2). The selection of the Allowable Value and associated trip setpoint is such that adequate protection is provided when all sensor and processing time delays are taken into account. To allow for calibration tolerances, instrumentation uncertainties, instrument drift, and severe environment errors for those RPS channels that must function in harsh environments as defined by 10 CFR 50.49

BASES

BACKGROUND

Trip Setpoints/Allowable Value (continued)

(Ref. 3), the Allowable Values specified in Table 3.3.1-1 in the accompanying LCO are conservative with respect to the analytical limits to account for all known uncertainties for each channel. The actual trip setpoint entered into the bistable is more conservative than that specified by the Allowable Value to account for changes in random measurement errors detectable by a CHANNEL FUNCTIONAL TEST. One example of such a change in measurement error is drift during the Surveillance Frequency. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value. All field sensors and signal processing equipment for these channels are assumed to operate within the allowances of these uncertainty magnitudes. The trip setpoints are the nominal values at which the bistables are set. Any bistable is considered to be properly adjusted when the "as left" value is within the band for CHANNEL CALIBRATION accuracy. A detailed description of the methodology used to determine the Allowable Value, trip setpoints, and associated uncertainties is provided in Reference 4.

Setpoints in accordance with the Allowable Value ensure that the limits of Chapter 2.0, "Safety Limits," in the Technical Specifications are not violated during anticipated transients and that the consequences of accidents will be acceptable, providing the unit is operated from within the LCOs at the onset of the anticipated transient or accident and the equipment functions as designed. Note that in LCO 3.3.1 the Allowable Values listed in Table 3.3.1-1 for Functions 1 through 8 and 11 are the LSSS.

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

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

Each of the analyzed accidents and transients that require a reactor trip to meet the acceptance criteria can be detected by one or more RPS Functions. The accident analysis contained in the UFSAR, Chapter 15 (Ref. 2), takes credit for most RPS trip Functions. Functions not specifically credited in the accident analysis were qualitatively credited in the safety analysis and the NRC staff approved licensing basis for the unit. These Functions are high RB pressure, high RCS temperature, turbine trip, and loss of main feedwater. These Functions may provide protection for conditions that do not require dynamic transient analysis to demonstrate Function performance. These Functions also serve as backups to Functions that were credited in the safety analysis.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

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

Only the Allowable Values are specified for each RPS trip Function in the LCO. Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the setpoint measured by CHANNEL FUNCTIONAL TESTS does not exceed the Allowable Value if the bistable is performing as required. A trip setpoint found less conservative than the nominal trip setpoint, but within its Allowable Value, is considered OPERABLE with respect to the uncertainty allowances assumed for the applicable surveillance interval provided that operation, testing and subsequent calibration are consistent with the assumptions of the setpoint calculations. Each Allowable Value specified is more conservative than instrument uncertainties appropriate to the trip Function. These uncertainties are defined in Reference 4.

For most RPS Functions, the Allowable Value in conjunction with the nominal trip setpoint ensure that the departure from nucleate boiling (DNB), center line fuel melt, or RCS pressure SLs are not challenged. Cycle specific values for use during operation are contained in the COLR.

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

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

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

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

1. Nuclear Overpower

a. Nuclear Overpower – High Setpoint

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

The Nuclear Overpower – High Setpoint trip initiates a reactor trip when the neutron power reaches a predefined setpoint at the design overpower limit. Because THERMAL POWER lags the neutron power, tripping when the neutron power reaches the design overpower will limit THERMAL POWER to prevent exceeding acceptable fuel damage limits.

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

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

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

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

b. Nuclear Overpower – Low Setpoint

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

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

2. RCS High Outlet Temperature

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

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

3. RCS High Pressure

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

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

3. RCS HIGH PRESSURE (continued)

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

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

4. RCS Low Pressure

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

The RCS Low Pressure setpoint Allowable Value is selected to ensure that a reactor trip occurs before RCS pressure is reduced below the lowest point at which the RCS Variable Low Pressure trip is analyzed. The RCS Low Pressure trip provides protection for primary system depressurization events and has been credited in the accident analysis calculations for small break loss of coolant accidents (LOCAs). Harsh RB conditions created by small break LOCAs cannot affect performance of the RCS pressure sensors and transmitters within the time frame for a reactor trip. Therefore, degraded environmental conditions are not considered in the Allowable Value determination.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

5. RCS Variable Low Pressure

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

The RCS Variable Low Pressure setpoint Allowable Value is selected to ensure that a trip occurs when temperature and pressure approach the conditions necessary for DNB while operating in a temperature pressure region constrained by the low pressure and high temperature trips. The RCS Variable Low Pressure trip is assumed for transient protection in the main steam line break analysis. The setpoint allowable value does not include errors induced by the harsh environment, because the trip actuates prior to the harsh environment.

6. Reactor Building High Pressure

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

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

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

7. Reactor Coolant Pump to Power

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

The Reactor Coolant Pump to Power trip has been credited in the accident analysis calculations for the loss of more than two RCPs.

The Allowable Value for the Reactor Coolant Pump to Power trip setpoint is selected to prevent normal power operation unless at least three RCPs are operating. RCP status is monitored by power transducers on each pump. These relays indicate a loss of an RCP on underpower. The underpower setpoint is selected to reliably trip on loss of voltage to the RCPs. Neither the reactor power nor the pump power setpoint account for instrumentation errors caused by harsh environments because the trip Function is not required to respond to events that could create harsh environments around the equipment.

8. Nuclear Overpower Flux/Flow Imbalance

The Nuclear Overpower Flux/Flow Imbalance trip provides steady state protection for the power imbalance SLs. A reactor trip is initiated prior to the core power, AXIAL POWER IMBALANCE, and reactor coolant flow conditions exceeding the DNB or fuel centerline temperature limits.

This trip supplements the protection provided by the Reactor Coolant Pump to Power trip, through the power to flow ratio, for loss of reactor coolant flow events. The power to flow ratio provides direct protection for the DNBR SL for the loss of one or more RCPs and for locked RCP rotor accidents.

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

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

8. Nuclear Overpower Flux/Flow Imbalance (continued)

The Allowable Value is selected to ensure that a trip occurs when the core power, axial power peaking, and reactor coolant flow conditions indicate an approach to DNB or fuel centerline temperature limits. By measuring reactor coolant flow and by tripping only when conditions approach an SL, the unit can operate with the loss of one pump from a four pump initial condition at power levels at least as low as approximately 80% RTP. The Allowable Value for the Function is given in the unit COLR because the cycle specific core peaking changes affect the Allowable Value.

9. Main Turbine Trip (Hydraulic Fluid Pressure)

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

Each of the four turbine hydraulic fluid pressure switches feeds one protective channel through buffers that continuously monitor the status of the contacts.

For the Main Turbine Trip (Hydraulic Fluid Pressure) bistable, the Allowable Value of 800 psig is selected to provide a trip whenever main turbine hydraulic fluid pressure drops below the normal operating range. To ensure that the trip is enabled as required by the LCO, the reactor power bypass is set with an Allowable Value of 30% RTP. The turbine trip is not required to protect against events that can create a harsh environment in the turbine building. Therefore, errors induced by harsh environments are not included in the determination of the setpoint Allowable Value.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

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

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

For the feedwater pump hydraulic oil pressure bistables, the Allowable Value of 75 psig is selected to provide a trip whenever feedwater pump hydraulic oil pressure drops below the normal operating range. To ensure that the trip is enabled as required by the LCO, the reactor power bypass is set with an Allowable Value of 2% RTP. The Loss of Main Feedwater Pumps (Hydraulic Oil Pressure) trip is not required to protect against events that can create a harsh environment in the turbine building. Therefore, errors caused by harsh environments are not included in the determination of the setpoint Allowable Value.

11. Shutdown Bypass RCS High Pressure

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

Accidents analyzed in the UFSAR, Chapter 15 (Ref. 2), do not describe events that occur during shutdown bypass operation, because the consequences of these events are enveloped by the events presented in the UFSAR.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

11. Shutdown Bypass RCS High Pressure (continued)

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

1.a Nuclear Overpower – High Setpoint;

3. RCS High Pressure;

4. RCS Low Pressure;

5. RCS Variable Low Pressure;

7. Reactor Coolant Pump to Power; and

8. Nuclear Overpower Flux/Flow Imbalance.

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

General Discussion

The RPS satisfies Criterion 3 of 10 CFR 50.36 (Ref. 8). In MODES 1 and 2, the following trips shall be OPERABLE because the reactor can be critical in these MODES. These trips are designed to take the reactor subcritical to maintain the SLs during anticipated transients and to assist the ESPS in providing acceptable consequences during accidents.

1a. Nuclear Overpower – High Setpoint;

2. RCS High Outlet Temperature;

3. RCS High Pressure;

4. RCS Low Pressure;

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

General Discussion (continued)

5. RCS Variable Low Pressure;

6. Reactor Building High Pressure;

7. Reactor Coolant Pump to Power; and

8. Nuclear Overpower Flux/Flow Imbalance.

Functions 1, 3, 4, 5, 7, and 8 just listed may be bypassed in MODE 2 when RCS pressure is below 1720 psig, provided the Shutdown Bypass RCS High Pressure and the Nuclear Overpower – Low setpoint trip are placed in operation. Under these conditions, the Shutdown Bypass RCS High Pressure trip and the Nuclear Overpower – Low setpoint trip act to prevent unit conditions from reaching a point where actuation of these Functions is necessary.

The Main Turbine Trip (Hydraulic Fluid Pressure) Function is required to be OPERABLE in MODE 1 at ≥ 30% RTP. The Loss of Main Feedwater Pumps (Hydraulic Oil Pressure) Function is required to be OPERABLE in MODE 1 and in MODE 2 at ≥ 2% RTP. Analyses presented in BAW-1893 (Ref. 6) have shown that for operation below these power levels, these trips are not necessary to minimize challenges to the PORVs as required by NUREG-0737 (Ref. 5).

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

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

ACTIONS

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

BASES

ACTIONS (continued)

A.1

For Required Action A.1, if one or more Functions in a required protective channel becomes inoperable, the affected protective channel must be placed in trip. This Required Action places all RPS Functions in a one-out-of-two logic configuration. The "non-required" channel is placed in bypass when the required inoperable channel is placed in trip to prevent bypass of a second required channel. In this configuration, the RPS can still perform its safety functions in the presence of a random failure of any single Channel. The 1 hour Completion Time is sufficient time to perform Required Action A.1.

B.1

Required Action B.1 directs entry into the appropriate Condition referenced in Table 3.3.1-1. The applicable Condition referenced in the table is Function dependent. If the Required Action and the associated Completion Time of Condition A are not met or if more than two channels are inoperable, Condition B is entered to provide for transfer to the appropriate subsequent Condition.

C.1 and C.2

If the Required Action and associated Completion Time of Condition A are not met and Table 3.3.1-1 directs entry into Condition C, the unit must be brought to a MODE in which the specified RPS trip Functions are not required to be OPERABLE. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and to open all CRD trip breakers without challenging unit systems.

D.1

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

BASES

ACTIONS (continued)

E.1

If the Required Action and associated Completion Time of Condition A are not met and Table 3.3.1-1 directs entry into Condition E, the unit must be brought to a MODE in which the specified RPS trip Function is not required to be OPERABLE. To achieve this status, THERMAL POWER must be reduced < 30% RTP. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach 30% RTP from full power conditions in an orderly manner without challenging unit systems.

F.1

If the Required Action and associated Completion Time of Condition A are not met and Table 3.3.1-1 directs entry into Condition F, the unit must be brought to a MODE in which the specified RPS trip Function is not required to be OPERABLE. To achieve this status, THERMAL POWER must be reduced < 2% RTP. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach 2% RTP from full power conditions in an orderly manner without challenging unit systems.

SURVEILLANCE REQUIREMENTS

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

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

SR 3.3.1.1

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

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1 (continued)

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

The Frequency, equivalent to once every shift, is based on operating experience that demonstrates channel failure is rare. Since the probability of two random failures in redundant channels in any 12 hour period is extremely low, the CHANNEL CHECK minimizes the chance of loss of protective function due to failure of redundant channels. The CHANNEL CHECK supplements less formal but more frequent checks of channel OPERABILITY during normal operational use of the displays associated with the LCO's required channels.

For Functions that trip on a combination of several measurements, such as the Nuclear Overpower Flux/Flow Imbalance Function, the CHANNEL CHECK must be performed on each input.

SR 3.3.1.2

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

The power range channel's output shall be adjusted consistent with the calorimetric results if the calorimetric exceeds the power range channel's output by ≥ 2% RTP. The value of 2% is adequate because this value is

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.2 (continued)

assumed in the safety analyses of UFSAR, Chapter 15 (Ref. 2). These checks and, if necessary, the adjustment of the power range channels ensure that channel accuracy is maintained within the analyzed error margins. The 24 hour Frequency is adequate, based on unit operating experience, which demonstrates the change in the difference between the power range indication and the calorimetric results rarely exceeds a small fraction of 2% in any 24 hour period. Furthermore, the control room operators monitor redundant indications and alarms to detect deviations in channel outputs.

SR 3.3.1.3

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

SR 3.3.1.4

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

The as found and as left values must also be recorded and reviewed for consistency with the assumptions of the surveillance interval extension analysis. The requirements for this review are outlined in BAW-10167 (Ref. 7).

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.4 (continued)

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

SR 3.3.1.5

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

A CHANNEL CALIBRATION is a complete check of the instrument channel, including the sensor. The test verifies that the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift to ensure that the instrument channel remains operational between successive tests. CHANNEL CALIBRATION shall find that measurement errors and bistable setpoint errors are within the assumptions of the setpoint analysis. CHANNEL CALIBRATIONS must be performed consistent with the assumptions of the setpoint analysis. Whenever a sensing element is replaced, the next required CHANNEL CALIBRATION of the resistance temperature detectors (RTD)sensors is accomplished by an inplace cross calibration that compares the other sensing elements with the recently installed sensing element.

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

REFERENCES

1. UFSAR, Chapter 7.

2. UFSAR, Chapter 15.

3. 10 CFR 50.49.

4. EDM-102, "Instrument Setpoint/Uncertainty Calculations."

5. NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1979.

BASES

REFERENCES (continued)

6. BAW-1893, "Basis for Raising Arming Threshold for Anticipating Reactor Trip on Turbine Trip," October 1985.

7. BAW-10167, May 1986.

8. 10 CFR 50.36.

## B 3.3 INSTRUMENTATION

### B 3.3.2 Reactor Protective System (RPS) Manual Reactor Trip

BASES

**BACKGROUND**

The RPS Manual Reactor Trip provides the operator with the capability to trip the reactor from the control room. Manual trip is provided by a trip push button on the main control board. This push button operates four electrically independent switch contacts, one for each train. This trip is independent of the automatic trip system. As shown in Figure 7.1, UFSAR, Chapter 7 (Ref. 1), power for the control rod drive (CRD) breaker undervoltage coils and contactor coils comes from the reactor trip modules (RTMs). The manual trip switch contacts are located between the RTM output and the breaker undervoltage coils. Opening of the switch contacts opens the lines to the breakers, tripping them. The switch contacts also energize the breaker shunt trip mechanisms. There is a separate switch contact in series, with the output of each of the four RTMs. All switch contacts are actuated through a mechanical linkage from a single push button.

**APPLICABLE SAFETY ANALYSES**

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

The Manual Reactor Trip Function satisfies Criterion 3 of 10 CFR 50.36 (Ref. 2).

**LCO**

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

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

BASES (continued)

APPLICABILITY

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

ACTIONS

A.1

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

B.1 and B.2

With the Required Action and associated Completion Time not met in MODE 1, 2, or 3, the unit must be placed in a MODE in which manual trip is not required. Required Action B.1 and Required Action B.2 place the unit in at least MODE 3 with all CRD trip breakers open within 12 hours. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging unit systems.

C.1

With the Required Action and associated Completion Time not met in MODE 4 or 5, the unit must be placed in a MODE in which manual trip is not required. To achieve this status, all CRD trip breakers must be opened. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to open all CRD trip breakers without challenging unit systems.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.3.2.1

This SR requires the performance of a CHANNEL FUNCTIONAL TEST of the Manual Reactor Trip Function. This test verifies the OPERABILITY of the Manual Reactor Trip by actuation of the CRD trip breakers. The Frequency shall be once prior to each reactor startup if not performed within the preceding 7 days to ensure the OPERABILITY of the Manual Reactor Trip Function prior to achieving criticality. The Frequency was developed in consideration that these Surveillances are only performed during a unit outage.

REFERENCES

1. UFSAR, Chapter 7.

2. 10 CFR 50.36.

## B 3.3 INSTRUMENTATION

### B 3.3.3 Reactor Protective System (RPS) – Reactor Trip Module (RTM)

BASES

BACKGROUND

The RPS consists of four independent protection channels, each containing an RTM. Figure 7.1, UFSAR, Chapter 7 (Ref. 1), shows a typical RPS protection channel and the relationship of the RTM to the RPS instrumentation, manual trip, and CONTROL ROD drive (CRD) trip devices. The RTM receives bistable trip signals from the functions in its own channel and channel trip signals from the other three RPS – RTMs. The RTM provides these signals to its own two-out-of-four trip logic and transmits its own channel trip signal to the two-out-of-four logic of the RTMs in the other three RPS channels. Whenever any two RPS channels transmit channel trip signals, the RTM logic in each channel actuates to remove 120 VAC power from its associated CRD trip devices.

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

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

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

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

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

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

BASES

BACKGROUND (continued)

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

APPLICABLE SAFETY ANALYSES

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

The RTMs satisfy Criterion 3 of 10 CFR 50.36 (Ref. 2).

LCO

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

An OPERABLE RTM must be able to receive and interpret trip signals from its own and other OPERABLE RPS channels and to open its associated trip device.

The requirement of four RTMs to be OPERABLE ensures that a minimum of two RTMs will remain OPERABLE if a single failure has occurred in one RTM and if a second RTM is out of service. This two-out-of-four trip logic also ensures that a single RTM failure will not cause an unwanted reactor trip. Violation of this LCO could result in a trip signal not causing a reactor trip when needed.

APPLICABILITY

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

BASES (continued)

ACTIONS

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

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

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

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

Condition B applies if two or more RTMs are inoperable or if the Required Action and associated Completion Time of Condition A are not met in MODE 1, 2, or 3. In this case, the unit must be placed in a MODE in which the LCO does not apply. This is done by placing the unit in at least MODE 3 with all CRD trip breakers open or with power from all CRD trip breakers removed within 12 hours. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging unit systems.

C.1 and C.2

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

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.3.3.1

The SRs include performance of a CHANNEL FUNCTIONAL TEST every 31 days. This test shall verify the OPERABILITY of the RTM and its ability to receive and properly respond to channel trip and reactor trip signals.

The Frequency of 31 days is based on operating experience, which has demonstrated that failure of more than one channel of a given function in any 31 day interval is a rare event.

Testing in accordance with this SR is normally performed on a rotational basis, with one RTM being tested each week. Testing one RTM each week reduces the likelihood of the same systematic test errors being introduced into each redundant RTM.

REFERENCES

1. UFSAR, Chapter 7.

2. 10 CFR 50.36.

## B 3.3 INSTRUMENTATION

### B 3.3.4 Control Rod Drive (CRD) Trip Devices

BASES

BACKGROUND

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

Figure 7.1, UFSAR, Chapter 7 (Ref. 1), illustrates the configuration of CRD trip devices. To trip the reactor, power to the CRDs must be removed. Loss of power causes the CRD mechanisms to release the CONTROL RODS, which then fall by gravity into the core.

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

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

In addition to the DC power supplies, the redundant buses also supply power to the regulating rod, APSR and auxiliary power supplies. These power supplies contain silicon controlled rectifiers (SCRs) that are gated on and off to provide power to, and remove power from, the phases of the CRD mechanisms. The gating control signal for these SCRs is supplied through the closed contacts of the ETA relays. These contacts are referred to as E and F contactors, and are controlled by the C and D RPS channels respectively.

BASES

BACKGROUND (continued)

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

a. If the A AC circuit breaker opens:

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

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

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

1. the output of the DC power supply is lost, and

2. when the F contactor opens, SCR gating power is lost.

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

In summary, two tripped RPS channels will cause a reactor trip. For example, a reactor trip occurs if RPS channel B senses a low Reactor Coolant System (RCS) pressure condition and if RPS channel C senses a variable low RCS pressure condition. When the channel B bistable relay de-energizes, the channel trip relay de-energizes and opens its associated contacts. The same thing occurs in channel C, except the variable lower pressure bistable relay de-energizes the channel C trip relay. When the output logic relays in channel B and C de-energize, the B and C contacts in the trip logic of each channel's reactor trip module (RTM) open causing an undervoltage to each trip breaker. All trip breakers and the ETA relay contactors open, and power is removed from all CRD mechanisms. All rods fall into the core, resulting in a reactor trip.

APPLICABLE SAFETY ANALYSES

Accident analyses rely on a reactor trip for protection of reactor core integrity, reactor coolant pressure boundary integrity, and reactor building OPERABILITY. A reactor trip must occur when needed to prevent accident consequences from exceeding those calculated in the accident analyses. The CONTROL ROD position limits ensure that adequate rod worth is

BASES

APPLICABLE SAFETY ANALYSES (continued)

available upon reactor trip to shut down the reactor to the required SDM. Further, OPERABILITY of the CRD trip devices ensures that all CONTROL RODS will trip when required. More detailed descriptions of the applicable accident analyses are found in the Bases for each of the individual RPS trip Functions in LCO 3.3.1, "Reactor Protective System (RPS) Instrumentation."

The CRD trip devices satisfy Criterion 3 of CFR 50.36 (Ref. 2).

LCO

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

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

Both ETA relays associated with each of the three regulating rod groups and the two ETA relays associated with the auxiliary power supply must be OPERABLE to satisfy the LCO. The ETA relays associated with the APSR power supply are not required to be OPERABLE because the APSRs are not designed to fall into the core upon initiation of a reactor trip.

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

APPLICABILITY

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

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

BASES (continued)

ACTIONS

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

A.1 and A.2

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

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

If one of the diverse trip Functions on a CRD trip breaker or breaker pair becomes inoperable, actions must be taken to preclude the inoperable CRD trip device from preventing a reactor trip when needed. This is done by manually tripping the inoperable CRD trip breaker or by removing power from the inoperable CRD trip breaker. Either of these actions places the affected CRDs in a one-out-of-two trip configuration, which precludes a single failure from preventing a reactor trip. The 48 hour Completion Time has been shown to be acceptable through operating experience.

B.1 and B.2

Condition B represents a loss of redundancy for the CRD trip Function. Condition B applies when both diverse trip Functions are inoperable in one or more trip breaker(s) or breaker pairs.

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

C.1 and C.2

Condition C represents a loss of redundancy for the CRD trip Function. Condition C applies when one or more ETA relays are inoperable. The preferred action is to restore the ETA relay to OPERABLE status. If this cannot be done, the operator can perform one of two actions to eliminate

BASES

ACTIONS

C.1 and C.2 (continued)

reliance on the failed ETA relay. This first option is to switch the affected CONTROL ROD group to an alternate power supply. This removes the failed ETA relay from the trip sequence, and the unit can operate indefinitely. The second option is to trip the corresponding AC CRD trip breaker. This results in the safety function being performed, thereby eliminating the failed ETA relay from the trip sequence. The 1 hour Completion Time is sufficient to perform the Required Action.

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

With the Required Action and associated Completion Time of Condition A, B, or C not met in MODE 1, 2, or 3, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to MODE 3, with all CRD trip breakers open or with power from all CRD trip breakers removed within 12 hours. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging unit systems.

E.1 and E.2

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

SURVEILLANCE REQUIREMENTS

SR 3.3.4.1

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

BASES (continued)

REFERENCES

1. UFSAR, Chapter 7.

2. 10 CFR 50.36.

## B 3.3 INSTRUMENTATION

### B 3.3.5 Engineered Safeguards Protective System (ESPS) Analog Instrumentation

BASES

BACKGROUND

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

ESPS actuates the following systems:

- High pressure injection (HPI);
- Low pressure injection (LPI);
- Reactor building (RB) cooling;
- Penetration room ventilation;
- RB Spray;
- RB Isolation; and
- Keowee Hydro Unit Emergency Start.

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

Four Parameters are used for actuation:

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

BASES

BACKGROUND (continued)

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

The ESPS contains three analog channels. Each analog channel provides input to digital logic channels that initiate equipment with a two-out-of-three logic on each digital logic channel. Each analog channel includes inputs from one analog instrumentation channel of Low RCS Pressure, Low Low RCS Pressure, High RB Pressure, and High High RB Pressure. Digital automatic actuation logic channels combine the three analog channel trips to actuate the individual Engineered Safeguards (ES) components needed to initiate each ES System. Figure 7.5, UFSAR, Chapter 7 (Ref. 1), illustrates how analog instrumentation channel trips combine to cause digital logic channel trips.

The following matrix identifies the analog instrumentation (measurement) channels and the Digital Automatic Actuation Logic Channels actuated by each.

| Digital Logic Channels | Actuated Systems/ Functions | RCS PRESS LOW | RCS PRESS LOW LOW | RB PRESS HIGH | RB PRESS HIGH HIGH |
|---|---|---|---|---|---|
| 1 and 2 | HPI and RB Non-Essential Isolation, Keowee Emergency Start, Load Shed and Standby Breaker Input, and Keowee Standby Bus Feeder Breaker Input | X | | X | |
| 3 and 4 | LPI and RB Essential isolation | | X | X | |
| 5 and 6 | RB Cooling, RB Essential isolation, and Penetration Room Vent. | | | X | |
| 7 and 8 | RB Spray | | | | X |

The ES equipment is generally divided between the two redundant digital actuation logic channels. The division of the equipment between the two digital actuation logic channels is based on the equipment redundancy and

BASES

BACKGROUND (continued)

function and is accomplished in such a manner that the failure of one of the digital actuation logic channels and the related safeguards equipment will not inhibit the overall ES Functions. Redundant ES pumps are controlled from separate and independent digital actuation logic channels with the exception of HPI B pump which is actuated by both.

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

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

Engineered Safeguards Protective System Bypasses

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

The ESPS RCS pressure instrumentation channels include permissive bistables that allow manual bypass when reactor pressure is below the point at which the low and low low pressure trips are required to be OPERABLE. Once permissive conditions are sensed, the RCS pressure trips may be manually bypassed. Bypasses are automatically removed when bypass permissive conditions are exceeded. This bypass provides an operational provision only outside the Applicability for this parameter, and provides no safety function.

Reactor Coolant System Pressure

The RCS pressure is monitored by three independent pressure transmitters located in the RB. These transmitters are separate from the transmitters that feed the Reactor Protective System (RPS). Each of the pressure signals generated by these transmitters is monitored by four bistables to provide two trip signals, at ≥ 1590 psig and ≥ 500 psig, and two bypass permissive signals, at ≤ 1750 psig and ≤ 900 psig.

BASES

BACKGROUND

Reactor Coolant System Pressure (continued)

The outputs of the three bistables, associated with the low RCS pressure, ≥ 1590 psig, trip drive relays in two sets of identical and independent channels. These two sets of HPI channels each use a two-out-of-three coincidence network for HPI Actuation. The outputs of the three bistables associated with the Low Low RCS Pressure 500°psig trip drive relays in two sets of identical and independent channels. These two sets of LPI channels each use a two-out-of-three coincidence networks for LPI Actuation. The outputs of the three Low Low RCS Pressure bistables also trip the drive relays in the corresponding HPI Actuation channel as previously described.

Reactor Building Pressure

There are three Reactor Building pressure sensors. The output of each sensor terminates in an input isolation amplifier, which provides individually isolated outputs. One isolated output of each pressure measurement goes to the unit computer for monitoring. One output of each pressure measurement goes to a bistable which initiates action when its high building pressure trip point is exceeded. Each input isolation amplifier module contains an analog meter for indicating the measured pressure. Each of the three bistables has contact outputs that are combined in series with the output of the High and Low Pressure Injection System bistables as previously described.

The outputs of the three bistables are brought together in two identical two-out-of-three coincidence logics which provide two ESPS channels. Either of the two channels is independently capable of initiating the required protective action.

The ESPS channels of the Reactor Building Spray System are formed by two identical two-out-of-three logic networks with the active elements onginating in six Reactor Building pressure sensing pressure switches.

Three independent pressure switches containing normally open contacts from one protective channel's two-out-of-three logic inputs. Three other identical pressure switches from the two-out-of-three logic inputs of the second protective channel. Either of the two protective channels is capable of initiating the required protective action.

BASES

BACKGROUND (continued)

Trip Setpoints and Allowable Values

Trip setpoints are the nominal value at which the bistables are set. Any bistable is considered to be properly adjusted when the "as left" value is within the band for CHANNEL CALIBRATION accuracy.

The trip setpoints used in the bistables are selected such that adequate protection is provided when all sensor and processing time delays are taken into account. To allow for calibration tolerances, instrumentation uncertainties, instrument drift, and severe environment induced errors for those ESPS channels that must function in harsh environments as defined by 10 CFR 50.49 (Ref. 2), the Allowable Values specified in Table 3.3.5–1 in the accompanying LCO are conservatively adjusted with respect to the analytical limits. A detailed description of the methodology used to calculate the trip setpoints and associated uncertainties is provided in the Reference 3. The actual nominal trip setpoint entered into the bistable is more conservative than that specified by the Allowable Value to account for changes in random measurement errors detectable by a CHANNEL FUNCTIONAL TEST. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value.

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

Each channel can be tested online to verify that the setpoint accuracy is within the specified allowance requirements. Once a designated channel is taken out of service for testing, a simulated signal may be injected in place of the field instrument signal.

APPLICABLE SAFETY ANALYSES

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

High Pressure Injection

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

Low Pressure Injection

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

BASES

APPLICABLE SAFETY ANALYSES (continued)

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

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

Penetration Room Ventilation Actuation

The ESPS actuation of the penetration room ventilation system has been assumed for LOCAs. Accident dose calculations have credited penetration room ventilation.

Keowee Hydro Unit Emergency Start

The ESPS initiated Keowee Hydro Unit Emergency Start has been included in the design to ensure that emergency power is available throughout the limiting LOCA scenarios.

The small break LOCA analyses assume a conservative 48 second delay time for the actuation of HPI and LPI in UFSAR, Chapter 15 (Ref. 4). The large break LOCA analyses assume LPI flow starts in 38 seconds while full LPI flow does not occur until 15 seconds later, or 53 seconds total (Ref. 4). This delay time includes allowances for Keowee Hydro Unit starting, Emergency Core Cooling Systems (ECCS) pump starts, and valve openings. Similarly, the RB Cooling, RB Isolation, and RB Spray have been analyzed with delays appropriate for the entire system analyzed.

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

The ESPS channels satisfy Criterion 3 of 10 CFR 50.36 (Ref. 5).

LCO

The LCO requires three analog channels of ESPS instrumentation for each Parameter in Table 3.3.5–1 to be OPERABLE in each ESPS digital automatic actuation logic channel. Failure of any instrument renders the affected analog channel(s) inoperable and reduces the reliability of the affected Functions.

BASES

LCO (continued)

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

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

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

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

Three analog channels of RCS Pressure–Low, RCS Pressure–Low Low, RB Pressure–High and RB Pressure–High High are required OPERABLE. Each analog channel includes a sensor, trip bistable, bypass bistable, bypass relays, and output relays. Failure of a bypass bistable or bypass circuitry, such that an analog channel cannot be bypassed, does not render the analog channel inoperable since the analog channel is still capable of performing its safety function, i.e., this is not a safety related bypass function.

APPLICABILITY

Three analog channels of ESPS instrumentation for each of the following Parameters shall be OPERABLE.

1. Reactor Coolant System Pressure – Low

The RCS Pressure – Low actuation Parameter shall be OPERABLE during operation at or above 1750 psig. This requirement ensures the capability to automatically actuate safety systems and components during conditions indicative of a LOCA or secondary unit overcooling. Below 1750 psig, the low RCS Pressure actuation Parameter can be bypassed to avoid actuation during normal unit cooldowns when safety systems actuations are not required.

BASES

APPLICABILITY 1. Reactor Coolant System Pressure – Low (continued)

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

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

2. Reactor Coolant System Pressure – Low Low

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

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

In MODES 5 and 6, there is adequate time for the operator to evaluate unit conditions and respond by manually starting individual systems, pumps, and other equipment to mitigate the

BASES

APPLICABILITY

2. Reactor Coolant System Pressure – Low Low (continued)

consequences of an abnormal condition or accident. RCS pressure and temperature are very low, and many ES components are administratively controlled or otherwise prevented from actuating to prevent inadvertent overpressurization of unit systems.

3, 4. Reactor Building Pressure – High and Reactor Building Pressure –High High

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

ACTIONS

Required Actions A and B apply to all ESPS analog instrumentation Parameters listed in Table 3.3.5–1.

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

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

BASES

ACTIONS (continued)

A.1

Condition A applies when one analog channel becomes inoperable in one or more Parameters. If one ESPS analog instrument channel is inoperable, placing it in a tripped condition leaves the system in a one-out-of-two condition for actuation. Thus, if another analog channel were to fail, the ESPS instrumentation could still perform its actuation functions. This action is completed when all of the affected output relays are tripped. This can normally be accomplished by tripping the affected bistables.

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

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

Condition B applies when the Required Action and associated Completion Time of Condition A are not met or when one or more parameters have two or more inoperable analog channels. If Condition B applies, 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 12 hours and, for the RCS Pressure–Low Parameter, to < 1750 psig, for the RCS Pressure–Low Low Parameter, to < 900 psig, and for the RB Pressure–High Parameter and RB Pressure–High High Parameter, to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

The ESPS Parameters listed in Table 3.3.5–1 are subject to CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION. The operational bypasses associated with each RCS Pressure ESPS instrumentation channel are also subject to these SRs to ensure OPERABILITY of the ESPS instrumentation channel.

SR 3.3.5.1

Performance of the CHANNEL CHECK every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.5.1 (continued)

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

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

The Frequency, equivalent to every shift, is based on operating experience that demonstrates channel failure is rare. Since the probability of two random failures in redundant channels in any 12 hour period is extremely low, the CHANNEL CHECK minimizes the chance of loss of protective function due to failure of redundant channels. The CHANNEL CHECK supplements less formal, but potentially more frequent, checks of channel operability during normal operational use of the displays associated with the LCO's required channels.

SR 3.3.5.2

A CHANNEL FUNCTIONAL TEST is performed on each required ESPS analog channel to ensure the entire channel, including the bypass function, will perform the intended functions. Any setpoint adjustment shall be consistent with the assumptions of the current unit specific setpoint analysis.

The Frequency of 31 days is based on operating experience, with regard to channel OPERABILITY and drift, which demonstrates that failure of more than one channel of a given function in any 31 day interval is a rare event.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.5.3

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

This Frequency is justified by the assumption of an 18 month calibration interval to determine the magnitude of equipment drift in the setpoint analysis.

REFERENCES

1. UFSAR, Chapter 7.
2. 10 CFR 50.49.
3. EDM-102, "Instrument Setpoint/Uncertainty Calculations."
4. UFSAR, Chapter 15.
5. 10 CFR 50.36.

## B 3.3 INSTRUMENTATION

### B 3.3.6 Engineered Safeguards Protective System (ESPS) Manual Initiation

BASES

BACKGROUND

The ESPS manual initiation capability allows the operator to actuate ESPS Functions from the main control room in the absence of any other initiation condition. This ESPS manual initiation capability is provided in the event the operator determines that an ESPS Function is needed and has not been automatically actuated. Furthermore, the ESPS manual initiation capability allows operators to rapidly initiate Engineered Safeguards (ES) Functions.

LCO 3.3.6 covers only the system level manual initiation of these Functions. LCO 3.3.5, "Engineered Safeguards Protective System (ESPS) Analog Instrumentation," and LCO 3.3.7, "Engineered Safeguards Protective System (ESPS) Digital Automatic Actuation Logic Channels," provide requirements on the portions of the ESPS that automatically initiate the Functions described earlier.

The ESPS manual initiation Function relies on the OPERABILITY of the digital automatic actuation logic channels (LCO 3.3.7) to perform the actuation of the systems. A manual trip push button is provided on the control room console for each of the digital automatic actuation logic channels. Operation of the push button energizes relays whose contacts perform a logical "OR" function with the automatic actuation.

The ESPS manual initiation channel is defined as the instrumentation between the console switch and the digital automatic actuation logic channel, which actuates the end devices. Other means of manual initiation, such as controls for individual ES devices, may be available in the control room and other unit locations. These alternative means are not required by this LCO, nor may they be credited to fulfill the requirements of this LCO.

APPLICABLE SAFETY ANALYSES

The ESPS, in conjunction with the actuated equipment, provides protective functions necessary to mitigate accidents, specifically, the loss of coolant accident and steam line break events.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The ESPS manual initiation ensures that the control room operator can rapidly initiate ES Functions. The manual initiation trip Function is required as a backup to automatic trip functions and allows operators to initiate ESPS whenever any parameter is rapidly trending toward its trip setpoint.

The ESPS manual initiation functions satisfy Criterion 3 of 10 CFR 50.36 (Ref. 1).

LCO

Two ESPS manual initiation channels of each ESPS Function shall be OPERABLE whenever conditions exist that could require ES protection of the reactor or RB. Two OPERABLE channels ensure that no single random failure will prevent system level manual initiation of any ESPS Function. The ESPS manual initiation Function allows the operator to initiate protective action prior to automatic initiation or in the event the automatic initiation does not occur.

The required Function is provided by two associated channels as indicated in the following table:

| Function | Associated Channels |
|---|---|
| HPI and RB Non-Essential Isolation, Keowee Emergency Start, Load Shed and Standby Breaker Input, and Keowee Standby Bus Feeder Breaker Input | 1 & 2 |
| LPI and RB Essential isolation | 3 & 4 |
| RB Cooling, RB Essential isolation, and Penetration Room Vent. | 5 & 6 |
| RB Spray | 7 & 8 |

APPLICABILITY

The ESPS manual initiation Functions shall be OPERABLE in MODES 1 and 2, and in MODES 3 and 4 when the associated engineered safeguard equipment is required to be OPERABLE. The manual initiation channels are required because ES Functions are designed to provide protection in these MODES. ESPS initiates systems that are either reconfigured for decay heat removal operation or disabled while in MODES 5 and 6. Accidents in these MODES are slow to develop and would be mitigated by manual operation of individual components. Adequate time is available to evaluate unit conditions and to respond by manually operating the ES components, if required.

BASES (continued)

ACTIONS

A Note has been added to the ACTIONS indicating separate Condition entry is allowed for each ESPS manual initiation Function.

A.1

Condition A applies when one manual initiation channel of one or more ESPS Functions becomes inoperable. Required Action A.1 must be taken to restore the channel to OPERABLE status within the next 72 hours. The Completion Time of 72 hours is based on operating experience and administrative controls, which provide alternative means of ESPS Function initiation via individual component controls. The 72 hour Completion Time is generally consistent with the allowed outage time for the safety systems actuated by ESPS.

B.1 and B.2

With the Required Action and associated Completion Time 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 12 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required MODES from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.3.6.1

This SR requires the performance of a CHANNEL FUNCTIONAL TEST of the ESPS manual initiation. This test verifies that the initiating circuitry is OPERABLE and will actuate the automatic actuation logic channels. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. This Frequency is demonstrated to be sufficient, based on operating experience, which shows these components usually pass the Surveillance when performed on the 18 month Frequency.

REFERENCES

1. 10 CFR 50.36.

## B 3.3 INSTRUMENTATION

### B 3.3.7 Engineered Safeguards Protective System (ESPS) Digital Automatic Actuation Logic Channels

BASES

BACKGROUND

The digital automatic actuation logic channels of ESPS are defined as the instrumentation from the buffers of the ESPS analog instrument channels through the unit controllers that actuate ESPS equipment. Each of the components actuated by the ESPS Functions is associated with one or more digital automatic actuation logic channels. If two-out-of-three ESPS analog instrumentation channels indicate a trip, or if channel level manual initiation occurs, the digital automatic actuation logic channel is activated and the associated equipment is actuated. The purpose of requiring OPERABILITY of the ESPS digital automatic actuation logic channels is to ensure that the Functions of the ESPS can be automatically initiated in the event of an accident. Automatic actuation of some Functions is necessary to prevent the unit from exceeding the Emergency Core Cooling Systems (ECCS) limits in 10 CFR 50.46 (Ref. 1). It should be noted that OPERABLE digital automatic actuation logic channels alone will not ensure that each Function can be activated; the analog instrumentation channels and actuated equipment associated with each Function must also be OPERABLE to ensure that the Functions can be automatically initiated during an accident.

LCO 3.3.7 covers only the digital automatic actuation logic channels that initiates these Functions. LCO 3.3.5, "Engineered Safeguards Protective System (ESPS) Analog Instrumentation," and LCO 3.3.6, "Engineered Safeguards Protective System (ESPS) Manual Initiation," provide requirements on the analog instrumentation and manual initiation channels that input to the digital automatic actuation logic channels.

The ESPS, in conjunction with the actuated equipment, provides protective functions necessary to mitigate accidents, specifically, the loss of coolant accident (LOCA) and main steam line break (MSLB) events. The ESPS relies on the OPERABILITY of the automatic actuation logic for each component to perform the actuation of the selected systems.

The small break LOCA analyses assume a conservative 48 second delay time for the actuation of high pressure injection (HPI) and low pressure injection (LPI) in UFSAR, Chapter 15 (Ref. 2). The large break LOCA analyses assume LPI flow starts in 38 seconds while full LPI flow does not occur until 15 seconds later, or 53 seconds total (Ref. 2). This delay time

BASES

BACKGROUND (continued)

includes allowances for Keowee Hydro Unit startup and loading, ECCS pump starts, and valve openings. Similarly, the reactor building (RB) Cooling, RB Isolation, and RB Spray have been analyzed with delays appropriate for the entire system.

The ESPS automatic initiation of Engineered Safeguards (ES) Functions to mitigate accident conditions is assumed in the accident analysis and is required to ensure that consequences of analyzed events do not exceed the accident analysis predictions. Automatically actuated features include HPI, LPI, RB Cooling, RB Spray, and RB Isolation.

APPLICABLE SAFETY ANALYSES

Accident analyses rely on automatic ESPS actuation for protection of the core and RB and for limiting off site dose levels following an accident. The digital automatic actuation logic is an integral part of the ESPS.

The ESPS digital automatic actuation logic channels satisfy Criterion 3 of 10 CFR 50.36 (Ref. 3).

LCO

The digital automatic actuation logic channels are required to be OPERABLE whenever conditions exist that could require ES protection of the reactor or the RB. This ensures automatic initiation of the ES required to mitigate the consequences of accidents.

The required Function is provided by two associated digital channels as indicated in the following table:

| Function | Associated Channels |
|---|---|
| HPI and RB Non-Essential Isolation, Keowee Emergency Start, Load Shed and Standby Breaker Input, and Keowee Standby Bus Feeder Breaker Input | 1 & 2 |
| LPI and RB Essential isolation | 3 & 4 |
| RB Cooling, RB Essential isolation, and Penetration Room Vent. | 5 & 6 |
| RB Spray | 7 & 8 |

BASES (continued)

APPLICABILITY

The digital automatic actuation logic channels shall be OPERABLE in MODES 1 and 2 and in MODES 3 and 4 when the associated engineered safeguard equipment is required to be OPERABLE, because ES Functions are designed to provide protection in these MODES. Automatic actuation in MODE 5 or 6 is not required because the systems initiated by the ESPS are either reconfigured for decay heat removal operation or disabled. Accidents in these MODES are slow to develop and would be mitigated by manual operation of individual components. Adequate time is available to evaluate unit conditions and respond by manually operating the ES components, if required.

ACTIONS

A Note has been added to the ACTIONS indicating separate Condition entry is allowed for each ESPS digital automatic actuation logic channel.

A.1 and A.2

When one or more digital automatic actuation logic channels are inoperable, the associated component(s) can be placed in their engineered safeguard configuration. Required Action A.1 is equivalent to the digital automatic actuation logic channel performing its safety function ahead of time

In some cases, placing the component in its engineered safeguard configuration would violate unit safety or operational considerations. In these cases, the component status should not be changed, but the supported system component must be declared inoperable. Conditions which would preclude the placing of a component in its engineered safeguard configuration include, but are not limited to, violation of system separation, activation of fluid systems that could lead to thermal shock, or isolation of fluid systems that are normally functioning. The Completion Time of 1 hour is based on operating experience and reflects the urgency associated with the inoperability of a safety system component.

Required Action A.2 requires declaring the associated components of the affected supported systems inoperable, since the true effect of digital automatic actuation logic channel failure is inoperability of the supported system. The Completion Time of 1 hour is based on operating experience and reflects the urgency associated with the inoperability of a safety system component. A combination of Required Actions A.1 and A.2 may be used for different components associated with an inoperable digital automatic actuation logic channel.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.3.7.1

SR 3.3.7.1 is the performance of a CHANNEL FUNCTIONAL TEST on a 31 day Frequency. The test demonstrates that each digital automatic actuation logic channel successfully performs the two-out-of-three logic combinations every 31 days. The test simulates the required one-out-of-three inputs to the logic circuit and verifies the successful operation of the automatic actuation logic. The Frequency is based on operating experience that demonstrates the rarity of more than one channel failing within the same 31 day interval.

REFERENCES

1. 10 CFR 50.46.
2. UFSAR, Chapter 15.
3. 10 CFR 50.36.

## B 3.3 INSTRUMENTATION

### B 3.3.8 Post Accident Monitoring (PAM) Instrumentation

BASES

BACKGROUND

The primary purpose of the PAM instrumentation is to display unit variables that provide information required by the control room operators during accident situations. This information provides the necessary support for the operator to take the manual actions for which no automatic control is provided and that are required for safety systems to accomplish their safety functions for Design Basis Events.

The OPERABILITY of the accident monitoring instrumentation ensures that there is sufficient information available on selected unit parameters to monitor and to assess unit status and behavior following an accident.

The availability of accident monitoring instrumentation is important so that responses to corrective actions can be observed, and so that the need for and magnitude of further actions can be determined. These essential instruments are identified by the ONS specific Regulatory Guide 1.97 analysis (Ref. 1), UFSAR, Section 7.5 (Ref. 2), and the NRC's Safety Evaluation Report for the ONS Regulatory Guide 1.97 analysis (Ref. 3) which address the recommendations of Regulatory Guide 1.97 (Ref. 4), as required by Supplement 1 to NUREG-0737 (Ref. 5).

The instrument channels required to be OPERABLE by this LCO equate to two classes of parameters identified during unit specific implementation of Regulatory Guide 1.97 as Type A and Category 1 variables.

Type A variables are specified because they provide the primary information that permits the control room operator to take specific manually controlled actions that are required when no automatic control is provided and that are required for safety systems to accomplish their safety functions for accidents.

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

- Determine whether systems important to safety are performing their intended functions;

BASES

BACKGROUND (contonued)

- Provide information to the operators that will enable them to determine the potential for causing a gross breach of the barriers to radioactivity release; and

- Provide information regarding the release of radioactive materials to allow for early indication of the need to initiate action necessary to protect the public and to estimate the magnitude of any impending threat.

These key variables are identified by the ONS specific Regulatory Guide 1.97 analysis (Ref. 1). This analysis identifies the unit specific Type A and Category 1 variables and provides justification for deviating from the NRC proposed list of Category 1 variables.

The specific instrument Functions listed in Table 3.3.8-1 are discussed in the LCO Bases Section.

APPLICABLE SAFETY ANALYSES

The PAM instrumentation ensures the availability of information so that the control room operating staff can:

- Perform the diagnosis specified in the emergency operating procedures. These variables are restricted to preplanned actions for the primary success path of accidents (e.g., loss of coolant accident (LOCA));

- Take the specified, preplanned, manually controlled actions, for which no automatic control is provided, which are required for safety systems to accomplish their safety functions;

- Determine whether systems important to safety are performing their intended functions;

- Determine the potential for causing a gross breach of the barriers to radioactivity release;

- Determine if a gross breach of a barrier has occurred; and

- Initiate action necessary to protect the public and estimate the magnitude of any impending threat.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The ONS specific Regulatory Guide 1.97 analysis (Ref. 1) documents the process that identifies Type A and Category 1 non–Type A variables.

PAM instrumentation that meets the definition of Type A in Regulatory Guide 1.97 satisfies Criterion 3 of 10 CFR 50.36 (Ref. 6). Category 1, non-type A, instrumentation must be retained in Technical Specifications because it is intended to assist operators in minimizing the consequences of accidents. Category 1, non-Type A variables are important for reducing public risk, and therefore, satisfy Criterion 4 of 10 CFR 50.36 (Ref. 6).

LCO

LCO 3.3.8 requires two OPERABLE channels for all but one Function to ensure no single failure prevents the operators from being presented with the information necessary to determine the status of the unit and to bring the unit to, and maintain it in, a safe condition following that accident. Furthermore, provision of two channels allows a CHANNEL CHECK during the post accident phase to confirm the validity of displayed information.

Where a channel includes more than one control room indication, such as both an indicator and a recorder, the channel is OPERABLE when at least one indication is OPERABLE.

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

Each of the specified instrument Functions listed in Table 3.3.8-1 are discussed below:

BASES

LCO (continued)

1. Wide Range Neutron Flux

Wide Range Neutron Flux indication is a Type B, Category 1 variable provided to verify reactor shutdown. The Wide Range Neutron Flux channels consist of two channels of fission chamber based instrumentation with readout on one recorder. (Note: four channels are available only two are required). The channels provide indication over a range of 1E-8% to 200% RTP.

2. Reactor Coolant System (RCS) Hot Leg Temperature

RCS Hot Leg Temperature instrumentation is a Type B, Category 1 variable provided for verification of core cooling and long term surveillance. The two channels provide readout on two indicators. Control room display is through the inadequate core cooling monitoring system. The channels provide indication over a range of 50°F to 700°F.

3, 5. Reactor Vessel Head Level and RCS Hot Leg Level

Reactor Vessel Water Level instrumentation is a Type B, Category 1 variable provided for verification and long term surveillance of core cooling. The reactor vessel level monitoring system provides an indication of the liquid level from the top of the Hot Leg on each steam generator to the bottom of the Hot Leg as it exits the vessel and from the top of the reactor vessel head to the bottom of the Hot Leg as it exits the vessel. Compensation is provided for impulse line temperature variations.

The Reactor Vessel Water Level channels consist of two Reactor Vessel Head Level channels that provide readout on two indicators (RC-LT0125 and RC-LT0126) with one channel recorded in the control room and two RCS Hot Leg Level channels that provide readout on two indicators (RC-LT0123 and RC-LT0124) with one channel recorded in the control room.

4. RCS Pressure (Wide Range)

RCS Pressure (Wide Range) instrumentation is a Type A, Category 1 variable provided for verification of core cooling and RCS integrity long term surveillance.

BASES

LCO

4. RCS Pressure (Wide Range) (continued)

Wide range RCS loop pressure is measured by pressure transmitters with a span of 0 psig to 3000 psig. The pressure transmitters are located outside the RB. Redundant monitoring capability is provided by two trains of instrumentation. Control room indications are provided through the inadequate core cooling plasma display. The inadequate core cooling plasma display is the primary indication used by the operator during an accident. Therefore, the accident monitoring specification deals specifically with this portion of the instrument string.

RCS Pressure is a Type A, Category 1 variable because the operator uses this indication to monitor the cooldown of the RCS following a steam generator (SG) tube rupture or small break LOCA. Operator actions to maintain a controlled cooldown, such as adjusting SG pressure or level, would use this indication. In addition, high pressure injection (HPI) flow is throttled based on RCS Pressure and subcooled margin. For some small break LOCAs, low pressure injection (LPI) may actuate with RCS pressure stabilizing above the shutoff head of the LPI pumps. If this condition exists, the operator is instructed to verify HPI flow and then terminate LPI flow prior to exceeding 30 minutes of LPI pump operation against a deadhead pressure. RCS Pressure, in conjunction with LPI flow, is also used to determine if a core flood line break has occurred.

6. Containment Sump Water Level (Wide Range)

Containment Sump Water Level (Wide Range) instrumentation is a Type B, Category 1 variable provided for verification and long term surveillance of RCS integrity. The Containment Sump Water Level instrumentation consists of two channels with readout on two indicators (LT-90 and LT-91) and one recorder.. The indicated range is 0 to 15 feet.

BASES

LCO (continued)

7. Containment Pressure (Wide Range)

Containment Pressure (Wide Range) instrumentation is a Type B, Category 1 variable provided for verification of RCS and containment OPERABILITY. Containment Pressure instrumentation consists of two channels with readout on two indicators (PT-230 and PT-231) and one channel recorded. The indicated range is -5.0 psig to 175 psig.

8. Containment Isolation Valve Position

Containment isolation valve (CIV) position is a Type B, Category 1 variable provided for verification of electrically controlled containment isolation valve position. In the case of CIV position, the important information is the isolation status of the containment penetration. The LCO requires one channel of valve position indication in the control room to be OPERABLE for each electrically controlled CIV in a containment penetration flow path, i.e., two total channels of CIV position indication for a penetration flow path with two electrically controlled valves. For containment penetrations with only one electrically controlled CIV having control room indication, Note (b) requires a single channel of valve position indication to be OPERABLE. This is sufficient to redundantly verify the isolation status of each isolable penetration via indicated status of the electrically controlled valve, as applicable, and prior knowledge of passive valve or system boundary status. As indicated by Note (a) to the Required Channels, if a penetration flow path is isolated by at least one closed and deactivated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured, position indication for the CIV(s) in the associated penetration flow path is not needed to determine status. Therefore, the position indication for valves in an isolated penetration flow path is not required to be OPERABLE. Note (c) to the Required Channels indicates that position indication requirements apply only to CIVs that are electrically controlled. The CIV position PAM instrumentation consists of limit switches that operate both Closed-Not Closed and Open-Not Open control switch indication via indicating lights in the control room.

BASES

LCO (continued)

9. Containment Area Radiation (High Range)

Containment Area Radiation (High Range) instrumentation is a Type C, Category 1 variable provided to monitor the potential for significant radiation releases and to provide release assessment for use by operators in determining the need to invoke site emergency plans. The Containment Area Radiation instrumentation consists of two channels (RIA 57 and 58) with readout on two indicators and one channel recorded. The indicated range is 1 to $10^7$ R/hr.

10. Containment Hydrogen Concentration

Containment Hydrogen Concentration instrumentation is a Type A, Category 1 variable provided to detect high hydrogen concentration conditions that represent a potential for containment breach. This variable is also important in verifying the adequacy of mitigating actions. The Containment Hydrogen Concentration instrumentation consists of two channels (MT 80 and 81) with readout on two indicators and one channel recorded. The indicated range is 0 to 10% hydrogen concentration.

11. Pressurizer Level

Pressurizer Level instrumentation is a Type A, Category 1 variable used in combination with other system parameters to determine whether to terminate safety injection (SI), if still in progress, or to reinitiate SI if it has been stopped. Knowledge of pressurizer water level is also used to verify the unit conditions necessary to establish natural circulation in the RCS and to verify that the unit is maintained in a safe shutdown condition. The Pressurizer Level instrumentation consists of three channels (two for Train A and one for Train B) with two channels indicated and one channel recorded.

(Note: three channels are available only two are required). The indicated range is 0 to 400 inches (11% to 84% level as a percentage of volume).

BASES

LCO (continued)

12. Steam Generator Water Level

Steam Generator Water Level instrumentation is a Type A, Category 1 variable provided to monitor operation of decay heat removal via the SG. The indication of SG level is the extended startup range level instrumentation, covering a span of 0 inches to 388 inches above the lower tubesheet.

The operator relies upon SG level information following an accident (e.g., main steam line break, steam generator tube rupture) to isolate the affected SG to confirm adequate heat sinks for transients and accidents.

The extended startup range Steam Generator Level instrumentation consists of four transmitters (two per SG) that feed four gauges.

13. Steam Generator Pressure

Steam Generator Pressure instrumentation is a Type A, Category 1 variable provided to support operator diagnosis of a main steam line break or SG tube rupture accident to identify and isolate the affected SG. In addition, SG pressure is a key parameter used by the operator to evaluate primary-to-secondary heat transfer.

Steam generator pressure measurement is provided by two pressure transmitters per SG. Each instrument channel inputs to the ICCM cabinet that provide safety inputs to two indicators located on the main control board in the control room. One channel per SG also provides input to a recorder located in the control room.

14. Borated Water Storage Tank (BWST) Level

BWST Level instrumentation is a Type A, Category 1 variable provided to support action for long term cooling requirements, i.e., to determine when to initiate the switch over of the core cooling pump suction from the BWST to sump recirculation. BWST level measurement is provided by three channels with readout on two indicators and one recorder. (Note: three channels are available only two are required). Two of the three channels provide inputs

BASES

LCO

14. Borated Water Storage Tank (BWST) Level (continued)

to the ICCM cabinet which provides inputs to qualified indicators on the Control Board. The third channel provides a safety input to a dedicated recorder. The channels provide level indication over a range of 0 to 50 feet (13% to 100% of volume).

15. Upper Surge Tank (UST) Level

Upper Surge Tank Level instrumentation is a Type A, Category 1 variable provided to ensure a water supply for EFW. EFW draws condensate grade suction from the USTs and the Condenser Hotwell.

Two Category 1 instrumentation channels are provided for monitoring UST level. These instrument channels are inputs to corresponding train A and B Inadequate Core Cooling Monitoring (ICCM) system cabinets. The ICCM Train A cabinet provides UST level input to a dedicated qualified recorder and to a qualified indicator, both located in the Control Room. The ICCM Train B cabinet also provides an input to a qualified indicator located in the Control Room. The range of UST level indication is 0 to 12 feet.

UST Level is the primary indication used by the operator to identify loss of UST volume. The operator can then decide to replenish the UST or align suction to the EFW pumps from the hotwell.

16. Core Exit Temperature

Core Exit Temperature is a Type A, Category 1 variable provided for verification and long term surveillance of core cooling.

The operator relies on this information following a LOCA to secure HPI and throttle LPI, following a SBLOCA to throttle HPI and begin forced HPI cooling if needed, and following a MSLB and SG Tube Rupture to throttle HPI and isolate the affected SG.

BASES

LCO

16. Core Exit Temperature (continued)

There are a total of 52 Core Exit Thermocouples (CETs) per Oconee Unit. Twenty-four (12 per train) meet seismic and environmental qualification requirements (Category 1). The unit computer is the primary display for all 52 CETs. The CETs are distributed to provide monitoring of four or more in each quadrant for each train. The ICCM plasma displays (1 per train) located in the Control Room serve as safety related backup displays for the twenty-four Category 1 CETs. The range of the readouts is 50°F to 2300°F.

The ICCM CET function uses inputs from twelve incore thermocouples per train to calculate and display temperatures of the reactor coolant as it exits the core and to provide indication of thermal conditions across the core at the core exit. Each of the twelve qualified thermocouples per train is displayed on a spatially oriented core map on the plasma display. Trending of CET temperature is available continuously on the plasma display. The average of the five hottest CETs is trendable for the past forty minutes.

An evaluation was made of the minimum number of valid core exit thermocouples (CETs) necessary for inadequate core cooling detection. The evaluation determined the reduced complement of CETs necessary to detect initial core recovery and to trend the ensuing core heatup. The evaluations account for core nonuniformities and cold leg injection. Based on these evaluations, adequate or inadequate core cooling detection is ensured with two sets of five valid CETs.

Table 3.3.8-1 Note (d) indicates that the subcooling margin monitor takes the average of the five highest

CETs for each of the ICCM trains. Two channels ensure that a single failure will not disable the ability to determine the representative core exit temperature.

BASES

LCO (continued)

17. Subcooling Monitor

The Subcooling Monitor is a Type A, Category 1 variable provided for verification and long term surveillance of core cooling. This variable is a computer calculated value using various inputs from the Primary System.

Two channels of indication are provided. An operable Subcooling Monitor shall consist of: 1) One direct indication from one channel for RCS Loop Saturation margin and one direct indication from the other channel for Core Saturation margin, or 2) One direct indication from each of the two channels for RCS Loop Saturation margin. The indication readouts are located in the control room. This variable also inputs to the unit computer through isolation buffers and is available for trend recording upon operator demand. The range of the readouts is 200°F subcooled to 50°F superheat. The control room display is through the ICCM plasma display unit.

A backup method for determining subcooling margin ensures the capability to accurately monitor RCS subcooling margin (Refer to Specification 5.5.17).

18. HPI System Flow

HPI System Flow instrumentation is a Type A, Category 1 variable provided to support action for short term cooling requirements, to prevent HPI pump runout and inadequate NPSH, and to indicate the need for flow cross connect. HPI flow is throttled based on RCS pressure, subcooled margin, and pressurizer level. Flow measurement is provided by one channel per train with readout on an indicator and recorder. There are two HPI trains. The channels provide flow indication over a range of 0 to 750 gpm.

BASES

LCO (continued)

19. LPI System Flow

LPI System Flow instrumentation is a Type A, Category 1 variable provided to support action for long term cooling requirements, to prevent LPI pump runout and for flow balance. The indication is also used to identify an LPI pump operating at system pressures above its shutoff head. Flow measurement is provided by one channel per train with readout on an indicator and recorder. There are two LPI trains. The LPI channels provide flow indication over a range of 0 to 6000 gpm.

20. Reactor Building Spray Flow

Reactor Building Spray Flow instrumentation is a Type A, Category 1 variable provided to support action for long term cooling requirements and iodine removal and to prevent Reactor Building Spray and LPI pump runout. Flow measurement is provided by one channel per train with readout on an indicator and recorder. There are two RBS trains. The channels provide flow indication over a range from 0 to 2000 gpm.

21. Emergency Feedwater Flow

EFW Flow instrumentation is a Type D, Category 1 variable provided to monitor operation of RCS heat removal via the SGs. Two channels provide indication of EFW Flow to each SG over a range of approximately 100 gpm to 1200 gpm. Redundant monitoring capability is provided by the two independent channels of instrumentation for each SG. Each flow transmitter provides an input to a control room indicator. One channel also provides input to a recorder.

EFW Flow is the primary indication used by the operator to verify that the EFW System is delivering the correct flow to each SG. However, the primary indication used by the operator to ensure an adequate inventory is SG level.

BASES

LCO (continued)

22. Low Pressure Service Water (LPSW) flow to LPI Coolers

LPSW flow to LPI Coolers is a Type A, Category 1 variable is provided to prevent LPSW pump runout and inadequate NPSH. LPSW flow to LPI Coolers is throttled to maintain proper flow balance in the LPSW System.

Flow measurement is provided by one channel per train with readout on an indicator and recorder. The channels provide flow indication over a range from 0-8000 gpm.

APPLICABILITY

The PAM instrumentation LCO is applicable in MODES 1, 2, and 3. These variables are related to the diagnosis and preplanned actions required to mitigate accidents and transients. The applicable accidents and transients are assumed to occur in MODES 1, 2, and 3. In MODES 4, 5, and 6, unit conditions are such that the likelihood of an event occurring that would require PAM instrumentation is low; therefore, the PAM instrumentation is not required to be OPERABLE in these MODES.

ACTIONS

The ACTIONS are modified by two Notes. Note 1 is added to the ACTIONS to exclude the MODE change restriction of LCO 3.0.4. This exception allows entry into an applicable MODE while relying on the ACTIONS even though the ACTIONS may eventually require a unit shutdown. This exception is acceptable due to the passive function of the instruments, the operator's ability to respond to an accident utilizing alternate instruments and methods, and the low probability of an event requiring these instruments.

Note 2 is added to the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed in Table 3.3.8-1. When the Required Channels for a function in Table 3.3.8-1 are specified on a "per" basis (e.g., per loop, per SG, per penetration flow path), then the Condition may be entered separately for each loop, SG, penetration flow path, etc., as appropriate. The Completion Time(s) of the inoperable channels of a Function are tracked separately for each Function starting from the time the Condition is entered for that Function.

BASES

ACTIONS (continued)

A.1

When one or more Functions have one required channel inoperable, the inoperable channel must be restored to OPERABLE status within 30 days. The 30 day Completion Time is based on operating experience. This takes into account the remaining OPERABLE channel, 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.

Condition A is modified by a Note indicating this Condition is not applicable to PAM Functions 14, 18, 19, 20, and 22.

B.1

Required Action B.1 specifies initiation of action described in Specification 5.6.6 that requires a written report to be submitted to the NRC. This report discusses the results of the root cause evaluation of the inoperability and identifies proposed restorative actions. This action is appropriate in lieu of a shutdown requirement since alternative actions are identified before loss of functional capability and given the likelihood of unit conditions that would require information provided by this instrumentation. The Completion Time of "Immediately" for Required Action B.1 ensures the requirements of Specification 5.6.6 are initiated.

C.1

When one or more Functions have two required channels inoperable (i.e., two channels inoperable in the same Function), one channel in the Function should be restored to OPERABLE status within 7 days. This Condition does not apply to the hydrogen monitor channels. The Completion Time of 7 days is based on the relatively low probability of an event requiring PAM instrumentation action operation and the availability of alternative means to obtain the required information. Continuous operation with two required channels inoperable in a Function is not acceptable because the alternate indications may not fully meet all performance of qualification requirements applied to the PAM instrumentation. Therefore, requiring restoration of one inoperable channel of the Function limits the risk that the PAM Function will be in a degraded condition should an accident occur.

BASES

ACTIONS

C.1 (continued)

Condition C is modified by a Note indicating this Condition is not applicable to PAM Functions 10, 14, 18, 19, 20, and 22.

D.1

When two required hydrogen monitor channels are inoperable, Required Action D.1 requires one channel to be restored to OPERABLE status. This action restores the monitoring capability of the hydrogen monitor. The 72 hour Completion Time is based on the relatively low probability of an event requiring hydrogen monitoring. Continuous operation with two required channels inoperable is not acceptable because alternate indications are not available.

Condition D is modified by a Note indicating this Condition is only applicable to PAM Function 10.

E.1

When one required BWST water level channel is inoperable, Required Action E.1 requires the channel to be restored to OPERABLE status. The 24 hour Completion Time is based on the relatively low probability of an event requiring BWST water and the availability of the remaining BWST water level channel. Continuous operation with one of the two required channels inoperable is not acceptable because alternate indications are not available. This indication is crucial in determining when the water source for ECCS should be swapped from the BWST to the reactor building sump.

Condition E is modified by a Note indicating this Condition is only applicable to PAM Function 14.

F.1

When a flow instrument channel is inoperable, Required Action F.1 requires the affected HPI, LPI, or RBS train to be declared inoperable and the requirements of LCO 3.5.2, LCO 3.5.3, or LCO 3.6.5 apply. For Function 22, LPSW flow to LPI coolers, the affected train is the

BASES

ACTIONS

F.1 (continued)

associated LPI train. For Function 18, HPI flow, an inoperable flow instrument channel causes the affected HPI train's automatic function to be inoperable. The HPI train continues to be manually OPERABLE provided the HPI discharge crossover valves and associated flow instruments are OPERABLE. Therefore, HPI is in a condition where one HPI train is incapable of being automatically actuated but capable of being manually actuated. The required Completion Time for declaring the train(s) inoperable is immediately. Therefore, LCO 3.5.2, LCO 3.5.3, or LCO 3.6.5 is entered immediately, and the Required Actions in the LCOs apply without delay. This action is necessary since there is no alternate flow indication available and these flow indications are key in ensuring each train is capable of performing its function following an accident. HPI, LPI, and RBS train OPERABILITY assumes that the associated PAM flow instrument is OPERABLE because this indication is used to throttle flow during an accident and assure runout limits are not exceeded or to ensure the associated pumps do not exceed NPSH requirements.

Condition F is modified by a Note indicating this Condition is only applicable to PAM Functions 18, 19, 20, and 22.

G.1

Required Action G.1 directs entry into the appropriate Condition referenced in Table 3.3.8-1. The applicable Condition referenced in the Table is Function dependent. Each time an inoperable channel has not met the Required Action and associated Completion Time of Condition C, D, or E, as applicable, Condition G is entered for that channel and provides for transfer to the appropriate subsequent Condition.

H.1 and H.2

If the Required Action and associated Completion Time of Conditions C, D or E are not met and Table 3.3.8-1 directs entry into Condition H, the unit must be brought to a MODE in which the requirements of this LCO do not apply. To achieve this status, the unit must be brought to at least MODE 3 within 12 hours and MODE 4 within 18 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

ACTIONS (continued)

I.1

If the Required Action and associated Completion Time of Condition C, D or E are not met and Table 3.3.8-1 directs entry into Condition I, alternate means of monitoring the parameter should be applied and the Required Action is not to shut down the unit, but rather to follow the directions of Specification 5.6.6 in the Administrative Controls section of the Technical Specifications. These alternative means may be temporarily installed if the normal PAM channel cannot be restored to OPERABLE status within the allowed time. The report provided to the NRC should discuss the alternative means used, describe the degree to which the alternative means are equivalent to the installed PAM channels, justify the areas in which they are not equivalent, and provide a schedule for restoring the normal PAM channels.

Both the RCS Hot Leg Level and the Reactor Vessel Level are methods of monitoring for inadequate core cooling capability. The subcooled margin monitors (SMM), and core-exit thermocouples (CET) provide an alternate means of monitoring for this purpose. The function of the ICC instrumentation is to increase the ability of the unit operators to diagnose the approach to and recovery from ICC. Additionally, they aid in tracking reactor coolant inventory.

The alternate means of monitoring the Reactor Building Area Radiation (High Range) consist of a combination of installed area radiation monitors and portable instrumentation.

SURVEILLANCE REQUIREMENTS

As noted at the beginning of the SRs, the SRs apply to each PAM instrumentation Function in Table 3.3.8-1 except where indicated.

SR 3.3.8.1

Performance of the CHANNEL CHECK once every 31 days for each required instrumentation channel that is normally energized ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel with a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.8.1 (continued)

instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; therefore, it is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION. The high radiation instrumentation should be compared with similar unit instruments located throughout the unit. If the radiation monitor uses keep alive sources or check sources OPERABLE from the control room, the CHANNEL CHECK should also note the detector's response to these sources.

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

The Frequency is based on operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal but more frequent checks of channels during normal operational use of the displays associated with this LCO's required channels.

SR 3.3.8.2 and SR 3.3.8.3

A CHANNEL CALIBRATION is a complete check of the instrument channel, including the sensor. This test verifies the channel responds to measured parameters within the necessary range and accuracy.

Note 1 to SR 3.3.8.3 clarifies that the neutron detectors are not required to be tested as part of the CHANNEL CALIBRATION. There is no adjustment that can be made to the detectors. Furthermore, adjustment of the detectors is unnecessary because they are passive devices, with minimal drift. Slow changes in detector sensitivity are compensated for by performing the daily calorimetric calibration and the monthly axial channel calibration.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.8.2 and SR 3.3.8.3 (continued)

For the Containment Area Radiation instrumentation, a CHANNEL CALIBRATION may consist of an electronic calibration of the channel, not including the detector, for range decades above 10 R/hr, and a one point calibration check of the detector below 10 R/hr with a gamma source.

Whenever a sensing element is replaced, the next required CHANNEL CALIBRATION of the resistance temperature detectors (RTD)sensors or Core Exit thermocouple sensors is accomplished by an inplace cross calibration that compares the other sensing elements with the recently installed sensing element.

SR 3.3.8.2 is modified by a Note indicating that it is applicable only to Functions 7, 10 and 22. SR 3.3.8.3 is modified by Note 2 indicating that it is not applicable to Functions 7, 10 and 22. The Frequency of each SR is based on operating experience and is justified by the assumption of the specified calibration interval in the determination of the magnitude of equipment drift.

REFERENCES

1. Duke Power Company letter from Hal B. Tucker to Harold M. Denton (NRC) dated September 28, 1984.

2. UFSAR, Section 7.5.

3. NRC Letter from Helen N. Pastis to H. B. Tucker, "Emergency Response Capability - Conformance to Regulatory Guide 1.97," dated March 15, 1988.

4. Regulatory Guide 1.97, "Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," Revision 3, May 1983.

5. NUREG-0737, "Clarification of TMI Action Plan Requirements," 1980.

6. 10 CFR 50.36.

## B 3.3 INSTRUMENTATION

### B 3.3.9 Source Range Neutron Flux

BASES

BACKGROUND

The source range neutron flux channels provide the operator with an indication of the approach to criticality at lower power levels than can be seen on the wide range neutron flux instrumentation. These channels also provide the operator with a flux indication that reveals changes in reactivity and helps to verify that SDM is being maintained.

The source range instrumentation has four redundant count rate channels originating in four fission chambers. Four source range detectors are externally located symmetrically around the core. These channels are used over a counting range of 0.1 cps to 1E5 cps and are displayed on the operator's control console in terms of log count rate. The channels also measure the rate of change of the neutron flux level, which is displayed for the operator in terms of startup rate from -0.1 decades to +7 decades per minute. An interlock provides a control rod withdraw "inhibit" on a high startup rate of +2 decades per minute in either channel.

APPLICABLE SAFETY ANALYSES

The source range neutron flux channels are necessary to monitor core reactivity changes. They are the primary means for detecting reactivity changes and triggering operator actions to respond to reactivity transients initiated from conditions in which the Reactor Protection System (RPS) is not required to be OPERABLE. They also trigger operator actions to anticipate RPS actuation in the event of reactivity transients starting from shutdown or low power conditions.

The source range neutron flux channels satisfy Criterion 2 of 10 CFR 50.36 (Ref. 1).

LCO

Two source range neutron flux channels shall be OPERABLE to provide the operator with redundant source range neutron instrumentation. The source range instrumentation provides the primary power indication at low power levels < 4E-4% RTP on wide range instrumentation and must remain OPERABLE for the operator to continue increasing power.

BASES (continued)

APPLICABILITY

Two source range neutron flux channels shall be OPERABLE in MODE 2 to provide redundant indication during an approach to criticality. Neutron flux level is sufficient for monitoring on the wide range and on the power range instrumentation prior to entering MODE 1; therefore, source range instrumentation is not required in MODE 1.

In MODES 3, 4, and 5, source range neutron flux instrumentation shall be OPERABLE to provide the operator with a means of monitoring neutron flux and to provide an early indication of reactivity changes.

The requirements for source range neutron flux instrumentation during MODE 6 refueling operations are addressed in LCO 3.9.2, "Nuclear Instrumentation."

ACTIONS

A.1

The Required Action for one required channel of the source range neutron flux indication inoperable with THERMAL POWER ≤ 4E-4% RTP on the wide range neutron flux instrumentation is to delay increasing reactor power until the channel is repaired and restored to OPERABLE status. This limits power increases in the range where the operators rely solely on the source range instrumentation for power indication. The Completion Time ensures the source range is available prior to further power increases. Furthermore, it ensures that power remains below the point where the wide range channels provide primary protection.

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

With both required source range neutron flux channels inoperable with THERMAL POWER ≤ 4E-4% RTP on the wide range neutron flux instrumentation, the operators must take actions to limit the possibilities for adding positive reactivity. This is done by immediately suspending positive reactivity additions, initiating action to insert all CONTROL RODS, and opening the control rod drive trip breakers within 1 hour. Periodic SDM verification is then required to provide a means for detecting the slow reactivity changes that could be caused by mechanisms other than CONTROL ROD withdrawal or operations involving positive reactivity changes. Since the source range instrumentation provides the only reliable direct indication of power in this condition, the operators must continue to verify the SDM every 12 hours until at least one channel of the source range instrumentation is returned to OPERABLE status. Required

BASES

ACTIONS

B.1, B.2, B.3, and B.4 (continued)

Action B.1, Required Action B.2, and Required Action B.3 preclude rapid positive reactivity additions. The 1 hour Completion Time for Required Action B.3 and Required Action B.4 provides sufficient time for operators to accomplish the actions. The 12 hour Frequency for performing the SDM verification provides reasonable assurance that the reactivity changes possible with CONTROL RODS inserted are detected before SDM limits are challenged.

C.1

With reactor power > 4E-4% RTP in MODE 2, 3, 4, or 5 on the wide range neutron flux instrumentation, continued operation is allowed with one or more required source range neutron flux channels inoperable. The ability to continue operation is justified because the instrumentation does not provide a safety function during high power operation. However, actions are initiated within 1 hour to restore the channel(s) to OPERABLE status for future availability. The Completion Time of 1 hour is sufficient to initiate the action. The action must continue until channels are restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

SR 3.3.9.1

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

Agreement criteria are determined, based on a combination of the channel instrument uncertainties, including isolation, indication, and readability. If a channel is outside the criteria, it may be an indication that the signal processing equipment has drifted outside its limit. If the channels are within the criteria, it is an indication that the channels are OPERABLE. If

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.9.1 (continued)

the channels are normally off scale during times when surveillance is required, the CHANNEL CHECK will only verify that they are off scale in the same direction.

The Frequency, equivalent to every shift, is based on operating experience that demonstrates channel failure is rare. Since the probability of two random failures in redundant channels in any 12 hour period is extremely low, the CHANNEL CHECK minimizes the chance of loss of protective function due to failure of redundant channels. The CHANNEL CHECK supplements less formal, but potentially more frequent, checks of channel OPERABILITY during normal operational use of the displays associated with the LCO's required channels. When operating in Required Action A.1, CHANNEL CHECK is still required. However, in this condition, a redundant source range may not be available for comparison. CHANNEL CHECK may still be performed via comparison with wide range detectors, if available, and verification that the OPERABLE source range channel is energized and indicating a value consistent with current unit status.

SR 3.3.9.2

For source range neutron flux channels, CHANNEL CALIBRATION is a complete check and readjustment of the channels from the preamplifier input to the indicators. This test verifies the channel responds to measured parameters within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift to ensure that the instrument channel remains operational between successive tests.

The SR is modified by a Note excluding neutron detectors from CHANNEL CALIBRATION. It is not necessary to test the detectors because generating a meaningful test signal is difficult. The detectors are of simple construction, and any failures in the detectors will be apparent as change in channel output.

The Frequency of 18 months is based on demonstrated instrument CHANNEL CALIBRATION reliability over an 18 month interval, such that the instrument is not adversely affected by drift.

REFERENCES

1. 10 CFR 50.36.

## B 3.3 INSTRUMENTATION

### B 3.3.10 Wide Range Neutron Flux

BASES

| | |
|---|---|
| BACKGROUND | The wide range neutron flux channels provide the operator with an indication of reactor power from 1E-8 to 200% of RTP and fully overlap the source and power range channels providing continuity of information needed during startup.<br><br>The wide range instrumentation has four log N channels originating in four electrically identical fission chambers. Each channel provides ten decades of flux level information in terms of the log of chamber count rate and startup rate. The startup rate which measures the rate of change of the neutron flux level, is displayed for the operator in a range from -0.1 decades to +7 decades per minute. A high startup rate of +2 decades per minute in either channel will initiate a control rod withdrawal inhibit. |
| APPLICABLE SAFETY ANALYSES | Wide range neutron flux channels are necessary to monitor core reactivity changes and are the primary indication to trigger operator actions to anticipate Reactor Protection System actuation in the event of reactivity transients starting from low power conditions.<br><br>The wide range neutron flux channels satisfy Criterion 2 of 10 CFR 50.36 (Ref. 1). |
| LCO | Two wide range neutron flux instrumentation channels shall be OPERABLE to provide the operator with redundant neutron flux indication. These enable operators to control the increase in power and to detect neutron flux transients. This indication is used until the power range instrumentation is on scale. Violation of this requirement could prevent the operator from detecting and controlling neutron flux transients that could result in reactor trip during power escalation. |
| APPLICABILITY | The wide range neutron flux channels shall be OPERABLE in MODE 2 and in MODES 3, 4, and 5 with any CONTROL ROD drive (CRD) trip breaker in the closed position and the CRD System capable of rod withdrawal. |

BASES

APPLICABILITY (continued)

The wide range instrumentation is designed to detect power changes during initial criticality and power escalation when the power range and source range instrumentation cannot provide reliable indications. Since these conditions can exist in, or propagate from, all of these MODES, the wide range instrumentation must be OPERABLE.

ACTIONS

A.1

If one required wide range channel becomes inoperable, the unit is exposed to the possibility that a single failure will disable all neutron monitoring instrumentation. To avoid this, the inoperable channel must be repaired or power must be reduced to the point where source range channels can provide neutron flux indication. Completion of Required Action A.1 places the unit in this state, and LCO 3.3.9, "Source Range Neutron Flux," requires OPERABILITY of two source range channels once this state is reached. If the one channel failure occurs when indicated power is < 4E-4% RTP, the Required Action prohibits increases in power above the source range capability.

The 2 hour Completion Time allows controlled reduction of power into the source range and is based on unit operating experience that demonstrates the improbability of the second wide range channel failing during the allowed interval.

B.1 and B.2

With two required wide range neutron flux channels inoperable when THERMAL POWER is ≤ 5% RTP, the operators must place the reactor in the next lowest condition for which the wide range instrumentation is not required. This involves providing power level indication on the source range instrumentation by immediately suspending operations involving positive reactivity changes and, within 1 hour, placing the reactor in the tripped condition with the CRD trip breakers open. The Completion Times are based on unit operating experience and allow the operators sufficient time to manually insert the CONTROL RODS prior to opening the CRD breakers.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.3.10.1

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

Agreement criteria are based on a combination of the channel instrument uncertainties, including isolation, indication, and readability. If a channel is outside the criteria, it may be an indication that the signal processing equipment has drifted outside its limit. If the channels are within the criteria, it is an indication that the channels are OPERABLE.

The Frequency, equivalent to every shift, is based on operating experience that demonstrates channel failure is rare. Since the probability of two random failures in redundant channels in any 12 hour period is extremely low, the CHANNEL CHECK minimizes the chance of loss of protective function due to failure of redundant channels. The CHANNEL CHECK supplements less formal, but potentially more frequent, checks of channel OPERABILITY during normal operational use of the displays associated with the LCO's required channels.

When operating in Required Action A.1, CHANNEL CHECK is still required. However, in this condition, a redundant wide range may not be available for comparison. CHANNEL CHECK may still be performed via comparison with power or source range detectors, if available, and verification that the OPERABLE wide range channel is energized and indicates a value consistent with current unit status.

SR 3.3.10.2

For wide range neutron flux channels, CHANNEL CALIBRATION is a complete check and readjustment of the channels, from the preamplifier input to the indicators. This test verifies the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift to ensure that the instrument channel remains operational between successive tests.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.10.2 (continued)

The SR is modified by a Note excluding neutron detectors from CHANNEL CALIBRATION. It is not necessary to test the detectors because generating a meaningful test signal is difficult. In addition, the detectors are of simple construction, and any failures in the detectors will be apparent as a change in channel output. The Frequency is based on operating experience and consistency with the typical industry refueling cycle and is justified by demonstrated instrument reliability over an 18 month interval such that the instrument is not adversely affected by drift.

SR 3.3.10.3

SR 3.3.10.3 is the verification once each reactor startup of one decade of overlap with the source range neutron flux instrumentation. The wide range detector should be on scale and indicating ≥ 1E-8% of RTP when the source range detector is indicating ≤ $10^4$ counts per second in order for the wide range detector to indicate a one decade change prior to the source range detector going off scale. This ensures a continuous source of power indication during the approach to criticality.

The test may be omitted if performed within the previous 7 days based on operating experience, which shows that source range and wide range instrument overlap does not change appreciably within this test interval.

REFERENCES

1. 10 CFR 50.36.

## B 3.3 INSTRUMENTATION

### B 3.3.11 Main Steam Line Break (MSLB) Detection and Main Feedwater (MFW) Isolation Instrumentation

BASES

BACKGROUND

The MSLB Detection and MFW Isolation instrumentation is designed to address containment overpressurization concerns by isolating main feedwater (MFW) to both steam generators during an MSLB and to mitigate core overcooling concerns.

Steam generator header pressure is used as input signals to the MSLB circuitry for detection and feedwater isolation. When a MSLB is sensed, or upon manual actuation, the main feedwater control valves (MFCVs) and startup feedwater control valves (SFCVs) will be closed to isolate the MFW flow paths to both steam generators. In addition, the MFW pumps are tripped. The turbine-driven emergency feedwater (TDEFW) pump will be inhibited from auto-starting or will be auto-stopped if it has already started. A manual override for the TDEFW pump inhibit is provided to allow the operator to subsequently start the TDEFW pump if necessary for decay heat removal. These functions are credited for mitigating an MSLB. The function of closing the main and startup feedwater block valves is not credited in the MSLB analysis for mitigation of containment overpressurization during a MSLB. However, the MSLB detection and MFW isolation circuitry performs this function.

There are three pressure transmitters per steam generator with each feeding a steam pressure signal to a signal isolator (when used) and bistable. These bistables are calibrated to provide an ON/OFF signal at the desired setpoint for actuation of the feedwater isolation circuitry. A pressure transmitter and its associated signal isolator(s) and bistable(s) constitute a MSLB detection analog channel.

The six MSLB detection analog channels feed two redundant feedwater isolation digital channels consisting of two single failure proof two-out-of-three logic circuits. If the logic is satisfied, a master relay coil is energized. The use of an energized master relay ensures that a loss of power to the digital channels will not result in an in advertent feedwater isolation. If either digital channel is actuated, an MFW isolation will occur. Energizing the master relay results in closure of contacts in various control circuits for systems and components used for the MSLB containment overpressurization protection. Therefore, when the master relay is

BASES

BACKGROUND (continued)

energized, the systems and components perform their isolation functions. Other features of the digital channels include a test/manual actuation pushbutton, a circuit seal-in after the master relay is energized, a 2 second time delay to prevent spurious actuation, and an "enable" or "arming" switch. The two two-out-of-three logic circuits, along with their associated enable switch, master relay, seal-in, time delay, and test/manual actuation pushbutton are considered a feedwater isolation digital channel.

The feedwater isolation digital channels are enabled and disabled administratively rather than automatically. Appropriate operating procedures contain provisions to enable/disable the digital channels.

APPLICABLE SAFETY ANALYSES

Based on the containment pressure response reanalysis, the containment design pressure would be exceeded for a MSLB inside containment without operator action to isolate main feedwater and installed equipment necessary to automatically isolate main feedwater to both steam generators during a MSLB.

Steam generator header pressure is used as input to the MSLB circuitry for detection and feedwater isolation. When a MSLB is sensed, or upon manual actuation, the MFCVs and SFCVs are closed to isolate the MFW flow paths to both steam generators. In addition, the MFW pumps are tripped. The TDEFW pump will be inhibited from auto-starting or will be auto-stopped if it has already started. A manual override for the TDEFW pump inhibit is provided to allow the operator to subsequently start the TDEFW pump if necessary for decay heat removal. All of these functions are credited for mitigating a MSLB inside containment.

The MSLB Detection and MFW Isolation Instrumentation satisfies Criterion 3 of 10 CFR 50.36 (Ref. 3).

LCO

This LCO requires that instrumentation necessary to initiate a MFW isolation shall be OPERABLE. Failure of any instrument renders the affected channel(s) inoperable and reduces the reliability of the Function.

Three channels per SG are required to be OPERABLE to ensure that no single failure prevents MFW isolation. Each MSLB Detection and MFW Isolation instrumentation channel includes the sensor and measurement channel.

BASES (continued)

APPLICABILITY

The MSLB Detection and MFW Isolation Function shall be OPERABLE in MODES 1 and 2, and MODE 3 with main steam header pressure ≥ 700 psig because the SG inventory can be at a high energy level and contribute significantly to the peak pressure with a secondary side break. The main feedwater must be able to be isolated on each SG to limit mass and energy releases to the reactor building. Once the SG pressures have decreased below 700 psig, the MFW Isolation Function can be bypassed to avoid actuation during normal unit cooldowns. Also during MODE 3 the MFW isolation Function is not required to be OPERABLE when all main feedwater control valves (MFCVs) and startup feedwater control valves (SFCVs) are closed since the function of the instrumentation is already fulfilled. In MODES 4, 5, and 6, the energy level is low and the secondary side feedwater flow rate is low or nonexistent. In MODES 4, 5, and 6, the primary system temperatures are too low to allow the SGs to effectively remove energy and MSLB Detection and MFW Isolation instrumentation is not required to be OPERABLE.

ACTIONS

If a channel's trip setpoint is found nonconservative with respect to the Allowable Value, or any of the transmitter or signal processing electronics, are found inoperable, then the Function provided by that channel must be declared inoperable and the unit must enter the appropriate Conditions.

A Note has been added to the ACTIONS indicating that a separate Condition entry is allowed for instrumentation channels associated with each SG (MFW isolation function).

A.1

Condition A applies to failures of a single MSLB Detection and MFW Isolation instrumentation channel in one or more MFW Isolation Functions.

With one channel inoperable in one or more MSLB Detection and MFW Isolation Function, the channel(s) must be placed in trip within 4 hours. Tripping the affected channel places the Function in a one-out-of-two configuration. Operation in this configuration may continue indefinitely since the MSLB Detection and MFW Isolation Function is capable of performing its isolation function in the presence of any single random failure. The Completion Time of 4 hours is adequate to perform Required Action A.1.

BASES

ACTIONS (continued)

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

With two channels in one or more MSLB Detection and MFW Isolation Function inoperable or the Required Action and associated Completion Time of Condition A not met, the unit must be placed in MODE 3 within 12 hours and main steam header pressure must be reduced to less than 700 psig or all MFCVs and SFCVs must be closed within 18 hours. The allowed Completion Time is reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.3.11.1

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

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

The Frequency, about once every shift, is based on operating experience that demonstrates channel failure is rare. Since the probability of two random failures in redundant channels in any 12 hour period is extremely low, the CHANNEL CHECK minimizes the chance of loss of protective function due to failure of redundant channels. The CHANNEL CHECK supplements less formal, but potentially more frequent, checks of channel OPERABILITY during normal operational use of the displays associated with the LCO required channels.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.11.2

A CHANNEL FUNCTIONAL TEST is performed on each required instrumentation channel to ensure the channel will perform its intended function.

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 in any 31 day interval is a rare event.

This SR is modified by a Note indicating that it is only applicable when modifications are implemented that allow online testing.

SR 3.3.11.3

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

The Frequency is based on the assumption of an 18 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

REFERENCES

1. 10 CFR 50.36.

## B 3.3 INSTRUMENTATION

### B 3.3.12 Main Steam Line Break (MSLB) Detection and Main Feedwater (MFW) Isolation Manual Initiation

BASES

**BACKGROUND**

The MSLB Detection and MFW Isolation manual initiation capability provides the operator with the capability to actuate the isolation function from the control room. This Function is provided in the event the operator determines that the Function is needed and does not automatically actuate. This is a backup Function to the automatic MFW isolation.

The MSLB Detection and MFW Isolation manual initiation circuitry satisfies the manual initiation and single-failure criterion requirements of IEEE-279-1971 (Ref. 1).

**APPLICABLE SAFETY ANALYSES**

The MFW Isolation Function credited in the safety analysis is automatic. However, the manual initiation Function is required by design as backup to the automatic Function and allows operators to actuate MFW Isolation whenever the Function is needed. Furthermore, the manual initiation of MFW Isolation may be specified in unit operating procedures.

The MSLB Detection and MFW Isolation manual initiation function satisfy Criterion 3 of 10 CFR 50.36 (Ref. 2).

**LCO**

One manual initiation switch per actuation channel (A and B) is required to be OPERABLE. The MFW Isolation function, has two actuation or "trip" channels, channels A and B. Within each channel actuation logic there is one manual trip switch. When the manual switch is depressed, a full trip of actuation channel A or B occurs.

**APPLICABILITY**

The MFW Isolation manual initiation Function shall be OPERABLE in MODES 1 and 2, and MODE 3 with main steam header pressure $\geq$ 700 psig because SG inventory can be at a sufficiently high energy level to contribute significantly to the peak containment pressure with a secondary side break. During MODE 3, the MFW Isolation manual initiation Function is not required to be OPERABLE when all main feedwater control valves (MFCVs) and startup feedwater control valves (SFCVs) are closed since its function is already fulfilled. In MODES 4, 5, and 6, the SG energy level is low and secondary side feedwater flow rate is low or nonexistent.

BASES (continued)

ACTIONS

A.1

With one manual initiation switch inoperable, the manual initiation switch must be restored to OPERABLE status within 72 hours. The Completion Time of 72 hours is based on unit operating experience and administrative controls, which provide alternative means of MSLB Detection and MFW Isolation Function initiation via individual component controls. The 72 hour Completion Time is consistent with the allowed outage time for the components actuated by the MSLB Detection and MFW Isolation Function.

B.1

With both manual initiation switches inoperable or the Required Action and associated Completion Time of Condition A not met, the unit must be placed in MODE 3 within 12 hours and the main steam header pressure reduced to less than 700 psig or all MFCVs and SFCVs must be closed within 18 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required MODES from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.3.12.1

This SR requires the performance of a CHANNEL FUNCTIONAL TEST to ensure that the channels can perform their intended functions. The Frequency of 18 months is based on engineering judgment and operating experience that determined testing on an 18 month interval provides reasonable assurance that the circuitry is available to perform its safety function, while the risks of testing during unit operation is avoided.

REFERENCES

1. IEEE-279-1971, April 1972.

2. 10 CFR 50.36.

## B 3.3 INSTRUMENTATION

### B 3.3.13 Main Steam Line Break (MSLB) Detection and Main Feedwater (MFW) Isolation Logic Channels

BASES

BACKGROUND

The six MSLB detection analog channels feed two redundant feedwater isolation digital channels consisting of two single failure proof two-out-of-three logic circuits. If the logic is satisfied, a master relay coil is energized. The use of an energized master relay ensures that a loss of power to the digital channels will not result in an inadvertent feedwater isolation. If either digital channel is actuated, an MFW isolation will occur. Energizing the master relay results in closure of contacts in various control circuits for systems and components used for the MSLB containment overpressurization protection. Therefore, when the master relay is energized, the systems and components perform their isolation functions. Other features of the digital channels include a test/manual actuation pushbutton, a circuit seal-in after the master relay is energized, a 2 second time delay to prevent spurious actuation, and an "enable" or "arming" switch. Each of the two two-out-of-three logic circuits, along with their associated enable switch, master relay, seal-in, and time delay is considered a feedwater isolation digital channel.

APPLICABLE SAFETY ANALYSES

MSLB circuitry is installed equipment necessary to automatically isolate main feedwater to both steam generators during a MSLB.

Steam generator outlet pressure is used as input to the MSLB circuitry for detection and feedwater isolation. When a MSLB is sensed, or upon manual actuation, the MFCVs and SFCVs will be closed to isolate the MFW flow paths to both steam generators. In addition, the MFW pumps are tripped. The TDEFW pump will be inhibited from auto-starting or will be auto-stopped if it has already started and the switch for MS-93 is in the AUTO position. A manual override for the TDEFW pump inhibit is provided to allow the operator to subsequently start the TDEFW pump if necessary for heat removal. All of these functions are credited for mitigating a MSLB inside containment.

The MSLB Detection and MFW Isolation logic channels satisfy Criterion 3 of 10 CFR 50.36 (Ref. 1).

BASES (continued)

LCO

Two channels of MSLB Detection and MFW Isolation automatic actuation logic shall be OPERABLE. There are only two channels of automatic actuation logic. Therefore, violation of this LCO could result in a complete loss of the automatic Function assuming a single failure of the other channel.

APPLICABILITY

The MSLB Detection and MFW Isolation automatic actuation logic channels shall be OPERABLE in MODES 1 and 2, and MODE 3 with main steam header pressure ≥ 700 psig because SG inventory can be at a high energy level and can contribute significantly to the peak containment pressure during a secondary side line break. Also, during MODE 3, the MFW Isolation function is not required to be OPERABLE when all main feedwater control valves (MFCVs) and startup feedwater control valves (SFCVs) are closed since its function is already fulfilled. In MODES 4, 5, and 6, the energy level is low and the secondary side feedwater flow rate is low or nonexistent.

ACTIONS

A.1

With one automatic actuation logic channel inoperable, the channel must be restored to OPERABLE status within 72 hours. The Completion Time of 72 hours is based on unit operating experience and administrative controls, which provide alternative means of MSLB Detection and MFW Isolation Function initiation via individual component controls. The 72 hour Completion Time is consistent with the allowed outage time for the components actuated by the MSLB Detection and MFW Isolation Function.

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

With both logic channels inoperable or the Required Action and associated Completion Time not met, the unit must be placed in MODE 3 within 12 hours and the main steam header pressure must be reduced to less than 700 psig or all MFCVs and SFCVs must be closed within 18 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required MODES from full power conditions in an orderly manner and without challenging unit systems.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.3.13.1

This SR requires the performance of a CHANNEL FUNCTIONAL TEST to ensure that the channels can perform their intended functions. This test verifies MFW Isolation automatic actuation logics are functional. This test simulates the required inputs to the logic circuit and verifies successful operation of the automatic actuation logic. The Frequency of 18 months is based on engineering judgment and operating experience that determined testing on an 18 month interval provides reasonable assurance that the circuitry is available to perform its safety function, while the risks of testing during unit operation is avoided.

REFERENCES

1. 10 CFR 50.36.

## B 3.3 INSTRUMENTATION

### B 3.3.14 Emergency Feedwater (EFW) Pump Initiation Circuitry

BASES

BACKGROUND

EFW pump initiation circuitry is designed to provide safety grade means of controlling the secondary system as a heat sink for core decay heat removal. To ensure the secondary system remains a heat sink, the EFW pump initiation circuitry takes action to initiate EFW when the primary source of feedwater is lost. These actions ensure that a source of cooling water is available to be supplied to a steam generator (SG), thereby establishing the heat sink temperature at the saturation temperature of the secondary system.

EFW is initiated to restore a source of cooling water to the secondary system when conditions indicate that the normal source of feedwater is not available. Loss of both MFW Pumps was chosen as an EFW automatic initiating parameter because it is a direct and immediate indicator of loss of MFW. The EFW pump initiation circuitry contains devices that generate an EFW pump initiation signal when loss of main feedwater pumps are indicated by low hydraulic oil pressure. Each EFW Pump initiation circuit is fed by two loss of main feedwater (LOMF) instrumentation channels (hydraulic oil pressure switches) common only to that circuit which feed a two-out-of-two logic circuit that automatically starts each EFW pump. Each EFW pump also has a dedicated manual start circuit.

EFW is also initiated by a low level in the SG (after a 30 second delay to prevent spurious actuation) for SG dryout protection. EFW initiation for SG dryout protection is not required by this Specification. Finally, EFW is also initiated by a loss of both MFW pumps as indicated by low hydraulic oil pressure as part of the ATWS Mitigation Circuitry (AMSAC), which is a system provided to comply with the requirements to reduce risk from an anticipated transient without scram (ATWS). EFW initiation for ATWS mitigation is not required by this Specification.

Each motor driven EFW pump is normally controlled by a four-position, OFF-AUTO1-AUTO2-RUN, control switch located in the control room. The pump can be manually started by turning the control switch to the RUN position. In the AUTO1 mode, each motor-driven EFW pump starts automatically after a sustained low water level in either steam generator for greater than 30 seconds. In the AUTO2 Mode, each pump starts automatically on low steam generator level or loss of both main feedwater pumps.

BASES

BACKGROUND (continued)

The turbine-driven EFW pump is started by opening valve MS-93 which admits steam to the pump turbine. A four-position, RUN-AUTO-OFF-PULL TO LOCK, control switch is provided to control operation of MS-93. The switch is maintained in the AUTO position. In the AUTO mode, MS-93 opens on low hydraulic oil pressure in both MFW pumps. When the switch is in the RUN position, MS-93 is opened.

APPLICABLE SAFETY ANALYSES

The transient which forms the basis for initiation of the EFW systems is a loss of MFW transient. In the analysis of the transient, MFW pump turbine low hydraulic oil pressure is the parameter assumed to automatically initiate EFW.

The EFW pump initiation circuitry satisfies Criterion 3 of 10 CFR 50.36 (Ref. 2).

LCO

Two loss of main feedwater (LOMF) pump instrumentation channels and an automatic initiation circuit and a manual initiation circuit are required OPERABLE for each EFW pump. Each LOMF instrumentation channel is considered to include the sensors and measurement channels. The LCO is modified by a Note that limits the OPERABILITY required for the automatic initiation circuitry to MODES 1 and 2.

APPLICABILITY

The initiation circuitry for EFW pumps shall be OPERABLE in MODES 1, 2 and 3 and in MODE 4 when the steam generator is relied upon for heat removal. In MODE 4 when the steam generator is not relied upon for heat removal, and MODES 5, and 6, the primary system temperatures are too low to allow the SGs to effectively remove energy and EFW Pump initiation instrumentation is not required to be OPERABLE.

ACTIONS

The ACTIONS are modified by a Note indicating that this Specification may be entered independently for each EFW pump initiation circuit. The Completion Time(s) of the inoperable channels for each EFW automatic initiation circuit are tracked separately for each circuit starting from the time the Condition is entered for that circuit.

BASES

ACTIONS (continued)

A.1

With one or more required EFW pump initiation circuits with one LOMF channel inoperable, the channel(s) must be placed in trip within 1 hour. With the channel in trip, the resultant logic is one-out-of-one. This channel may be considered placed in trip, after tripping, by installing jumpers or by other means that assure the channel remains in the tripped condition.

B.1

With one or more EFW pump initiation circuits inoperable or the Required Action and associated Completion Time of Condition A not met, the affected EFW pump(s) must be declared inoperable immediately since the initiation function is no longer capable of performing its safety function.

SURVEILLANCE REQUIREMENTS

SR 3.3.14.1

This SR requires the performance of a CHANNEL FUNCTIONAL TEST to ensure the LOMF pump instrumentation channels can perform their intended function.

The Frequency of 31 days is based on operating experience with regard to channel OPERABILITY, which demonstrates that failure of more than one channel of a given function in any 31 day interval is a rare event.

SR 3.3.14.2

This SR requires the performance of a CHANNEL FUNCTIONAL TEST of the manual initiation circuit. This test verifies that the initiating circuitry is OPERABLE and will actuate the emergency feedwater pumps by either starting a motor driven emergency feedwater pump or opening the steam isolation valve that isolates the supply of steam to the drive for the turbine driven emergency feedwater pump.

SR 3.3.14.3

This SR requires the performance of a CHANNEL FUNCTIONAL TEST of the automatic initiation circuit. This test verifies that the two-out-of-two logic circuit is functional. This test simulates the required inputs to the logic

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.14.3 (continued)

circuit and verifies successful operation of the automatic initiation circuit. The Frequency of 18 months is based on engineering judgment and operating experience that determined testing on an 18 month interval provides reasonable assurance that the circuitry is available to perform its safety function, while the risks of testing during operation are avoided.

SR 3.3.14.4

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

The Frequency is based on the assumption of an 18 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

REFERENCES

1. UFSAR, Chapters 7 and 15.

2. 10 CFR 50.36.

## B 3.3 INSTRUMENTATION

### B 3.3.15 Turbine Stop Valve (TSV) Closure

BASES

BACKGROUND

The Turbine Stop Valves (TSV) Closure function partially isolates the main steam lines from the SGs by closing the TSVs on both main steam lines following a turbine or reactor trip signal.

Two TSVs are provided for each main steam line and are located outside of containment. The TSVs are downstream from the main steam safety valves (MSSVs) and emergency feedwater pump turbine's steam supply to prevent the MSSVs and EFW pump's steam supply from being isolated from the steam generators by TSV closure. Closing the TSVs partially isolates each steam generator from the other, and isolates the turbine from the steam generators.

TSV Closure is initiated by a reactor trip. To keep from rapidly cooling down the primary plant by drawing off too much steam, the turbine is tripped when the reactor trips. Two independent and redundant "Reactor Trip Confirmed" signals in the form of contact closures from the control rod drive system will energize two independent turbine trip mechanisms. The Channel A trip circuit will close all four TSVs within a maximum of 1 second. The Channel B trip circuit will close the TSVs within a maximum of 15 seconds.

APPLICABLE SAFETY ANALYSES

The design basis of the TSV Closure function is established by the analysis for the main steam line break (MSLB) as discussed in the UFSAR, Section 15.13 (Ref. 1). TSV closure is necessary to stop steam flow to the turbine (to prevent overcooling) following all reactor trips.

The accident analysis compares several different MSLB events. The MSLB outside containment upstream of the TSV is limiting for offsite dose, although a break in this section of main steam header has a very low probability. The MSLB with ICS low level control and without operator action prior to ten minutes is the limiting case for a post-trip return to power. The analysis includes scenarios with offsite power available and with a loss of offsite power following turbine trip. With offsite power available, the reactor coolant pumps continue to circulate coolant through the steam generators, maximizing the Reactor Coolant System (RCS) cooldown. With a loss of offsite power, the response of mitigating systems, such as the High Pressure Injection (HPI) System pumps, is delayed.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The TSVs remain open during power operation. These valves close upon a reactor trip.

a. For an HELB or an MSLB inside containment, steam is discharged into containment from both steam generators until closure of the TSVs. After TSV closure, steam is discharged into containment only from the affected steam generator.

b. An MSLB outside of containment and upstream from the TSVs is not a containment pressurization concern. The uncontrolled blowdown of both steam generators must be prevented to limit the potential for uncontrolled RCS cooldown and positive reactivity addition. Closure of the TSVs isolates the break and limits the blowdown to a single steam generator.

c. An event such as increased steam flow through the turbine will terminate on closing the TSVs.

d. Following a steam generator tube rupture, closure of the TSVs isolates the ruptured steam generator from the intact steam generator.

The TSV Closure function satisfies Criterion 3 of 10 CFR 50.36 (Ref. 2).

LCO

Two TSV Closure channels are required to be OPERABLE.

This LCO provides assurance that the TSVs will perform their design safety function to mitigate the consequences of accidents that could result in offsite exposures comparable to the 10 CFR 100 limits (Ref. 3).

APPLICABILITY

Both TSV Closure channels must be OPERABLE in MODES 1, 2 and 3 with any TSVs open. In these conditions when there is significant mass and energy in the RCS and steam generators, the TSV Closure function must be OPERABLE or the TSVs closed. When the TSVs are closed, they are already performing the safety function.

In MODE 4, the steam generator energy is low. Therefore, the TSV Closure channels are not required to be OPERABLE. In MODES 5 and 6, the steam generators do not contain a significant amount of energy because their temperature is below the boiling point of water; therefore, the TSV Closure channels are not required for isolation of potential high energy secondary system pipe breaks in these MODES.

BASES (continued)

ACTIONS

A.1

With one or more TSV Closure channels inoperable, all TSVs must be declared inoperable. A Completion Time of 1 hour is provided to return the TSV Closure channels to OPERABLE status. The 1 hour Completion Time is sufficient time to correct minor problems.

SURVEILLANCE REQUIREMENTS

SR 3.3.15.1

This SR requires the performance of a CHANNEL FUNCTIONAL TEST to ensure that the channels can perform their intended function. This test verifies the TSV Closure automatic actuation channels are functional. This test simulates the required inputs to the logic circuit and verifies successful operation of the automatic actuation logic channels. The test need not include actuation of the end device. This is due to the risk of a unit transient caused by the closure of TSVs during testing at power. The Frequency of 31 days is based on engineering judgment and operating experience, which determined the interval provided adequate confidence that the TSV Closure channels are available to perform their safety function, while the risks of testing at operation are avoided.

REFERENCES

1. UFSAR, Section 15.13.

2. 10 CFR 50.36.

3. 10 CFR 100.

## B 3.3 INSTRUMENTATION

### B 3.3.16 Reactor Building (RB) Purge Isolation–High Radiation

BASES

BACKGROUND

The RB Purge Isolation–High Radiation Function closes the RB purge valves. This action isolates the RB atmosphere from the environment to minimize releases of radioactivity in the event an accident occurs.

The radiation monitoring system measures the activity in a representative sample of air drawn in succession through a particulate sampler, an iodine sampler, and a gas sampler. The LCO addresses only the gas sampler portion of this system (RIA-45).

The trip setpoint is chosen sufficiently below hazardous radiation levels to ensure that the consequences of an accident will be acceptable, provided the unit is operated within the LCOs at the onset of an accident or transient and the equipment functions as designed.

The closure of the purge valves ensures the RB remains as a barrier to fission product release. There is no bypass for this function.

APPLICABLE SAFETY ANALYSES

During movement of irradiated fuel assemblies within containment, the most severe radiological consequences result from a fuel handling accident. The fuel handling accident is a postulated event that involves damage to irradiated fuel (Ref. 1). A minimum fuel transfer canal water level and the minimum decay time of 72 hours prior to movement of irradiated fuel assemblies from the reactor ensure that the release of fission product radioactivity subsequent to a fuel handling accident results in doses that are within the guideline values specified in 10 CFR 100. The design basis for fuel handling accidents has historically separated the radiological consequences from the containment capability. The NRC staff has treated the containment capability for fuel handling conditions as a logical part of the "primary success path" to mitigate fuel handling accidents, regardless of the assumptions used to calculate the radiological consequences of such accidents (Ref. 1).

The RB Purge Isolation System satisfies Criterion 3 of 10 CFR 50.36 (Ref. 2).

BASES (continued)

LCO

One channel of RB Purge Isolation-High Radiation instrumentation is required to be OPERABLE. OPERABILITY of the instrumentation includes proper operation of the sample pump. This LCO addresses only the gas sampler portion of the System.

APPLICABILITY

The RB purge isolation–high radiation instrumentation shall be OPERABLE whenever CORE ALTERATIONS or movement of irradiated fuel assemblies within the RB is taking place. These conditions are those under which the potential for fuel damage, and thus radiation release, is the greatest. While in MODES 1, 2, 3, and 4, the Purge Valve Isolation System does not need to be OPERABLE because the purge valves are required to be sealed closed. While in MODES 5 and 6, without fuel handling in progress, the Purge Valve Isolation System does not need to be OPERABLE because the potential for a radioactive release is minimized. The need to use the purge valves in MODES 5 and 6 is in preparation for entry. This capability is required to minimize doses for personnel entering the building and is independent of the automatic isolation capability.

ACTIONS

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

Condition A applies to failure of the high radiation purge function during CORE ALTERATIONS or during movement of irradiated fuel assemblies within the RB.

With one channel inoperable during CORE ALTERATIONS or during movement of irradiated fuel assemblies within the RB, the RB purge valves must be closed, or CORE ALTERATIONS and movement of irradiated fuel assemblies within the RB must be suspended. Required Action A.1 accomplishes the function of the high radiation channel. Required Action A.2.1 and Required Action A.2.2 place the unit in a configuration in which purge isolation on high radiation is not required. The Completion Time of "Immediately" is consistent with the urgency associated with the loss of RB isolation capability under conditions in which the fuel handling accidents are possible and the high radiation function provides the only automatic actions to mitigate radiation release.

SURVEILLANCE REQUIREMENTS

SR 3.3.16.1

SR 3.3.16.1 is the performance of the CHANNEL CHECK for the RB purge isolation–high radiation instrumentation once every 12 hours to ensure that a gross failure of instrumentation has not occurred. The CHANNEL CHECK is normally a comparison of the parameter indicated on the

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.16.1 (continued)

radiation monitoring instrumentation channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. Performance of the CHANNEL CHECK helps to ensure that the instrumentation continues to operate properly between each CHANNEL CALIBRATION. The high radiation instrumentation should be compared to similar unit instruments located throughout the unit. If the radiation monitor uses keep alive sources or check sources OPERABLE from the control room, the CHANNEL CHECK should also note the detector's response to these sources.

Agreement criteria are based on a combination of the channel instrument uncertainties, including isolation, indication, and readability. If a channel is outside the criteria, it may be an indication that the transmitter or the signal processing equipment has drifted outside its limit. If the channels are within the criteria, it is an indication that the channels are OPERABLE. The 12 hour Frequency, about once every shift, is based on operating experience that demonstrates channel failure is rare. Additionally, control room alarms and annunciators are provided to alert the operator to various "trouble" conditions associated with the instrument.

SR 3.3.16.2

This SR requires the performance of a CHANNEL FUNCTIONAL TEST to ensure that the channel can perform its intended function. The frequency requires the isolation capability of the reactor building purge valves to be verified functional once each refueling outage prior to CORE ALTERATIONS or movement of irradiated fuel assemblies within containment. This ensures that this function is verified prior to irradiated fuel assembly handling within containment. This test verifies the capability of the instrumentation to provide the RB isolation.

SR 3.3.16.3

CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy.

The 18 month Frequency is based on engineering judgment and industry accepted practice.

BASES (continued)

REFERENCES

1. NRC Letter to RG & E dated December 7, 1995 R.E. Ginna Nuclear Power Plant conversion to Improved Standard Technical Specifications - Resolution of Ginna Design Basis for Refueling Accidents.

2. 10 CFR 50.36.

## B 3.3 INSTRUMENTATION

### B 3.3.17 Emergency Power Switching Logic (EPSL) Automatic Transfer Function

BASES

BACKGROUND

The transfer circuits of the EPSL are designed with sufficient redundancy to assure that power is supplied to the unit Main Feeder Buses (MFBs) and, hence, to the unit's essential loads, under accident conditions. The logic system monitors the normal and emergency power sources and, upon loss of the normal power source (the unit auxiliary transformer), the logic seeks an available alternate source of power.

The Load Shed and Transfer to Standby Circuits are designed to energize the MFBs from the Standby Buses powered from either Keowee or Lee when voltage is lost or is insufficient from the Normal and Startup sources. The Load Shed signal is generated to separate nonessential loads from the MFBs to ensure the CT-4 or CT-5 transformers supplying the Standby Buses are not overloaded. The Load Shed timers and Transfer to Standby timers are set such that, if no power is available from the startup source for approximately 11 seconds, the startup source breakers are prohibited from closing and the standby bus to MFB breakers receive a permissive to close.

The Retransfer to Startup logic provides the emergency power switching logic the capability to retransfer essential loads from the Standby Bus to the startup source, if available, should power to both standby buses be lost for more than 5 seconds.

The EPSL automatic transfer function is designed to perform their function assuming a single failure. There are two automatic transfer channels, with one channel consisting of Channel A of the Load Shed and Transfer to Standby function and Channel A of the Retransfer to Startup function and the other consisting of Channel B of both of these functions.

APPLICABLE SAFETY ANALYSES

The EPSL Automatic Transfer function is required for the engineered safeguards (ES) equipment to function in any accident with a loss of offsite power.

The limiting accident for the EPSL transfer functions is a LOCA with a simultaneous loss of offsite power (Ref. 1).The loss of offsite power is considered to occur coincident with ES actuation. In this scenario, the

BASES

APPLICABLE SAFETY ANALYSES (continued)

Load Shed and Transfer to Standby function reenergizes the affected unit's MFBs from the standby buses which are powered from Keowee or Lee.

The analyses assume that the maximum time the MFBs will be deenergized is 33 seconds. This time is derived from the 53 second time requirement for full LPI injection minus the 15 second ECCS valve stroke time requirement and 5 seconds for the pump to get to rated speed.

EPSL automatic transfer functions are part of the primary success path and function to mitigate an accident or transient that presents a challenge to the integrity of a fission product barrier. The EPSL automatic transfer function satisfies Criterion 3 of 10 CFR 50.36 (Ref. 2).

LCO

Two channels of the Automatic Transfer Function, with one channel consisting of Channel A of the Load Shed and Transfer to Standby function and Channel A of the Retransfer to Startup function and the other consisting of Channel B of both of these functions, are required to be OPERABLE. Failure of one channel reduces the reliability of the affected Functions.

The requirement for two channels to be OPERABLE ensures that one channel of the function will remain OPERABLE if a single failure has occurred. The remaining channel can perform the safety function.

APPLICABILITY

The automatic transfer function of EPSL is required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure that power is provided from AC Sources to the AC Distribution system within the time assumed in the accident analyses.

The EPSL automatic transfer function is not required to be OPERABLE in MODES 5 and 6 since more time is available for the operator to respond to a loss of power event.

ACTIONS

A.1

If one channel is inoperable, it must be restored to OPERABLE status within 24 hours. With one channel inoperable, the remaining channel is capable of providing necessary transfer functions to ensure power is provided to the MFBs. The 24 hour Completion Time is considered appropriate based on engineering judgement, taking into consideration the time required to complete the required action.

BASES

ACTIONS

A.1 (continued)

Required Action A.1 is modified by a Note which indicates that the Completion Time is reduced when in Condition L of LCO 3.8.1. Condition L limits the Completion Time for restoring an inoperable channel to 4 hours when emergency power source(s) or offsite power source(s) are inoperable for extended time periods or for specific reasons.

B.1 and B.2

With the Required Action and associated Completion Time 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 in 12 hours and to MODE 5 within 84 hours. The allowed Completion Times are reasonable, based on operating experience, to allow for a controlled shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.3.17.1

This SR requires the performance of a CHANNEL FUNCTIONAL TEST of the EPSL automatic transfer function. The ES inputs to the Load Shed and Transfer to Standby function and the Retransfer to Startup function are verified to operate properly during an automatic transfer of the Main Feeder Buses to the Startup Transformer, Standby Buses, and retransfer to the Startup Transformers. The Frequency of 18 months is based on engineering judgment and operating experience that determined testing on an 18 month interval provides reasonable assurance that the circuitry is available to perform its safety function.

REFERENCES

1. UFSAR, Chapters 6 and 15.

2. 10 CFR 50.36.

B 3.3 INSTRUMENTATION

B 3.3.18 Emergency Power Switching Logic (EPSL) Voltage Sensing Circuits

BASES

BACKGROUND

The EPSL voltage sensing circuits for the Startup Transformer, Standby Bus #1, Standby Bus #2, and the Auxiliary Transformer provide input to the EPSL controls to actuate breakers and initiate transfer control sequences. Each phase of each source has an individual potential transformer feeding a 2 out of 3 logic for determining the status of the power source. The voltage sensing circuits also provide trip signals to the breaker control circuitry for the normal incoming breakers (N breakers), startup incoming breakers (E breakers), and CT-5 incoming breakers (SL breakers).

The EPSL system is designed to ensure power is supplied to the main feeder buses (MFBs) during a LOCA. In order for it to perform this function, the voltage sensing circuits for the Startup Transformer, Auxiliary Transformer, Standby Bus #1, and Standby Bus #2 must be OPERABLE. These voltage sensing circuits provide input to the EPSL transfer functions. The transfer functions utilize the voltage sensing circuits to initiate breaker operations to ensure the MFBs are connected to an energized source (startup or standby). The N and E breakers also get direct trips from the two-out-of-three logic.

No protective relay lockouts or inhibits can be present to prevent the connection of required AC power source(s) to the MFBs from the Control Room because their presence will not allow closure of the associated breaker.

APPLICABLE SAFETY ANALYSES

The EPSL voltage sensing circuits are required for the engineered safeguards (ES) equipment to function in any accident with a loss of offsite power. The limiting accident for the EPSL voltage sensing circuits is a loss-of-coolant accident (LOCA) with a simultaneous loss of offsite power (Ref. 1).

The EPSL voltage sensing circuits satisfy Criterion 3 of 10 CFR 50.36 (Ref. 2).

BASES (continued)

LCO

Three channels of each EPSL voltage sensing circuit (Auxiliary Transformer, Startup Transformer, Standby Bus #1, Standby Bus #2) are required to be OPERABLE. These circuits and associated channels ensure that no single failure can cause a loss of required ES equipment.

The LCO is modified by two Notes. Note 1 removes Auxiliary Transformer voltage sensing requirements when both N breakers are open. The function of the Auxiliary Transformer Voltage Sensing circuits is to provide a trip signal to the N breakers. When the N breakers are open, the Auxiliary Transformer voltage sensing circuits are not required and, therefore, need not be OPERABLE. Note 2 requires only the EPSL voltage sensing circuits associated with required AC power source(s) to be OPERABLE when not in MODES 1, 2, 3, and 4.

APPLICABILITY

The EPSL voltage sensing circuits are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure that power is provided from AC Sources to the AC Distribution system within the time assumed in the accident analyses.

The EPSL voltage sensing circuits associated with required AC power source(s) required to be OPERABLE in MODES 5 and 6 and during movement of irradiated fuel assemblies provide assurance that:

a. Systems needed to provide adequate coolant inventory makeup are available for the irradiated fuel assemblies;

b. Systems needed to mitigate a fuel handling accident are available;

c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

ACTIONS

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

BASES

ACTIONS (continued)

A.1

If one required channel is inoperable in one or more voltage sensing circuits, it must be restored to OPERABLE status within 24 hours. With one channel inoperable, the remaining two channels are capable of providing the voltage sensing function. The 24 hour Completion Time is considered appropriate based on engineering judgement taking into consideration the time required to complete the required action.

Required Action A.1 is modified by a Note which indicates that the Completion Time is reduced when in Condition L of LCO 3.8.1. Condition L limits the Completion Time for restoring an inoperable channel to 4 hours when emergency power source(s) or offsite power source(s) are inoperable for extended time periods or for specific reasons.

B.1 and B.2

With the Required Action and associated Completion Time not met in MODES 1, 2, 3 and 4, 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 in 12 hours and to MODE 5 within 84 hours. The allowed Completion Times are reasonable, based on operating experience, to allow for a controlled shutdown.

C.1 and C.2

With two or more channels of a required circuit inoperable when not in MODES 1, 2, 3, and 4 or the Required Action and associated Completion Time not met when not in MODES 1, 2, 3, and 4, the affected AC power sources(s) must be declared inoperable immediately. The appropriate Required Actions will be implemented in accordance with LCO 3.8.2, "AC Sources—Shutdown."

D.1

With the Required Action and associated Completion Time not met during movement of irradiated fuel assemblies, movement of fuel assemblies must be suspended immediately. Suspension does not preclude completion of actions to establish a safe conservative condition. This action minimizes the probability or the occurrence of postulated events. The Completion Time of immediately is consistent with the required times for actions requiring prompt attention.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.3.18.1

A CHANNEL FUNCTIONAL TEST is performed on each voltage sensing circuit channel to ensure the channel will perform its function. A circuit is defined as three channels, one for each phase. Each channel consists of components from the sensing power transformer through the circuit auxiliary relays which operate contacts in the EPSL logic and breaker trip circuits. Minimum requirements consist of individual channel relay operation causing appropriate contact responses within associated loadshed/breaker circuits, alarm activations, and proper indications for the sensing circuit control power status. The Frequency of 18 months is based on engineering judgment and operating experience that determined testing on an 18 month interval provides reasonable assurance that the circuitry is available to perform its safety function.

REFERENCES

1. UFSAR, Chapters 6 and 15.

2. 10 CFR 50.36.

## B 3.3 INSTRUMENTATION

### B 3.3.19 Emergency Power Switching Logic (EPSL) 230 kV Switchyard Degraded Grid Voltage Protection (DGVP)

BASES

BACKGROUND

Two levels of protection are provided to assure the degradation of voltage from offsite sources does not adversely impact the function of safety-related systems and components. The first level of protection is provided by the EPSL Degraded Grid Protection System (DGPS). The second level of protection is provided by undervoltage relaying on the E and N breakers (reference LCO 3.3.18, "EPSL Voltage Sensing Circuits") which protects from loss of voltage.

The DGPS, upon indication of inadequate voltage, provides an alarm to the Unit 1 & 2 Control Room. If an engineered safeguards (ES) Channel 1 or 2 signal from any unit is sensed by the DGPS, while the voltage is below acceptable levels, the DGPS will initiate an isolation of the 230 kV switchyard Yellow Bus to ensure the onsite overhead emergency power path is available. Each DGPS actuation logic channel is capable of initiating isolation of the overhead emergency power path. This ensures the startup transformers are not connected to a degraded source of power. In this event, ES loads are provided power from the standby buses.

Based on operating experience, degradation of voltage in the 230 kV switchyard does not last for an extended period of time. Administrative procedures are in place to assure timely actions are taken to restore the voltage.

There are three undervoltage relays installed to monitor the switchyard voltage, one on each phase (X, Y, Z) of the 230 kV Yellow Bus. The undervoltage relay contacts are arranged in a two-out-of-three logic sequence which feeds two redundant time delay relays. The time delay relays prevent spurious actuations, but still provide adequate response time for voltage transients. Either of the two redundant time-delay relays will cause either of the two sets of actuating relays to initiate switchyard isolation. Circuit control power is fed from the 230 kV Switchyard 125 VDC system.

BASES (continued)

APPLICABLE SAFETY ANALYSES

The EPSL Degraded Grid Voltage Protection function is required to ensure adequate voltage is available during an ES actuation when system grid voltages are not adequate (Ref. 1). Based on calculations, 219 kV is the minimum switchyard voltage that will ensure proper operation of loads during ES actuation.

The EPSL Degraded Grid Voltage Protection satisfies Criterion 3 of 10 CFR 50.36 (Ref. 2).

LCO

Three degraded grid voltage sensing relay channels are required to be OPERABLE. Failure of one channel reduces the reliability of the function. The requirement for three channels to be OPERABLE ensures that two channels will remain OPERABLE if a failure has occurred in one channel. The remaining channels can perform the safety function.

Two channels of the Degraded Grid Voltage Protection Actuation Logic function are required to be OPERABLE. The switchyard isolation circuit is considered a part of this logic channel. Therefore, if a switchyard isolation channel is inoperable, then one DGVP actuation channel is inoperable. The requirement for two channels to be OPERABLE ensures that one channel will remain OPERABLE if a failure has occurred in one channel. The remaining channel can perform the safety function.

APPLICABILITY

The DGPS functions are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure that power is provided from AC Sources to the AC Distribution system within the time assumed in the accident analyses.

The EPSL DGVP functions are not required to be OPERABLE in MODES 5 and 6 since more time is available for the operator to respond to a loss of power event.

ACTIONS

The ACTIONS are modified by a Note indicating that the Completion Times for Required Actions A and B are reduced when in Condition L of LCO 3.8.1. Condition L limits the Completion Time for restoring inoperable channels to 4 hours when emergency power source(s) or offsite power source(s) are inoperable for extended time periods or for specific reasons.

BASES

ACTIONS (continued)

A.1

If one DGVP voltage sensing channel is inoperable, the channel must be placed in trip within 72 hours. Tripping the affected channel places the function in a one-out-of-two configuration. Operation in this configuration may continue indefinitely since the DGVP function is capable of performing its DGVP function in the presence of a single failure. With one channel inoperable, the remaining channels are capable of providing the DGVP function. The 72 hour completion time is based on engineering judgement taking into consideration the infrequency of actual grid system voltage degradation, and the probability of an event requiring ES operation.

B.1

If one DGVP actuation logic channel is inoperable, the actuation logic channel must be restored to OPERABLE status within 72 hours. With one actuation logic channel inoperable, the remaining actuation logic channel is capable of providing the DGVP function. The 72 hour completion time is based on engineering judgement taking into consideration the infrequency of actual grid system voltage degradation, and the probability of an event requiring ES operation.

C.1 and C.2

With the Required Action and associated Completion Time of Condition A or B 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 in 12 hours and to MODE 5 within 84 hours. The allowed Completion Times are reasonable based on operating experience and to allow for a controlled shutdown.

D.1

With two or more voltage sensing channels or both actuation logic channels inoperable, degraded grid protection is no longer available to the Station during an ES actuation. The condition also prevents switchyard isolation during a LOCA. Since switchyard isolation is inoperable, the overhead power path must be declared inoperable immediately. The appropriate Required Actions will be implemented in accordance with LCO 3.8.1, AC "Sources–Operating."

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.3.19.1

A CHANNEL FUNCTIONAL TEST is performed on each DGVP voltage sensing channel and DGVP actuation logic channel to ensure the entire channel will perform its intended function. Any setpoint adjustments shall be consistent with the assumptions of the setpoint analysis. The CHANNEL FUNCTIONAL TEST of the DGVP actuation logic channels includes verifying actuation of the switchyard isolation circuitry. The Frequency of 18 months is based on engineering judgment and operating experience that determined testing on an 18 month interval provides reasonable assurance that the circuitry is available to perform its safety function.

SR 3.3.19.2

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

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

REFERENCES

1. UFSAR, Chapter 8.

2. 10 CFR 50.36.

## B 3.3 INSTRUMENTATION

### B 3.3.20 Emergency Power Switching Logic (EPSL) CT-5 Degraded Grid Voltage Protection (DGVP)

BASES

BACKGROUND

Two levels of protection are provided for the standby buses to assure that degradation of voltage from the 100 kV transmission system does not adversely impact the function of safety related systems and components. The first level of protection is provided by the EPSL CT-5 Degraded Grid Protection System. The second level of protection is provided by undervoltage relaying on the standby buses (reference LCO 3.3.18, "EPSL Voltage Sensing Circuits") which protects from loss of voltage.

Three undervoltage sensing relays provide common input to two channels of actuating logic. In addition to the three phase undervoltage sensing relays, each channel includes one time-delay relay, one auxiliary relay, and one associated single phase undervoltage sensing relay. Each channel trip signal passes through a selector switch, which either allows or inhibits the trip signal, to actuate one trip coil in each SL breaker. Inoperability of any voltage sensing channel reduces the logic for the voltage sensing function to a two-out-of-two. Loss of two or more voltage sensing relays results in inoperability of both channels of actuation logic.

APPLICABLE SAFETY ANALYSES

The EPSL CT-5 Degraded Grid Voltage Protection function is required to ensure adequate voltage is available during an ES actuation concurrent with a loss of offsite power or degraded voltage from the 230 kV switchyard when ES loads are supplied by the standby buses (Ref.1). Based on calculations, 4.155 kV is the minimum voltage that will ensure proper operation of loads during ES actuation.

This system is only required to be OPERABLE when the unit is in MODES 1, 2, 3, and 4 and the standby buses are energized without being electrically separated from the grid and offsite loads. System design is to provide protection for ES components caused by voltage droop due to inrush as the unit connects to the standby buses. The system is not a substitute for the dedicated line from Lee Gas Turbines.

The Lee Feeder breakers (SL) have no automatic close functions. However, this system does provide additional flexibility for the Station electrical system and operators in available power source options.

The EPSL CT-5 Degraded Grid Voltage Protection satisfies Criterion 3 of 10 CFR 50.36 (Ref. 2).

BASES (continued)

LCO

Three CT-5 degraded grid voltage sensing relay channels are required to be OPERABLE. Failure of one channel reduces the reliability of the function. The requirement for three channels to be OPERABLE ensures that two channels will remain OPERABLE if a failure has occurred in one channel. The remaining voltage sensing channels can perform the safety function.

Two channels of the CT-5 Degraded Grid Voltage Protection Actuation Logic function are required to be OPERABLE. The requirement for two channels to be OPERABLE ensures that one channel will remain OPERABLE if a failure has occurred in one channel. The remaining channel can perform the safety function.

APPLICABILITY

The CT-5 DGPS functions are required to be OPERABLE in MODES 1, 2, 3, and 4 when standby buses are energized without being electrically separated from grid or loads to ensure adequate voltage protection should a unit be transferred to the standby bus during an event requiring an ES actuation.

The EPSL CT-5 DGVP functions are not required to be OPERABLE in MODES 5 and 6 since more time is available for the operator to respond to a loss of power event.

ACTIONS

A.1

If one CT-5 DGVP voltage sensing relay channel is inoperable, the channel must be placed in trip within 72 hours. Tripping the affected channel places the function in a one-out-of-two configuration. Operation in this configuration may continue indefinitely since the DGVP function is capable of performing its DGVP function in the presence of any single random failure. With one channel inoperable, the remaining voltage sensing channels are capable of providing the DGVP function. The 72 hour completion time is based on engineering judgement taking into consideration the infrequency of actual grid system voltage degradation, and the probability of an event requiring an ES actuation.

B.1

If one CT-5 DGVP actuation logic channel is inoperable, the actuation logic channel must be restored to OPERABLE status within 72 hours. With one actuation logic channel inoperable, the remaining actuation logic channel is

BASES

ACTIONS

B.1 (continued)

capable of providing the CT-5 DGVP function. The 72 hour completion time is based on engineering judgement taking into consideration the infrequency of actual grid system voltage degradation and the probability of an event requiring an ES actuation.

C.1 and C.2

If two or more voltage sensing relay channels or two actuation logic channels are inoperable, automatic protection from degraded grid voltage for the standby buses powered from the 100 kV transmission system is not available. Continued operation is allowed provided that the SL breakers are opened within one hour.

Additionally, with the Required Action and associated Completion Time of Condition A or B not met, the SL breakers must be opened within one hour. This arrangement provides a high degree of reliability for the emergency power system. The one hour Completion Time is based on engineering judgement taking into consideration the infrequency of actual grid system voltage degradation and the probability of an event requiring an ES actuation.

SURVEILLANCE REQUIREMENTS

SR 3.3.20.1

A CHANNEL FUNCTIONAL TEST is performed on each CT-5 DGVP voltage sensing channel and each CT-5 DGVP actuation logic channel to ensure the entire channel will perform its intended function. Any setpoint adjustments shall be consistent with the assumptions of the setpoint analysis. The Frequency of 18 months is based on engineering judgment and operating experience that determined testing on an 18 month interval provides reasonable assurance that the circuitry is available to perform its safety function.

SR 3.3.20.2

A CHANNEL CALIBRATION is a complete check of the instrument channel, including the sensor. The test verifies that the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift to ensure that the instrument channel remains operational

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.20.2 (continued)

between successive tests. CHANNEL CALIBRATION shall find that measurement errors and bistable setpoint errors are within the assumptions of the setpoint analysis. CHANNEL CALIBRATIONS must be performed consistent with the assumptions of the setpoint analysis.

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

REFERENCES

1. UFSAR, Chapter 8.

2. 10 CFR 50.36.

## B 3.3 INSTRUMENTATION

### B 3.3.21 Emergency Power Switching Logic (EPSL) Keowee Emergency Start Function

BASES

BACKGROUND

The Keowee Emergency Start function of EPSL provides a start signal to the two on-site emergency power sources and sets up controls for the emergency mode. There are two channels of the Emergency Start function. Each channel is capable of starting both Keowee units and activating the controls for the emergency mode.

The Emergency Start channels 1 and 2 are actuated from Engineered Safeguards channels 1 and 2 respectively. The Emergency Start channels can also be activated manually from each control room (i.e., two emergency start switches in the Unit 1 and 2 control room and two emergency start switches in the Unit 3 control room) or cable spread rooms. There are two independent channels associated with each Oconee unit.

During a loss-of-coolant accident (LOCA) with a simultaneous loss of offsite power, the Keowee Emergency Start function of EPSL sends a start signal to both Keowee units. Logic is also actuated that ensures separation of both Keowee units from the system grid. Connection of the Keowee Unit aligned to the overhead power path is allowed only after a separate logic sequence (indicating switchyard isolation logic is complete which is not associated with the Keowee Emergency Start function) verifies the yellow bus is separated from the grid.

The Keowee Emergency Start function also disables non critical protective interlocks and trips associated with the Keowee generators. This ensures the generators can remain available as an emergency power source despite minor failures or malfunctions.

The Keowee Emergency Start circuitry is designed such that no single failure can prevent an Emergency Start signal from reaching the Keowee units. Each channel is independent of the other and only one channel is required to perform the entire safety function.

APPLICABLE SAFETY ANALYSES

The EPSL Keowee Emergency Start function is required for the engineered safeguards (ES) equipment to function in any accident with a loss of offsite power. The limiting accident for the EPSL voltage sensing circuits is a loss-of-coolant accident (LOCA) with a simultaneous loss of offsite power (Ref. 1).

BASES

APPLICABLE SAFETY ANALYSES (continued)

The EPSL Keowee Emergency Start Function satisfies Criterion 3 of 10 CFR 50.36 (Ref. 2).

LCO

Two channels of the Keowee Emergency Start function are required to be OPERABLE. Failure of one channel reduces the reliability of the function.

The requirement for two channels to be OPERABLE ensures that one channel will remain OPERABLE if a failure has occurred. The remaining channel can perform the safety function.

APPLICABILITY

The EPSL Keowee Emergency Start function is required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure that power is provided from AC Sources to the AC Distribution system within the time assumed in the accident analyses.

The EPSL Keowee Emergency Start function is not required to be OPERABLE in MODES 5 and 6 since more time is available for the operator to respond to a loss of power event.

ACTIONS

A.1

If one channel is inoperable, then a failure of the other channel could prevent starting the Keowee units. With one channel inoperable, the remaining channel is capable of providing the Keowee Emergency Start function. The 72 hour Completion Time is considered appropriate based on engineering judgement taking into consideration the time required to complete the required action.

Required Action A.1 is modified by a Note which indicates that the Completion Time is reduced when in Condition L of LCO 3.8.1. Condition L limits the Completion Time for restoring an inoperable channel to 4 hours when emergency power source(s) or offsite power source(s) are inoperable for extended time periods or for specific reasons.

B.1 and B.2

With the Required Action and associated Completion Time 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 in 12

BASES

ACTIONS

B.1 and B.2 (continued)

hours and to MODE 5 within 84 hours. The allowed Completion Times are reasonable, based on operating experience, to allow for a controlled shutdown.

C.1

With both channels of the Keowee Emergency Start function inoperable then both Keowee Hydro Units must be declared inoperable immediately. The appropriate Required Actions will be implemented in accordance with LCO 3.8.1, "AC Sources–Operating."

SURVEILLANCE REQUIREMENTS

SR 3.3.21.1

A CHANNEL FUNCTIONAL TEST is performed on each Keowee Emergency Start channel to ensure the channel will perform its function during an automatic transfer of the Main Feeder Buses to the Startup Transfer, Standby Buses, and retransfer to the Startup Transformers. The Frequency of 18 months is based on engineering judgment and operating experience that determined testing on an 18 month interval provides reasonable assurance that the circuitry is available to perform its safety function.

REFERENCES

1. UFSAR, Chapters 6 and 15.

2. 10 CFR 50.36.

## B 3.3 INSTRUMENTATION

### B 3.3.22 Emergency Power Switching Logic (EPSL) Manual Keowee Emergency Start Function

BASES

**BACKGROUND**

The Keowee Emergency Start function of EPSL provides a start signal to the two on-site emergency power sources and sets up controls for the emergency mode. There are two channels of the Emergency Start function. Each channel is capable of starting both Keowee units and activating the controls for the emergency mode.

The Emergency Start channels 1 and 2 are actuated from Engineered Safeguards channels 1 and 2 respectively. The Emergency Start channels can also be activated manually from each control room (i.e., two emergency start switches in the Unit 1 and 2 control room and two emergency start switches in the Unit 3 control room) or cable spread rooms. There are two independent channels associated with each Oconee unit. The two emergency start switches in the ONS Unit 1 and 2 control room initiate through the ONS Unit 1 emergency start channels. Neither start switch provides a signal to the ONS Unit 2 emergency start channels. Therefore, neither can be used to satisfy the LCO requirements for ONS Unit 2. The keylock switches on the emergency start panels in its cable spread room are used to satisfy the ONS Unit LCO requirement.

**APPLICABLE SAFETY ANALYSES**

The OPERABILITY of the Manual Keowee Emergency Start Function during MODES 5 and 6 and during movement of irradiated fuel assemblies ensures that adequate AC electrical power is provided to mitigate events postulated during shutdown, such as a fuel handling accident.

The EPSL Manual Keowee Emergency Start Function satisfies Criterion 3 of 10 CFR 50.36 (Ref. 1).

**LCO**

One channel of the Manual Keowee Emergency Start function, consisting of a manual initiation switch and an Emergency Start channel, is required to be OPERABLE. The emergency start switches in the Unit 1 and 2 Control Room cannot be credited as a manual initiation switch for Unit 2.

BASES (continued)

APPLICABILITY

The Manual Keowee Emergency Start function required to be OPERABLE in MODES 5 and 6 and during movement of irradiated fuel assemblies provides assurance that:

a. Systems needed to provide adequate coolant inventory makeup are available for the irradiated fuel assemblies;

b. Systems needed to mitigate a fuel handling accident are available;

c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

ACTIONS

A.1

If the required Manual Keowee Emergency Start channel is inoperable, both Keowee Hydro Units must be declared inoperable immediately. Therefore LCO 3.8.2 is entered immediately, and the required Completion Times for the appropriate Required Actions apply without delay.

SURVEILLANCE REQUIREMENTS

SR 3.3.22.1

A CHANNEL FUNCTIONAL TEST is performed on the required Manual Keowee Emergency Start channel to ensure the channel will perform its function. The Frequency of 12 months is based on engineering judgment and operating experience that determined testing on a 12 month interval provides reasonable assurance that the circuitry is available to perform its safety function.

REFERENCES

1. 10 CFR 50.36.

## B 3.3 INSTRUMENTATION

### B 3.3.23 Main Feeder Bus Monitor Panel (MFBMP)

BASES

BACKGROUND

The Main Feeder Bus Monitor Panel (MFBMP) uses three undervoltage relays per bus to detect low voltage. The undervoltage relay contacts are arranged to ensure any 2 of 3 phases tripped will de-energize the time-delay relays to start the actuation sequence. There are two time-delay relays and two auxiliary relays per channel. The 20 second time-delay relays de-energize auxiliary relays (TX) to cause load shed, Keowee Emergency start, and through an interposing relay, RX1 or 2, provide permissives to the SK breakers and Retransfer-to-Standby logic. The two logic channels are completely redundant. Channel A provides input to Channel A Load shed/Transfer-to-Standby, Keowee Emergency Start A, and SK1/Retransfer-to-Startup via relay RX1. Channel B provides input to Channel B for the companion logic devices, i.e. Channel B Keowee Emergency Start, RX2 to SK2/Retransfer-to-Startup, and Channel B Load shed/Transfer-to-Standby. The Load shed logic also provides an auto-close permissive to the Startup breakers.

APPLICABLE SAFETY ANALYSES ANALYSES

In the event of a station blackout (SBO), power is required to be available in 4 hours. Operator actions are credited for SBO mitigation. Thus, in the event of a loss of offsite power (LOOP), it is acceptable to credit operator actions to restore power to the main feeder buses (MFBs). The MFBMP provides a convenient and automatic method of establishing safe and reliable power to the MFBs during non-ES events. The system is redundant to ES signals which actuate Load shed/Transfer-to-Standby, and Keowee Emergency Start. The system also arms the Retransfer-to-Startup logic and closure of the Startup and SK breakers indirectly. The MFBMP does not provide the only layer of protection in any DBE, but does provide defense-in-depth for any scenario which results in loss of power to the main feeder buses.

The MFBMP does not satisfy the criteria in 10 CFR 50.36 (Ref. 1).

LCO

All six of the undervoltage sensing relay channels (three per MFB) are required to be OPERABLE as a common input device to both channels of actuating logic. Inoperability of any undervoltage relay is defined as unable to trip. This condition reduces the logic for the given logic channel and

BASES

LCO (continued)

MFB to a two out of two logic. Loss of any two relay channels on a single bus defeats the entire logic of both logic channels. Each logic channel has two time-delay and auxiliary relays, one time delay relay and one auxiliary relay for each MFB. Both time delay relays and auxiliary relays for each logic channel must actuate for the associated channel to operate.

APPLICABILITY

The MFBMP functions are required to be OPERABLE in MODES 1, 2, 3, and 4 to coincide with requirements for ES and other support/protective systems used to ensure adequate power is available for core and containment protection.

The MFBMP functions are not required to be OPERABLE in MODES 5 and 6 since more time is available for the operator to respond to a loss of power event.

ACTIONS

A.1

If one MFBMP voltage sensing channel is inoperable on one or both MFBs, the channel must be placed in trip within 7 days. Tripping the affected channel places the function in a one-out-of-two configuration. Operation in this configuration may continue indefinitely since the MFBMP function is capable of performing its MFBMP function in the presence of a single failure. With one channel inoperable, the remaining channels are capable of providing the Degraded Grid Voltage Protection (DGVP) function. The 7 day completion time is based on engineering judgement and the probability of an event requiring power restoration to the main feeder buses.

The Condition is modified by a Note indicating that this condition may be entered independently for each set of channels associated with a main feeder bus. The Completion Time(s) of the inoperable channels are tracked separately from the time the Condition is entered for each main feeder bus.

B.1

If one MFBMP actuation logic channel is inoperable, the actuation logic channel must be restored to OPERABLE status within 7 days. With one actuation logic channel inoperable, the remaining actuation logic channel is capable of providing the MFBMP function. The 7 day completion time is based on engineering judgement and the availability of adequate time for operator response to a LOOP.

BASES

ACTIONS (continued)

C.1 and C.2

With two or more voltage sensing channels or both actuation logic channels inoperable, automatic protection for LOOP events is no longer available. This places additional burden on the operators, even though they are still the credible resource for restoring power in a LOOP event. EPSL response from ES events are not affected. Therefore, allowable time for this condition is limited to 24 hours. The completion time is based on engineering judgement and the availability of adequate time for operator response to a LOOP.

The Condition is modified by a Note indicating that this condition may be entered independently for each set of channels associated with a main feeder bus. The Condition may also be entered independently for inoperable logic channels or inoperable voltage sensing channels. The Completion Time(s) are tracked separately from the time the Condition is entered for each.

D.1

With the Required Action and associated Completion Time not met, Required Action D.1 specifies initiation of action described in Specification 5.6.6 that requires a written report to be submitted to the NRC. This report discusses the results of the root cause evaluation of the inoperability and identifies proposed restorative actions. This action is appropriate since the MFBMP does not provide the only layer of protection in any DBE, but does provide defense-in-depth for any scenario which results in loss of power to the Main Feeder Busses. Operator actions are credited for SBO mitigation. The Completion Time of "Immediately" for Required Action D.1 ensures the requirements of Specification 5.6.6 are initiated.

SURVEILLANCE REQUIREMENTS

SR 3.3.23.1

A CHANNEL FUNCTIONAL TEST is performed on each MFBMP voltage sensing channel and MFBMP actuation logic channel to ensure the MFBMP will perform its intended function. The Frequency of 18 months is based on engineering judgment and operating experience that determined testing on an 18 month interval provides reasonable assurance that the circuitry is available to perform its safety function.

BASES (continued)

REFERENCES 1. 10 CFR 50.36.

# B 3.4 REACTOR COOLANT SYSTEM (RCS)

## B 3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

BASES

BACKGROUND

These Bases address requirements for maintaining RCS pressure, temperature, and flow rate within limits assumed in the safety analyses. The safety analyses (Ref. 1) of normal operating conditions and anticipated transients assume initial conditions within the normal steady state envelope. The limits placed on DNB related parameters ensure that these parameters will not be less conservative than were assumed in the analyses and thereby provide assurance that the minimum departure from nucleate boiling ratio (DNBR) will meet the required criteria for each of the transients analyzed.

The LCO for minimum RCS pressure is consistent with operation within the nominal operating envelope and is above that used as the initial pressure in the analyses. A pressure greater than the minimum specified will produce a higher minimum DNBR. A pressure lower than the minimum specified will cause the unit to approach the DNB limit.

The LCO for maximum RCS coolant loop average temperature is consistent with full power operation within the nominal operating envelope and is lower than the initial loop average temperature in the analyses. A loop average temperature lower than that specified will produce a higher minimum DNBR. A loop average temperature higher than that specified will cause the unit to approach the DNB limit.

The RCS flow rate is not expected to vary during operation with all pumps running. The LCO for the minimum RCS flow rate corresponds to that assumed for the DNBR analyses. A higher RCS flow rate will produce a higher DNBR. A lower RCS flow will cause the unit to approach the DNB limit.

APPLICABLE SAFETY ANALYSES

The requirements of LCO 3.4.1 represent the initial conditions for DNB limited transients analyzed in the plant safety analyses (Ref. 1). The safety analyses have shown that transients initiated from the limits of this LCO will meet the DNBR criterion of ≥ 1.18 for BWC correlation or an equally valid limit when the statistical DNBR limit is employed (SCD methodology). This is the acceptance limit for the RCS DNBR parameters.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Changes to the facility that could impact these parameters must be assessed for their impact on the DNBR criterion. The transients analyzed for include loss of coolant flow events and dropped control rod events and control rod withdrawal events. A key assumption for the analysis of these events is that the core power distribution is within the limits of LCO 3.2.1, "Regulating Rod Position Limits," LCO 3.2.2, "AXIAL POWER IMBALANCE Operating Limits," and LCO 3.2.3, "QUADRANT POWER TILT (QPT)."

The normal operating band for RCS pressure is between 2125 psig and 2155 psig as measured at the hot leg pressure tap. The safety analyses assume a core exit pressure that is based on the measured pressure and concurrent pressure losses between the two locations. The pressure losses are a function of the loop flow rate, thus different values are allowed for 4 or 3 RCP operation.

Analyses have been performed to establish the pressure, temperature, and flow rate requirements for three pump and four pump operation. These limits are specified in the COLR. The flow limits for three pump operation are substantially lower than for four pump operation. To meet the DNBR criterion, a corresponding maximum power limit is required (see Bases for LCO 3.4.4, "RCS Loops–MODES 1 and 2").

Another set of limits on DNBR related parameters is provided in Safety Limit (SL) 2.1.1, "Reactor Core SLs." Those limits are less restrictive than the limits of LCO 3.4.1, but violation of an SL merits a stricter, more severe Required Action.

The RCS DNB limits satisfy Criterion 2 of 10 CFR 50.36 (Ref. 2).

LCO

This LCO specifies limits on the monitored process variables: RCS loop (hot leg) pressure, RCS loop average temperature, and RCS total flow rate to ensure that the core operates within the limits assumed for the plant safety analyses. Operating within these limits will result in meeting DNBR criteria in the event of a DNB limited transient.

The pressure limits are applied to the loop with the highest pressure. The temperature limits are applied to the loop with the lowest loop average temperature for the condition in which there is a 0°F $\Delta T_c$ setpoint.

BASES (continued)

APPLICABILITY

In MODE 1 during steady state operation, the limits on RCS loop pressure, RCS loop average temperature, and RCS flow rate must be maintained with four pump or three pump operation in order to ensure that DNBR criteria will be met in the event of an unplanned loss of forced coolant flow or other DNB limited transient. In all other MODES the power level is low enough so that DNB is not a concern. Steady state operation, for the purposes of this specification, is defined as operation within a 4% (e.g., 88% - 92% RTP) power band for ≥ 4 hours.

ACTIONS

A.1

Loop pressure and loop average coolant temperature are controllable and measurable parameters. With one or both of these parameters not within the LCO limits, action must be taken to restore the parameters. RCS flow rate is not a controllable parameter and is not expected to vary during steady state four pump or three pump operation. However, if the flow rate is below the LCO limit, the parameter must be restored to within limits or power must be reduced as required in Required Action B.1, to restore DNBR margin and eliminate the potential for violation of the accident analysis bounds. The 2 hour Completion Time for restoration of the parameters provides sufficient time to adjust unit parameters, determine the cause for the off normal condition, and restore the readings within limits. The Completion Time is based on plant operating experience.

B.1

If the Required Action and associated Completion Time are 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 2 within 12 hours. In MODE 2, the reduced power condition eliminates the potential for violation of the accident analysis bounds.

The 12 hour Completion Time is reasonable, based on operating experience, to reduce power in an orderly manner.

SURVEILLANCE REQUIREMENTS

SR 3.4.1.1

Since Required Action A.1 allows a Completion Time of 2 hours to restore parameters that are not within limits, the 12 hour Surveillance Frequency for loop (hot leg) pressure is sufficient to ensure that the pressure can be

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.1.1 (continued)

restored to a normal operation, steady state condition following load changes and other expected transient operations. The RCS pressure value specified in the COLR is dependent on the number of pumps in operation and has been adjusted to account for the pressure loss difference between the core exit and the measurement location. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess potential degradation and to verify operation is within safety analysis assumptions. A Note has been added to indicate the pressure limits for three pumps operating is applied to the loop with the highest pressure.

SR 3.4.1.2

Since Required Action A.1 allows a Completion Time of 2 hours to restore parameters that are not within limits, the 12 hour Surveillance Frequency for loop average temperature is sufficient to ensure that the RCS coolant temperature can be restored to a normal operation, steady state condition following load changes and other expected transient operations. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess potential degradation and to verify that operation is within safety analysis assumptions. A Note has been added to indicate the temperature limits for three pumps operating are applied to the loop with the lowest loop average temperature for the condition in which there is a 0°F $\Delta$Tc setpoint.

SR 3.4.1.3

The 12 hour Surveillance Frequency for RCS total flow rate is performed using the installed flow instrumentation. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess potential degradation and to verify that operation is within safety analysis assumptions.

SR 3.4.1.4

Measurement of RCS total flow rate by performance of a calorimetric heat balance once every 18 months allows the installed RCS flow instrumentation to be calibrated and verifies that the actual RCS flow is greater than or equal to the minimum required RCS flow rate specified in the COLR.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.1.4 (continued)

The Frequency of 18 months reflects the importance of verifying flow after a refueling outage when the core has been altered or RCS flow characteristics may have been modified, which may have caused change of flow. The Surveillance is modified by a Note that indicates the SR does not need to be performed until 7 days after stable thermal conditions are established at higher power levels. The Note is necessary to allow measurement of the flow rate at normal operating conditions at power in MODE 1. The Surveillance cannot be performed at low power or in MODE 2 or below because at low power the $\Delta T$ across the core may be too small to provide meaningful test results.

REFERENCES

1. UFSAR, Chapter 15.

2. 10 CFR 50.36

## B 3.4 REACTOR COOLANT SYSTEM (RCS)

### B 3.4.2 RCS Minimum Temperature for Criticality

BASES

**BACKGROUND**

Establishing the value for the minimum temperature for reactor criticality is based upon considerations for:

a. Operation within the existing instrumentation ranges and accuracies;

b. Operation with reactor vessel above its minimum nil ductility reference temperature when the reactor is critical.

The reactor coolant moderator temperature coefficient used in core operating and accident analysis is typically defined for the normal operating temperature range (532°F to 579°F). The Reactor Protection System (RPS) receives inputs from the narrow range hot leg temperature detectors, which have a range of 520°F to 620°F. The Integrated Control System controls average temperature using inputs of the same range. The nominal average temperature for making the reactor critical is 532°F. Safety and operating analyses for lower temperatures have not been made.

**APPLICABLE SAFETY ANALYSES**

There are no accident analyses that dictate the minimum temperature for criticality, but all zero power safety analyses assume initial temperatures near the 525°F limit (Ref. 1).

The RCS minimum temperature for criticality satisfies Criterion 2 of 10 CFR 50.36 (Ref. 2).

**LCO**

The purpose of the LCO is to prevent criticality below the normal operating regime (532°F to 579°F) and to prevent operation in an unanalyzed condition.

The LCO limit of 525°F has been selected to be within the instrument indicating range (520°F to 620°F). The limit is also set slightly below the lowest power range operating temperature (532°F).

BASES

APPLICABILITY

The reactor has been designed and analyzed to be critical in MODES 1 and 2 only and in accordance with this Specification. Criticality is not permitted in any other MODE. Therefore, this LCO is applicable in MODE 1 and MODE 2 when $k_{eff} \geq 1.0$.

ACTIONS

A.1

With the reactor coolant loop average temperature below 525°F, 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 MODE 2 with $k_{eff} < 1.0$ in 30 minutes. Rapid reactor shutdown can be readily and practically achieved in a 30 minute period. The Completion Time reflects the ability to perform this Action and maintain the unit within the analyzed range. If the reactor coolant loop average temperature can be restored within the 30 minute time period, shutdown is not required.

SURVEILLANCE REQUIREMENTS

SR 3.4.2.1

The reactor coolant loop average temperature is required to be verified above 525°F every 30 minutes. The 30 minute time period is frequent enough to prevent inadvertent violation of the LCO. The frequency has been modified by a Note indicating this SR is only required when any RCS loop average temperature is < 530°F. While Surveillance is required whenever the reactor is critical and temperature is below 530°F, in practice the Surveillance is most appropriate during the period when the reactor is brought critical.

REFERENCES

1. UFSAR, Chapter 15.

2. 10 CFR 50.36.

## B 3.4 REACTOR COOLANT SYSTEM (RCS)

### B 3.4.3 RCS Pressure and Temperature (P/T) Limits

BASES

BACKGROUND

All components of the RCS are designed to withstand effects of cyclic loads due to system pressure and temperature changes. These loads are introduced by startup (heatup) and shutdown (cooldown) operations, power transients, and reactor trips. This LCO limits the pressure and temperature changes during RCS heatup and cooldown, within the design assumptions and the stress limits for cyclic operation.

Figures 3.4.3-1 through 3.4.3-9 contain P/T limit curves for heatup, cooldown, and leak and hydrostatic (LH) testing. Tables 3.4.3-1 and 3.4.3-2 contain data for the maximum rate of change of reactor coolant temperature. The minimum temperature indicated in the P/T limit curves and tables of 60°F is the lowest unirradiated nil ductility reference temperature ($RT_{NDT}$) of all materials in the reactor vessel. This temperature (60°F) is the minimum allowable reactor pressure vessel temperature if any head closure stud is not fully detensioned.

Figures 3.4.3-1, 3.4.3-2, 3.4.3-4, 3.4.3-5, 3.4.3-7 and 3.4.3-8 define an acceptable region for normal operation. The usual use of the curves is operational guidance during heatup or cooldown maneuvering, when pressure and temperature indications are monitored and compared to the applicable curve to determine that operation is within the allowable region.

The LCO establishes operating limits that provide a margin to brittle failure of the reactor vessel and piping of the reactor coolant pressure boundary (RCPB). The vessel is the component most subject to brittle failure, and the LCO limits apply mainly to the vessel. The limits do not apply to the pressurizer, which has different design characteristics and operating functions.

10 CFR 50, Appendix G (Ref. 1), requires the establishment of P/T limits for material fracture toughness requirements of the RCPB materials. Reference 1 requires an adequate margin to brittle failure during normal operation, anticipated operational occurrences, and system hydrostatic tests. It mandates the use of the American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel Code, Section III, Appendix G (Ref. 2).

Linear elastic fracture mechanics (LEFM) methodology is used to determine the stresses and material toughness at locations within the RCPB. The LEFM methodology follows the guidance given by 10 CFR 50, Appendix G; ASME Code, Section III, Appendix G; and Regulatory Guide 1.99 (Ref. 3).

BASES

BACKGROUND (continued)

Material toughness properties of the ferritic materials of the reactor vessel are determined in accordance with ASTM E 185 (Ref. 4), and additional reactor vessel requirements. These properties are then evaluated in accordance with Reference 2.

The actual shift in the nil ductility reference temperature ($RT_{NDT}$) of the vessel material will be established periodically by evaluating the irradiated reactor vessel material specimens, in accordance with ASTM E 185 (Ref. 5) and Appendix H of 10 CFR 50 (Ref. 5). The operating P/T limit curves will be adjusted, as necessary, based on the evaluation findings and the recommendations of Reference 2.

The P/T limit curves are composite curves established by superimposing limits derived from stress analyses of those portions of the reactor vessel and head that are the most restrictive. At any specific pressure, temperature, and temperature rate of change, one location within the reactor vessel will dictate the most restrictive limit. Across the span of the P/T limit curves, different locations are more restrictive, and, thus, the curves are composites of the most restrictive regions.

The heatup curve represents a different set of restrictions than the cooldown curve because the directions of the thermal gradients through the vessel wall are reversed. The thermal gradient reversal alters the location of the tensile stress between the outer and inner walls.

The calculation to generate the LH testing curve uses different safety factors (per Ref. 2) than the heatup and cooldown curves.

The P/T limit curves and associated temperature rate of change limits are developed in conjunction with stress analyses for large numbers of operating cycles and provide conservative margins to nonductile failure. Although created to provide limits for these specific normal operations, the curves also can be used to determine if an evaluation is necessary for an abnormal transient.

The criticality limit curve includes the Reference 1 requirement that it be 40°F above the heatup curve or the cooldown curve, and not less than the minimum permissible temperature for LH testing. However, the criticality curve is not operationally limiting; a more restrictive limit exists in LCO 3.4.2, "RCS Minimum Temperature for Criticality."

The consequence of violating the LCO limits is that the RCS has been operated under conditions that can result in brittle failure of the RCPB, possibly leading to a nonisolable leak or loss of coolant accident. In the event these limits are exceeded, an evaluation must be performed to determine the effect on the structural integrity of the RCPB components. The ASME Code, Section XI, Appendix E (Ref. 6) provides a recommended methodology for evaluating an operating event that causes an excursion outside the limits.

BASES

APPLICABLE SAFETY ANALYSES

The P/T limits are not derived from accident analyses. They are prescribed during normal operation to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and cause nonductile failure of the RCPB, an unanalyzed condition. Reference 1 establishes the methodology for determining the P/T limits. Since the P/T limits are not derived from any accident analysis, there are no acceptance limits related to the P/T limits. Rather, the P/T limits are acceptance limits themselves since they preclude operation in an unanalyzed condition.

RCS P/T limits satisfy Criterion 2 of 10 CFR 50.36 (Ref. 7).

LCO

The three elements of this LCO are:

a. The limit curves for heatup and cooldown,

b. Limits on the rate of change of temperature, and

c. Allowable RC pump combinations.

The LCO is modified by three Notes. Note 1 states that for leak tests of the RCS and leak tests of connected systems where RCS pressure and temperature are controlling, the RCS may be pressurized to the limits of the specified figures. Note 2 states that for thermal steady state hydro tests required by ASME Section XI RCS may be pressurized to the limits Specification 2.1.2 and the specified figures. The limits on the rate of change of reactor coolant temperature RCS P/T Limits are the same ones used for normal heatup and cooldown operations. Note 3 states the RCS P/T limits are not applicable to the pressurizer.

The LCO limits apply to all components of the RCS, except the pressurizer. These limits define allowable operating regions and permit a large number of operating cycles while providing a wide margin to nonductile failure.

Table 3.4.3-1 includes temperature rate of change limits with allowable pump combinations for RCS heatup while Table 3.4.3-2 includes temperature rate of change limits with allowable pump combinations for RCS cooldown. The breakpoints between temperature rate of change limits in these two tables are selected to limit reactor vessel thermal gradients to acceptable limits. The breakpoint between allowable pump combinations was selected based on operational requirements and are used to determine the change of RCS pressure associated with the change in number of operating reactor coolant pumps.

BASES

LCO (continued)

The limits for the rate of change of temperature control the thermal gradient through the vessel wall and are used as inputs for calculating the heatup, cooldown, and LH P/T limit curves. Thus, the LCO for the rate of change of temperature restricts stresses caused by thermal gradients and also ensures the validity of the P/T limit curves.

The limits on allowable RC pump combinations controls the pressure differential between the vessel wall and the pressure measurement point and are used as inputs for calculating the heatup, cooldown and LH P/T limit curves. Thus, the LCO for the allowable RC pump combinations restricts the pressure at the vessel wall and ensures the validity of the P/T limit curves.

Violating the LCO limits places the reactor vessel outside of the bounds of the stress analyses and can increase stresses in other RCPB components. The consequences depend on several factors, as follows:

a. The severity of the departure from the allowable operating P/T regime or the severity of the rate of change of temperature;

b. The length of time the limits were violated (longer violations allow the temperature gradient in the thick vessel walls to become more pronounced); and

c. The existences, sizes, and orientations of flaws in the vessel material.

APPLICABILITY

The RCS P/T limits Specification provides a definition of acceptable operation for prevention of nonductile failure in accordance with 10 CFR 50, Appendix G (Ref. 1). Although the P/T limits were developed to provide guidance for operation during heatup or cooldown (MODES 3, 4, and 5) or LH testing, their applicability is at all times in keeping with the concern for nonductile failure. The limits do not apply to the pressurizer.

During MODES 1 and 2, other Technical Specifications provide limits for operation that can be more restrictive than or can supplement these P/T limits. LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits"; LCO 3.4.2, "RCS Minimum Temperature for Criticality"; and Safety Limit (SL) 2.1, "SLs," also provide operational restrictions for pressure and temperature and maximum pressure. MODES 1 and 2 are above the temperature range of concern for nonductile failure, and stress analyses have been performed for normal maneuvering profiles, such as power ascension or descent.

BASES

ACTIONS

A.1 and A.2

Operation outside the P/T limits during MODE 1, 2, 3, or 4 must be corrected so that the RCPB is returned to a condition that has been verified by stress analyses.

The 30 minute Completion Time reflects the urgency of restoring the parameters to within the analyzed range. Most violations will not be severe, and the activity can be accomplished in this time in a controlled manner.

Besides restoring operation to within limits, an evaluation is required to determine if RCS operation can continue. The evaluation must verify the RCPB integrity remains acceptable and must be completed before continuing operation. Several methods may be used, including comparison with pre-analyzed transients in the stress analyses, new analyses, or inspection of the components. The evaluation must be completed, documented, and approved in accordance with established plant procedures and administrative controls.

ASME Code, Section XI, Appendix E (Ref. 6) may be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline. The evaluation must extend to all components of the RCPB.

The 72 hour Completion Time is reasonable to accomplish the evaluation. The evaluation for a mild violation is possible within this time, but more severe violations may require special, event specific stress analyses or inspections. A favorable evaluation must be completed before continuing to operate.

Condition A is modified by a Note requiring Required Action A.2 to be completed whenever the Condition is entered. The Note emphasizes the need to perform the evaluation of the effects of the excursion outside the allowable limits. Restoration alone per Required Action A.1 is insufficient because higher than analyzed stresses may have occurred and may have affected the RCPB integrity.

B.1 and B.2

If a Required Action and associated Completion Time of Condition A are not met, the unit must be brought to a lower MODE because: (a) the RCS remained in an unacceptable pressure and temperature region for an extended period of increased stress, or (b) a sufficiently severe event caused entry into an unacceptable region. Either possibility indicates a need for more careful examination of the event, best accomplished with the RCS at reduced pressure and temperature. With reduced pressure and temperature conditions, the possibility of propagation of undetected flaws is decreased.

BASES

ACTIONS

B.1 and B.2 (continued)

If the required restoration activity cannot be accomplished within 30 minutes, Required Action B.1 and Required Action B.2 must be implemented to reduce pressure and temperature.

If the required evaluation for continued operation cannot be accomplished within 72 hours, or the results are indeterminate or unfavorable, action must proceed to reduce pressure and temperature as specified in Required Actions B.1 and B.2. A favorable evaluation must be completed and documented before returning to operating pressure and temperature conditions. However, if the favorable evaluation is accomplished while reducing pressure and temperature conditions, a return to power operation may be considered without completing Required Action B.2.

Pressure and temperature are reduced by bringing the unit to MODE 3 within 12 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required MODE from full power conditions in an orderly manner and without challenging unit systems.

C.1 and C.2

Actions must be initiated immediately to correct operation outside of the P/T limits at times other than MODE 1, 2, 3, or 4, so that the RCPB is returned to a condition that has been verified acceptable by stress analysis.

The immediate Completion Time reflects the urgency of initiating action to restore the parameters to within the analyzed range. Most violations will not be severe, and the activity can be accomplished within this time in a controlled manner.

In addition to restoring operation to within limits, an evaluation is required to determine if RCS operation can continue. The evaluation must verify that the RCPB integrity remains acceptable and must be completed prior to entry into MODE 4. Several methods may be used, including comparison with pre-analyzed transients in the stress analysis, or inspection of the components.

ASME Code, Section XI, Appendix E (Ref. 6), may also be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline.

BASES

ACTIONS

C.1 and C.2 (continued)

Condition C is modified by a Note requiring Required Action C.2 to be completed whenever the Condition is entered. The Note emphasizes the need to perform the evaluation of the effects of the excursion outside the allowable limits. Restoration alone, per Required Action C.1, is insufficient because higher than analyzed stresses may have occurred and may have affected RCPB integrity.

SURVEILLANCE REQUIREMENTS

SR 3.4.3.1

Verification that operation is within limits is required every 30 minutes when RCS pressure or temperature conditions are undergoing planned changes.

This Frequency is considered reasonable in view of the control room indication available to monitor RCS status. Thirty minutes permits assessment and correction for minor deviations within a reasonable time.

Surveillance for heatup, cooldown, or LH testing may be discontinued when the definition given in the relevant plant procedure for ending the activity is satisfied.

This SR is modified by a Note that requires this SR to be performed only during system heatup, cooldown, and LH testing.

REFERENCES

1. 10 CFR 50, Appendix G.
2. ASME, Boiler and Pressure Vessel Code, Section III, Appendix G.
3. Regulatory Guide 1.99, Revision 2, May 1988.
4. ASTM E 185-82, July 1982.
5. 10 CFR 50, Appendix H.
6. ASME, Boiler and Pressure Vessel Code, Section XI, Appendix E.
7. 10 CFR 50.36.

# B 3.4 REACTOR COOLANT SYSTEM (RCS)

## B 3.4.4 RCS Loops – MODES 1 and 2

BASES

BACKGROUND

The primary function of the reactor coolant is removal of the heat generated in the fuel due to the fission process, and transfer of this heat, via the steam generators (SGs), to the secondary plant.

The secondary functions of the reactor coolant include:

a. Moderating the neutron energy level to the thermal state, to increase the probability of fission;

b. Improving the neutron economy by acting as a reflector;

c. Carrying the soluble neutron poison, boric acid;

d. Providing a second barrier against fission product release to the environment; and

e. Removing the heat generated in the fuel due to fission product decay following a unit shutdown.

The RCS configuration for heat transport uses two RCS loops. Each RCS loop contains an SG and two reactor coolant pumps (RCPs). An RCP is located in each of the two SG cold legs. The pump flow rate has been sized to provide core heat removal with appropriate margin to departure from nucleate boiling (DNB) during power operation and for anticipated transients originating from power operation. This Specification requires two RCS loops with either three or four pumps to be in operation. With three pumps in operation the reactor power level is restricted to 75% RTP to preserve the core power to flow relationship, thus maintaining the margin to DNB. The intent of the specification is to require core heat removal with forced flow during power operation. Specifying the minimum number of pumps is an effective technique for designating the proper forced flow rate for heat transport, and specifying two loops provides for the needed amount of heat removal capability for the allowed power levels. Specifying two RCS loops also provides the minimum necessary paths (two SGs) for heat removal.

The Reactor Protection System (RPS) trip setpoint based on flux/flow/imbalance is automatically reduced when one pump is taken out of service; manual resetting is not necessary.

BASES (continued)

APPLICABLE SAFETY ANALYSES

Safety analyses contain various assumptions for the accident analyses initial conditions including: RCS pressure, RCS temperature, reactor power level, core parameters, and safety system setpoints. The important aspect for this LCO is the reactor coolant forced flow rate, which is represented by the number of pumps in service.

Both transient and steady state analyses have been performed to establish the effect of flow on DNB. The transient or accident analysis for the plant has been performed assuming either three or four pumps are in operation. The majority of the plant safety analysis is based on initial conditions at high core power or zero power. The analyses that are of most importance to RCP operation are the two pump coastdown, single pump locked rotor, and single pump broken shaft (Ref. 1).

Steady state DNB analysis has been performed for four, and three pump combinations. For four pump operation, the steady state DNB analysis, which generates the pressure and temperature protective limit (i.e., the departure from nucleate boiling ratio (DNBR) limit), assumes a maximum power level of 112% RTP. This is the design overpower condition for four pump operation. The 105.5% RTP value is the setpoint of the nuclear overpower (high flux) trip and is based on an analysis assumption that bounds possible instrumentation errors. The DNBR limit defines a locus of pressure and temperature points that result in a minimum DNBR greater than or equal to the critical heat flux correlation limit.

The three pump pressure temperature limit is tied to the steady state DNB analysis, which is evaluated each cycle. The flow used is the minimum allowed for three pump operation. The actual RCS flow rate will exceed the assumed flow rate. With three pumps operating, overpower protection is automatically provided by the power to flow ratio of the RPS nuclear overpower trip setpoint based on flux/flow/imbalance. The maximum power level for three pump operation is 75% RTP and is based on the three pump flow as a fraction of the four pump flow at full power.

Continued power operation with two RCPs removed from service is not allowed by this Specification.

RCS Loops – MODES 1 and 2 satisfy Criterion 2 of 10 CFR 50.36 (Ref. 2).

LCO

The purpose of this LCO is to require adequate forced flow for core heat removal. Flow is represented by the number of RCPs in operation in both RCS loops for removal of heat by the two SGs. To meet safety analysis acceptance criteria for DNB, four pumps are required at rated power; if only three pumps are available, power must be reduced.

BASES (continued)

APPLICABILITY

In MODES 1 and 2, the reactor is critical and has the potential to produce maximum THERMAL POWER. To ensure that the assumptions of the accident analyses remain valid, all RCS loops are required to be OPERABLE and in operation in these MODES to prevent DNB and core damage.

The decay heat production rate is much lower than the full power heat rate. As such, the forced circulation flow and heat sink requirements are reduced for lower, noncritical MODES as indicated by the LCOs for MODES 3, 4, and 5.

Operation in other MODES is covered by:

LCO 3.4.5, "RCS Loops – MODE 3";
LCO 3.4.6, "RCS Loops – MODE 4";
LCO 3.4.7, "RCS Loops – MODE 5, Loops Filled";
LCO 3.4.8, "RCS Loops – MODE 5, Loops Not Filled";
LCO 3.9.4, "Decay Heat Removal (DHR) and Coolant Circulation – High Water Level" (MODE 6); and
LCO 3.9.5, "Decay Heat Removal (DHR) and Coolant Circulation – Low Water Level" (MODE 6).

ACTIONS

A.1

If the requirements of the LCO are not met, the Required Action is to reduce power and bring the unit to MODE 3. This lowers power level and thus reduces the core heat removal needs and minimizes the possibility of violating DNB limits.

The Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging safety systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.4.1

This SR requires verification every 12 hours of the required number of loops in operation. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal while maintaining the margin to DNB. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess degradation and verify operation within safety analyses assumptions. In addition, control room indication and alarms will normally indicate loop status.

BASES (continued)

REFERENCES

1. UFSAR, Chapter 15.

2. 10 CFR 50.36.

## B 3.4 REACTOR COOLANT SYSTEM (RCS)

### B 3.4.5 RCS Loops – MODE 3

BASES

BACKGROUND

The primary function of the reactor coolant in MODE 3 is removal of decay heat and transfer of this heat, via the steam generators (SGs), to the secondary plant fluid. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid.

In MODE 3, reactor coolant pumps (RCPs) are used to provide forced circulation for heat removal during heatup and cooldown. The number of RCPs in operation will vary depending on operational needs, and the intent of this LCO is to provide forced flow from at least one RCP for core heat removal and transport. The flow provided by one RCP is adequate for heat removal and for boron mixing. However, two RCS loops are required to be OPERABLE to provide redundant paths for heat removal.

Reactor coolant natural circulation is not normally used; however, the natural circulation flow rate is sufficient for core cooling. If entry into natural circulation is required, the reactor coolant at the highest elevation of the hot leg must be maintained subcooled for single phase circulation. When in natural circulation, it is preferable to remove heat using both SGs to avoid idle loop stagnation that might occur if only one SG were in service. One generator will provide adequate heat removal. Boron reduction in natural circulation is prohibited because mixing to obtain a homogeneous concentration in all portions of the RCS cannot be ensured.

APPLICABLE SAFETY ANALYSES

No safety analyses are performed with initial conditions in MODE 3.

Failure to provide heat removal may result in challenges to a fission product barrier. The RCS loops are part of the primary success path that functions or actuates to prevent or mitigate an Accident or transient that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier.

RCS Loops – MODE 3 satisfy Criterion 3 of 10 CFR 50.36 (Ref. 1).

BASES (continued)

LCO

The purpose of this LCO is to require two loops to be available for heat removal thus providing redundancy. The LCO requires the two loops to be OPERABLE with the intent of requiring both SGs to be capable of transferring heat from the reactor coolant at a controlled rate. Forced reactor coolant flow is the required way to transport heat, although natural circulation flow provides adequate removal. A minimum of one running RCP meets the LCO requirement for one loop in operation.

The Note permits a limited period of operation without RCPs. All RCPs may not be in operation for ≤ 8 hours per 24 hour period for the transition to or from the Decay Heat Removal (DHR) System, and otherwise may be de-energized for ≤1 hour per 8 hour period. This means that natural circulation has been established. When in natural circulation, boron reduction is prohibited because an even concentration distribution throughout the RCS cannot be ensured. Core outlet temperature is to be maintained at least 10°F below the saturation temperature so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

In MODES 3, 4, and 5, it is sometimes necessary to stop all RCP or LPI pump forced circulation (e.g., change operation from one DHR loop to the other, to perform surveillance or startup testing, to perform the transition to and from DHR mode cooling, or to avoid operation below the RCP minimum net positive suction head limit). This is acceptable because natural circulation is adequate for heat removal, or the reactor coolant temperature can be maintained subcooled and boron stratification affecting reactivity control is not expected.

An OPERABLE RCS loop consists of at least one OPERABLE RCP and an SG that is capable of transferring decay heat to the secondary fluid. An RCP is OPERABLE if it is capable of being powered and is able to provide forced flow if required.

APPLICABILITY

In MODE 3, the heat load is lower than at power; therefore, one RCS loop in operation is adequate for transport and heat removal. A second RCS loop is required to be OPERABLE but not in operation for redundant heat removal capability.

BASES

APPLICABILITY (continued)

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops – MODES 1 and 2";
LCO 3.4.6, "RCS Loops – MODE 4";
LCO 3.4.7, "RCS Loops – MODE 5, Loops Filled";
LCO 3.4.8, "RCS Loops – MODE 5, Loops Not Filled";
LCO 3.9.4, "Decay Heat Removal (DHR) and Coolant Circulation – High Water Level" (MODE 6); and
LCO 3.9.5, "Decay Heat Removal (DHR) and Coolant Circulation – Low Water Level" (MODE 6).

ACTIONS

A.1

If one RCS loop is inoperable, redundancy for forced flow heat removal is lost. The Required Action is restoration of the RCS loop to OPERABLE status within a Completion Time of 72 hours. This time allowance is a justified period to be without the redundant nonoperating loop because a single loop in operation has a heat transfer capability greater than that needed to remove the decay heat produced in the reactor core.

B.1

If Required Action and associated Completion Time are not met, the unit must be brought to MODE 4. In MODE 4, the unit may be placed on the LPI System. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to achieve cooldown and depressurization from the existing unit conditions and without challenging unit systems.

C.1 and C.2

If no RCS loop is OPERABLE or a required RCS loop is not in operation, (no RCS loop is required to be in operation provided the conditions in the Note in the LCO section are met), all operations involving a reduction of RCS boron concentration must be immediately suspended. This is necessary because boron dilution requires forced circulation for proper homogenization. Action to restore one RCS loop to operation shall be immediately initiated and continued until one RCS loop is restored to operation and to OPERABLE status. The immediate Completion Time reflects the importance of maintaining operation for decay heat removal.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.5.1

This SR requires verification every 12 hours that the required number of loops and pumps is in operation. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess RCS loop status. In addition, control room indication and alarms will normally indicate loop status.

SR 3.4.5.2

Verification that the required number of RCPs are OPERABLE ensures that an additional RCS loop can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power availability to the required pump that is not in operation. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

REFERENCES

1. 10 CFR 50.36.

## B 3.4 REACTOR COOLANT SYSTEM (RCS)

### B 3.4.6 RCS Loops – MODE 4

BASES

BACKGROUND

In MODE 4, the primary function of the reactor coolant is the removal of decay heat and transfer of this heat to the steam generators (SGs) or LPI heat exchangers. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid.

In MODE 4, either reactor coolant pumps (RCPs) or LPI pumps can be used for coolant circulation. The number of pumps in operation can vary to suit the operational needs. The intent of this LCO is to provide forced flow from at least one RCP or one LPI pump for decay heat removal and transport. The flow provided by one RCP or one LPI pump is adequate for heat removal. The other intent of this LCO is to require that two paths (loops) be available to provide redundancy for heat removal.

APPLICABLE SAFETY ANALYSES

No safety analyses relating to RCS flow requirements are performed with initial condition in MODE 4.

RCS Loops – MODE 4 satisfy Criterion 4 of 10 CFR 50.36 (Ref. 1).

LCO

The purpose of this LCO is to require that two loops, RCS or DHR, be OPERABLE in MODE 4 and one of these loops be in operation. The LCO allows the two loops that are required to be OPERABLE to consist of any combination of RCS or DHR loops. Any one loop in operation provides enough flow to remove the decay heat from the core with forced circulation. The second loop that is required to be OPERABLE provides redundant paths for heat removal.

The LCO is modified by two Notes. Note 1 permits a limited period of operation without RCPs. All RCPs may not be in operation for ≤ 8 hours per 24 hour period for the transition to or from the DHR System and otherwise may be de-energized for ≤ 1 hour per 8 hour period. This means that natural circulation has been established using the SGs.

Note 1 prohibits boron dilution when forced flow is stopped because an even concentration distribution cannot be ensured. Core outlet temperature is to be maintained at least 10°F below saturation temperature

BASES

LCO (continued)

so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

Note 1 also permits the DHR pumps to be stopped for ≤ 1 hour per 8 hour period. When the DHR pumps are stopped, no alternate heat removal path exists, unless the RCS and SGs have been placed in service in forced or natural circulation. The response of the RCS without the DHR loop depends on the core decay heat load and the length of time that the DHR pumps are stopped. As decay heat diminishes, the effects on RCS temperature and pressure diminish. Without cooling by DHR, higher heat loads will cause the reactor coolant temperature and pressure to increase at a rate proportional to the decay heat load. Because pressure can increase, the applicable system pressure limits (pressure and temperature (P/T) or low temperature overpressure protection (LTOP) limits) must be observed and forced DHR flow or heat removal via the SGs must be re-established prior to reaching the pressure limit. The circumstances for stopping both DHR trains are to be limited to situations where:

a. Pressure and temperature increases can be maintained well within the allowable pressure (P/T and LTOP) and 10°F subcooling limits; or

b. An alternate heat removal path through the SG is in operation.

Note 2 allows a DHR loop to be considered OPERABLE if it is capable of being manually (locally or remotely) realigned to the DHR mode of operation and is not otherwise inoperable. This provision is necessary because of the dual function of the components that comprise the decay heat removal mode of the Low Pressure Injection System.

An OPERABLE RCS loop consists of at least one OPERABLE RCP and an SG that is capable of transferring decay heat to the secondary fluid.

Similarly for the DHR loops, an OPERABLE DHR loop is comprised of the OPERABLE LPI pump(s) capable of providing forced flow to the LPI heat exchanger(s). LPI pumps are OPERABLE if they are capable of being powered and are able to provide flow if required.

APPLICABILITY

In MODE 4, this LCO ensures forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing.

BASES

LCO (continued)

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops – MODES 1 and 2";
LCO 3.4.5, "RCS Loops – MODE 3";
LCO 3.4.7, "RCS Loops – MODE 5, Loops Filled";
LCO 3.4.8, "RCS Loops – MODE 5, Loops Not Filled";
LCO 3.9.4, "Decay Heat Removal (DHR) and Coolant Circulation – High Water Level" (MODE 6); and
LCO 3.9.5, "Decay Heat Removal (DHR) and Coolant Circulation – Low Water Level" (MODE 6).

ACTIONS

A.1

If only one required RCS loop or DHR loop is OPERABLE and in operation, redundancy for heat removal is lost. Action must be initiated to restore a second loop to OPERABLE status. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal.

A.2

If restoration is not accomplished and a DHR loop is OPERABLE, the unit must be brought to MODE 5 within the following 24 hours. Bringing the unit to MODE 5 is a conservative action with regard to decay heat removal. With only one DHR loop OPERABLE, redundancy for decay heat removal is lost and, in the event of a loss of the remaining DHR loop, it would be safer to incur that loss from MODE 5 rather than MODE 4. The Completion Time of 24 hours is reasonable, based on operating experience, to reach MODE 5 in an orderly manner and without challenging unit systems.

This Required Action is modified by a Note which indicates that the unit must be placed in MODE 5 only if a DHR loop is OPERABLE. With no DHR loop OPERABLE, the unit is in a condition with only limited cooldown capabilities. Therefore, the actions are to be concentrated on restoration of a DHR loop, rather than a cooldown of extended duration.

B.1 and B.2

If no RCS or DHR loops are OPERABLE or a required loop is not in operation, (no loop is required to be in operation provided the conditions of the Note in the LCO section are met) all operations involving a reduction of RCS boron concentration must be suspended and action to restore one

BASES

ACTIONS

B.1 and B.2 (continued)

RCS or DHR loop to OPERABLE status and operation must be initiated. Boron dilution requires forced circulation for proper mixing, and the margin to criticality must not be reduced in this type of operation. The immediate Completion Times reflect the importance of maintaining operation for decay heat removal. The action to restore must continue until one loop is restored to operation.

SURVEILLANCE REQUIREMENTS

SR 3.4.6.1

This Surveillance requires verification every 12 hours of the required DHR or RCS loop in operation to ensure forced flow is providing decay heat removal. Verification includes flow rate, temperature, or pump status monitoring. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess RCS loop status. In addition, control room indication and alarms will normally indicate loop status.

SR 3.4.6.2

Verification that the required pump is OPERABLE ensures that an additional RCS or DHR loop can be placed in operation if needed to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the required pumps. The Frequency of 7 days is considered reasonable in view of other administrative controls and has been shown to be acceptable by operating experience.

REFERENCES

1. 10 CFR 50.36.

## B 3.4 REACTOR COOLANT SYSTEM (RCS)

## B 3.4.7 RCS Loops – MODE 5, Loops Filled

BASES

BACKGROUND

In MODE 5 with RCS loops filled, the primary function of the reactor coolant is the removal of decay heat and transfer of this heat either to the steam generator (SG) secondary side coolant or the low pressure service water via the LPI heat exchangers. While the principal means for decay heat removal is via the DHR loops, the SGs are specified as a backup means for redundancy. Although the SGs do not typically remove heat unless steaming occurs (which is not possible in MODE 5), they are available as a temporary heat sink and can be used by allowing the RCS to heat up into the temperature region of MODE 4 where steaming can be effective for heat removal. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid.

In MODE 5 with RCS loops filled, DHR loops are the principal means for heat removal. The number of loops in operation can vary to suit the operational needs. The intent of this LCO is to provide forced flow from at least one DHR loop for decay heat removal and transport. The flow provided by one DHR loop is adequate for decay heat removal. The other intent of this LCO is to require that a second path be available to provide a backup method for heat removal.

The LCO provides for either SG heat removal or DHR loop heat removal. In this MODE, reactor coolant pump (RCP) operation may be restricted because of net positive suction head (NPSH) limitations, and the SG will not be able to provide steam for the turbine driven feed pumps. However, to ensure that the SGs can be used as a heat sink, a motor driven feedwater pump is needed, because it is independent of steam. Condensate pumps, auxiliary feedwater pump, or a motor driven emergency feedwater pump can be used. If RCPs are available, the steam generator level need not be adjusted. If RCPs are not available, the water level must be adjusted for natural circulation. The high entry point in the generator should be accessible from the feedwater pumps so that natural circulation can be stimulated. The SGs are primarily a backup to the DHR loops, which are used for forced flow. By requiring the SGs to be a backup heat removal path, the option to increase RCS pressure and temperature for heat removal in MODE 4 is provided.

BASES (continued)

APPLICABLE SAFETY ANALYSES

No safety analyses relating to RCS flow requirements are performed with initial conditions in MODE 5.

RCS Loops – MODE 5 (Loops Filled) satisfy Criterion 4 of 10 CFR 50.36 (Ref. 1).

LCO

The purpose of this LCO is to require that at least one of the DHR loops be OPERABLE and in operation with an additional DHR loop OPERABLE or both SGs with secondary side water level ≥ 50%. One DHR loop provides sufficient forced circulation to perform the safety functions of the reactor coolant under these conditions. The second DHR loop is normally maintained as a backup to the operating DHR loop to provide redundancy for decay heat removal. However, if the standby DHR loop is not OPERABLE, a sufficient alternate method of providing redundant heat removal paths is to provide both SGs with their secondary side water levels ≥ 50%. Should the operating DHR loop fail, the SGs could be used to remove the decay heat.

Note 1 permits the DHR pumps to not be in operation for up to 1 hour per 8 hour period. The circumstances for stopping both DHR loops are to be limited to situations where: (a) Pressure and temperature increases can be maintained well within the allowable pressure (P/T and low temperature overpressure protection) and 10°F subcooling limits; and (b) no operations are in progress that will result in a reduction of RCS boron concentration.

The Note prohibits boron dilution when DHR forced flow is stopped because an even concentration distribution cannot be ensured. Core outlet temperature is to be maintained at least 10°F below saturation temperature so that no vapor bubble would form and possibly cause a natural circulation flow obstruction. In this MODE, the steam generators are used as a backup for decay heat removal and, to ensure their availability, the RCS loop flow path is to be maintained with subcooled liquid.

In MODE 5, it is sometimes necessary to stop all RCP or DHR loop forced circulation. For example, this may be necessary to change operation from one DHR loop to the other, perform surveillance or startup testing, perform the transition to and from the DHR loops, or to avoid operation below the RCP minimum NPSH limit. The time period is acceptable because natural circulation is acceptable for heat removal, the reactor coolant temperature can be maintained subcooled, and boron stratification affecting reactivity control is not expected.

LCO (continued)

Note 2 allows one required DHR loop to be inoperable for a period of ≤ 2 hours provided that the other loop is OPERABLE and in operation. This permits periodic surveillance tests to be performed on the inoperable loop during the only time when such testing is safe and possible.

Note 3 provides for an orderly transition from MODE 5 to MODE 4 during a planned heatup by permitting DHR loops to not be in operation when at least one RCP is in operation. This Note provides for the transition to MODE 4 where an RCP is permitted to be in operation and replaces the RCS circulation function provided by the DHR loops.

Note 4 allows a DHR loop to be considered OPERABLE during alignment and when aligned for low pressure injection if it is capable of being manually (locally or remotely) realigned to the DHR mode of operation and is not otherwise inoperable. This provision is necessary because of the dual requirements of the components that comprise the low pressure injection/decay heat removal system.

To be considered OPERABLE, a DHR loop must consist of a pump, a heat exchanger, valves, piping, instruments, and controls to ensure an OPERABLE flow path and to determine the temperature. The flow path starts in one of the RCS hot legs and is returned to reactor vessel via one or both Core Flood tank injection nozzles. The BWST recirculation crossover line through valves LP-40 and LP-41 may be part of a flow path if it provides adequate decay heat removal capability.

To be considered OPERABLE, DHR loops must be capable of being powered and are able to provide flow if required. An OPERABLE SG can perform as a heat sink when it has an adequate water level and is OPERABLE in accordance with the Steam Generator Tube Surveillance Program.

APPLICABILITY

In MODE 5 with loops filled, forced circulation is provided by this LCO to remove decay heat from the core and to provide proper boron mixing. One loop of DHR in operation provides sufficient circulation for these purposes.

Operation in other MODES is covered by:
LCO 3.4.4, "RCS Loops – MODES 1 and 2";
LCO 3.4.5, "RCS Loops – MODE 3";
LCO 3.4.6, "RCS Loops – MODE 4";
LCO 3.4.8, "RCS Loops – MODE 5, Loops Not Filled";
LCO 3.9.4, "Decay Heat Removal (DHR) and Coolant Circulation – High Water Level" (MODE 6); and
LCO 3.9.5, "Decay Heat Removal (DHR) and Coolant Circulation – Low Water Level" (MODE 6).



(Bases Change of 12-21-99)

BASES (continued)

ACTIONS

A.1 and A.2

If one required DHR loop is inoperable and any required SG has secondary side water level < 50%, redundancy for heat removal is lost. Action must be initiated to restore a second DHR loop to OPERABLE status or initiate action to restore the secondary side water level in the SGs, and action must be taken immediately. Either Required Action A.1 or Required Action A.2 will restore redundant decay heat removal paths. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal.

B.1 and B.2

If no required DHR loop is in operation (no DHR loop is required to be in operation provided the conditions of Note 1 are met), or no required DHR loop is OPERABLE, all operations involving the reduction of RCS boron concentration must be suspended and action to restore a DHR loop to OPERABLE status and operation must be initiated. Boron dilution requires forced circulation for proper mixing, and the margin to criticality must not be reduced in this type of operation. The immediate Completion Time reflects the importance of maintaining operation for decay heat removal.

SURVEILLANCE REQUIREMENTS

SR 3.4.7.1

This SR requires verification every 12 hours that the required DHR loop is in operation. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal. The 12 hour Frequency has been shown by operating practice to be sufficient to regularly assess degradation. In addition, control room indication and alarms will normally indicate loop status.

SR 3.4.7.2

Verifying the SGs are OPERABLE by ensuring their secondary side water levels are ≥ 50% ensures that redundant heat removal paths are available if the second DHR loop is not OPERABLE. If both DHR loops are OPERABLE, this Surveillance is not needed. The 12 hour Frequency has been shown by operating practice to be sufficient to regularly assess degradation and verify operation within safety analyses assumptions.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.7.3

Verification that each required DHR pump is OPERABLE ensures that a DHR loop can be placed in operation if needed to maintain decay heat removal and reactor coolant circulation. If the secondary side water level is ≥ 50% in both SGs, this Surveillance is not needed. Verification is performed by verifying proper breaker alignment and power available to the required pumps. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

REFERENCES

1. 10 CFR 50.36.

## B 3.4 REACTOR COOLANT SYSTEM (RCS)

## B 3.4.8 RCS Loops – MODE 5, Loops Not Filled

BASES

**BACKGROUND**

In MODE 5 with loops not filled, the primary function of the reactor coolant is the removal of decay heat and transfer of this heat to the LPI heat exchangers. The steam generators (SGs) are not available as a heat sink when the loops are not filled. The secondary function of the reactor coolant is to act as a carrier for the soluble neutron poison, boric acid.

Loops are not filled when RCS draining is initiated as might be the case for refueling or maintenance. GL 88-17 (Ref. 1) expresses concerns for loss of decay heat removal for this operating condition. With water at this low level, the margin above the decay heat suction piping connection to the hot leg is small. The possibility of loss of level or inlet vortexing exists and if it were to occur, the operating pump could become air bound and fail resulting in a loss of forced flow for heat removal. As a consequence the water in the core will heat up and could boil with the possibility of core uncovering due to boil off. Because the containment hatch may be open at this time, a pathway to the outside for fission product release could exist if core damage were to occur.

In MODE 5 with loops not filled, only DHR loops can be used for coolant circulation. The number of loops in operation can vary to suit the operational needs. The intent of this LCO is to provide forced flow from at least one DHR loop for decay heat removal and transport, to require that two paths be available to provide redundancy for heat removal.

**APPLICABLE SAFETY ANALYSES**

No safety analyses are performed with initial conditions in MODE 5 with loops not filled. The flow provided by one DHR loop is adequate for heat removal and for boron mixing.

RCS Loops – MODE 5 (Loops Not Filled) satisfy Criterion 4 of 10 CFR 50.36 (Ref. 2)

**LCO**

The purpose of this LCO is to require that a minimum of two DHR loops be OPERABLE and that one of these loops be in operation. An OPERABLE loop is one that has the capability of transferring heat from the reactor coolant at a controlled rate. Heat cannot be removed via the DHR loops

LCO (continued)

unless forced flow is used. A minimum of one running pump meets the LCO requirement for one loop in operation. An additional DHR loop is required to be OPERABLE to provide redundancy for heat removal.

Note 1 permits the DHR pumps to not be in operation for ≤ 15 minutes when switching from one loop to the other or for testing. The circumstances for stopping both DHR pumps are to be limited to situations where the outage time is short and temperature is maintained ≤ 140°F. The Note prohibits boron dilution or draining operations when DHR forced flow is stopped.

Note 2 allows one DHR loop to be inoperable for a period of ≤ 2 hours provided that the other loop is OPERABLE and in operation. This permits periodic surveillance tests to be performed on the inoperable loop during the only time when these tests are safe and possible.

Note 3 allows a DHR loop to be considered OPERABLE if it is capable of being manually (locally or remotely) realigned to the DHR mode of operation and is not otherwise inoperable. This provision is necessary because of the dual function of the components that comprise the low pressure injection/decay heat removal system.

To be considered OPERABLE, a DHR loop must consist of a pump, a heat exchanger, valves, piping, instruments, and controls to ensure an OPERABLE flow path and to determine the temperature. The flow path starts in one of the RCS hot legs and is returned to reactor vessel via one or both Core Flood tank injection nozzles. The BWST recirculation crossover line through valves LP-40 and LP-41 may be part of a flow path if it provides adequate decay heat removal capability. To be considered OPERABLE DHR pumps must be capable of being powered and are able to provide flow if required.

APPLICABILITY

In MODE 5 with loops not filled, this LCO requires core heat removal and coolant circulation by the DHR loops.

Operation in other MODES is covered by:
LCO 3.4.4, "RCS Loops – MODES 1 and 2";
LCO 3.4.5, "RCS Loops – MODE 3";
LCO 3.4.6, "RCS Loops – MODE 4";
LCO 3.4.7, "RCS Loops – MODE 5, Loops Filled";
LCO 3.9.4, "Decay Heat Removal (DHR) and Coolant Circulation – High Water Level" (MODE 6); and
LCO 3.9.5, "Decay Heat Removal (DHR) and Coolant Circulation – Low Water Level" (MODE 6).



(Bases Change of 12-21-99)

BASES (continued)

ACTIONS

A.1

If one required DHR loop is inoperable, redundancy for heat removal is lost. Required Action A.1 is to immediately initiate activities to restore a second loop to OPERABLE status. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal.

B.1 and B.2

If no required loop is OPERABLE or the required loop is not in operation, (no loop is required to be in operation provided the conditions of Note 1 in the LCO are met), the Required Action requires immediate suspension of all operations involving boron reduction and requires initiation of action to immediately restore one DHR loop to OPERABLE status and operation. The Required Action for restoration does not apply to the condition of both loops not in operation when the exception Note in the LCO is in force. The immediate Completion Time reflects the importance of maintaining operations for decay heat removal. The action to restore must continue until one loop is restored.

SURVEILLANCE REQUIREMENTS

SR 3.4.8.1

This Surveillance requires verification every 12 hours that the required loop is in operation. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess degradation and verify operation within safety analyses assumptions.

SR 3.4.8.2

Verification that the required number of pumps are OPERABLE ensures that redundancy for heat removal is provided. The requirement also ensures that additional loops can be placed in operation if needed to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the required pumps. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

BASES (continued)

REFERENCES

1. Generic Letter 88-17, October 17, 1988.

2. 10 CFR 50.36.

## B 3.4 REACTOR COOLANT SYSTEM (RCS)

### B 3.4.9 Pressurizer

BASES

BACKGROUND

The pressurizer provides a point in the RCS where liquid and vapor are maintained in equilibrium under saturated conditions for pressure control purposes to prevent bulk boiling in the remainder of the RCS. Key functions include maintaining required primary system pressure during steady state operation and limiting the pressure changes caused by reactor coolant thermal expansion and contraction during normal load transients.

The pressure control components addressed by this LCO include the pressurizer water level, the required heaters, and their controls and emergency power supplies. Pressurizer safety valves are addressed by LCO 3.4.10, "Pressurizer Safety Valves."

The maximum water level limit has been established to ensure that a liquid to vapor interface exists to permit RCS pressure control during normal operation and proper pressure response for anticipated design basis transients. The water level limit thus serves two purposes:

a. Pressure control during normal operation maintains subcooled reactor coolant in the loops and thus is in the preferred state for heat transport; and

b. By restricting the level to a maximum, expected transient reactor coolant volume increases (pressurizer insurge) will not cause excessive level changes that could result in degraded ability for pressure control.

The maximum water level limit permits pressure control equipment to function as designed. The limit preserves the steam space during normal operation, thus both spray and heaters can operate to maintain the design operating pressure. If the level limits were exceeded prior to a transient that creates a large pressurizer insurge volume, the maximum RCS pressure might exceed the design Safety Limit (SL) of 2750 psig.

The pressurizer heaters are used to maintain a pressure in the RCS so reactor coolant in the loops is subcooled and thus in the preferred state for heat transport to the steam generators (SGs). This function must be

BASES

BACKGROUND (continued)

maintained with a loss of offsite power. Consequently, the emphasis of this LCO is to ensure that the essential power supplies and the associated heaters are adequate to maintain pressure for RCS loop subcooling with an extended loss of offsite power.

A minimum required available capacity of 126 kW ensures that the RCS pressure can be maintained. Unless adequate heater capacity is available, reactor coolant subcooling cannot be maintained indefinitely. Inability to control the system pressure and maintain subcooling under conditions of natural circulation flow in the primary system could lead to loss of single phase natural circulation and decreased capability to remove core decay heat.

APPLICABLE SAFETY ANALYSES

In MODES 1 and 2, the LCO requirement for a steam bubble is reflected implicitly in the accident analyses. No associated safety analyses are performed in lower MODES. All analyses performed from a critical reactor condition assume the existence of a steam bubble and saturated conditions in the pressurizer. In making this assumption, the analyses neglect the small fraction of noncondensible gases normally present.

Safety analyses presented in the UFSAR do not take credit for pressurizer heater operation; however, an implicit initial condition assumption of the safety analyses is that the RCS is operating at normal pressure.

The maximum level limit is of prime interest for the startup accident and Loss of Main Feedwater (LOMFW) event. Conservative safety analyses assumptions for the startup accident indicate that it produces the largest increase of pressurizer level caused by an analyzed event. Thus this event has been selected to establish the pressurizer water level limit. For pressurizer levels > than 285 inches, the LOMFW event may be more limiting.

Evaluations performed for the design basis large break loss of coolant accident (LOCA), which assumed a higher maximum level than assumed for the startup accident, have been made. The higher pressurizer level assumed for the LOCA is the basis for the volume of reactor coolant released to the containment. The containment analysis performed using the mass and energy release demonstrated that the maximum resulting containment pressure was within design limits.

The requirement for emergency power supplies is based on NUREG-0737 (Ref. 2). The intent is to allow maintaining the reactor coolant in a subcooled condition with natural circulation at hot, high pressure conditions

BASES

APPLICABLE SAFETY ANALYSES (continued)

for an undefined, but extended, time period after a loss of offsite power. While loss of offsite power is an initial condition or coincident event assumed in many accident analyses, maintaining hot, high pressure conditions over an extended time period is not evaluated as part of UFSAR accident analyses.

The maximum pressurizer water level limit satisfies Criterion 2 of 10 CFR 50.36 (Ref. 1). Although the heaters are not specifically used in accident analysis, the need to maintain subcooling in the long term during loss of offsite power, as indicated in NUREG-0737 (Ref. 2), is the reason for providing an LCO.

LCO

The LCO requirement for the pressurizer to be OPERABLE with a water level ≤ 285 inches ensures that a steam bubble exists. Limiting the maximum operating water level preserves the steam space for pressure control. The LCO has been established to ensure the capability to establish and maintain pressure control for steady state operation and to minimize the consequences of potential overpressure transients. Requiring the presence of a steam bubble is also consistent with analytical assumptions.

The LCO requires a minimum of 126 kW of pressurizer heaters OPERABLE and capable of being powered from an emergency power supply. As such, the LCO addresses both the heaters and the power supplies. The minimum heater capacity required is sufficient to maintain the system near normal operating pressure when accounting for heat losses through the pressurizer insulation. By maintaining the pressure near the operating conditions, a wide margin to subcooling can be obtained in the loops. The design value of 126 kW is derived from the use of nine heaters rated at 14 kW each. The amount needed to maintain pressure is dependent on the insulation losses, which can vary due to tightness of fit and condition.

APPLICABILITY

The need for pressure control is most pertinent when core heat can cause the greatest effect on RCS temperature, resulting in the greatest effect on pressurizer level and RCS pressure control. Thus Applicability has been designated for MODES 1 and 2. The Applicability is also provided for MODE 3 with RCS temperature > 325°F. The purpose is to prevent solid water RCS operation during heatup and cooldown to avoid rapid pressure rises caused by normal operational perturbations, such as reactor coolant

BASES

APPLICABILITY (continued)

pump startup. The temperature of 325°F has been designated as the cutoff for applicability because LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," provides a requirement for pressurizer level ≤ 325°F. The LCO does not apply in MODE 4, 5 or 6 since either pressurizer level is under the control of LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," or the RCS is open to the containment atmosphere.

In MODES 1, 2, and 3, there is the need to maintain the availability of pressurizer heaters capable of being powered from an emergency power supply. In the event of a loss of offsite power, the initial conditions of these MODES give the greatest demand for maintaining the RCS in a hot pressurized condition with loop subcooling for an extended period. For MODE 4, 5, or 6, it is not necessary to control pressure (by heaters) to ensure loop subcooling for heat transfer when the Decay Heat Removal loops are in service, and therefore the LCO is not applicable.

ACTIONS

A.1

With pressurizer water level in excess of the maximum limit, action must be taken to restore pressurizer operation to within the bounds assumed in the analysis. This is done by restoring the pressurizer water level to within the limit.

The 1 hour Completion Time is considered to be a reasonable time for draining excess liquid.

B.1 and B.2

If the water level cannot be restored, reducing core power constrains heat input effects that drive pressurizer insurge that could result from an anticipated transient. By shutting down the reactor and reducing reactor coolant temperature to at least MODE 3 with RCS temperature ≤ 325°F, the potential thermal energy of the reactor coolant mass for LOCA mass and energy releases is reduced.

Twelve hours is a reasonable time based upon operating experience to reach MODE 3 from full power without challenging unit systems and operators. Further pressure and temperature reduction to MODE 3 with RCS temperature ≤ 325°F places the unit into a MODE where the LCO is not applicable. The 18 hour Completion Time to reach the nonapplicable MODE is reasonable based upon operating experience.

BASES

ACTIONS (continued)

C.1

If the power supplies to the heaters are not capable of providing 126 kW, or the pressurizer heaters are inoperable, restoration is required in 72 hours. The Completion Time of 72 hours is reasonable considering the anticipation that a demand will not occur in this period.

D.1 and D.2

If pressurizer heater capability cannot be restored within the allowed Completion Time of Required Action C.1, 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 MODE 3 within 12 hours and to MODE 3 with RCS temperature ≤ 325°F within the following 6 hours. The Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging unit systems. Similarly, the Completion Time of 18 hours to be in MODE 3 with RCS temperature ≤ 325°F is reasonable based on operating experience to achieve power reduction from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.9.1

This SR requires that during steady state operation, pressurizer water level is maintained below the nominal upper limit to provide a minimum space for a steam bubble. The Surveillance is performed by observing the indicated level. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess the level for any deviation and verify that operation is within safety analyses assumptions. Alarms are also available for early detection of abnormal level indications.

SR 3.4.9.2

The SR verifies the power supplies are capable of producing the minimum power and the associated pressurizer heaters are at their design rating. (This may be done by testing the power supply output and heater current, or by performing an electrical check on heater element continuity and resistance.) The Frequency of 18 months is considered adequate to detect heater degradation and has been shown by operating experience to be acceptable.

BASES (continued)

REFERENCES

1. 10 CFR 50.36.

2. NUREG-0737, November 1980.

## B 3.4 REACTOR COOLANT SYSTEM (RCS)

### B 3.4.10 Pressurizer Safety Valves

BASES

BACKGROUND

The purpose of the two spring loaded pressurizer safety valves is to provide RCS overpressure protection. Operating in conjunction with the Reactor Protection System (RPS), two valves are used to ensure that the Safety Limit (SL) of 2750 psig is not exceeded for analyzed transients during operation in MODES 1 and 2. Two safety valves are used for portions of MODE 3. For the remainder of MODE 3, MODE 4, MODE 5, and MODE 6 with the reactor head on, overpressure protection is provided by operating procedures and LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."

The self actuated pressurizer safety valves are designed in accordance with the requirements set forth in the ASME Boiler and Pressure Vessel Code, Section III (Ref. 1). The setpoint of the pressurizer code safety valves is in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Article 9, Summer 1967. The safety valves discharge steam from the pressurizer to a quench tank located in the containment. The discharge flow is indicated by an increase in temperature downstream of the safety valves and by an increase in the quench tank temperature and level.

The required lift pressure is 2500 psig ± 3%. The upper and lower pressure limits are based on the requirements of ASME Boiler and Pressure Vessel Code, Section III, Article 9, Summer 1967, which limit the rise in pressure within the vessels which they protect to 10% above the design pressure. The lift setting is for the ambient conditions associated with MODES 1, 2, and 3. This requires either that the valves be set hot or that a correlation between hot and cold settings be established.

The pressurizer safety valves are part of the primary success path and mitigate the effects of postulated accidents. OPERABILITY of the safety valves ensures that the RCS pressure will be limited to 110% of design pressure.

The consequences of exceeding the ASME pressure limit could include damage to RCS components, increased leakage, or a requirement to perform additional stress analyses prior to resumption of reactor operation.

BASES (continued)

APPLICABLE SAFETY ANALYSES

All accident analyses in the UFSAR that require safety valve actuation assume operation of both pressurizer safety valves to limit increasing reactor coolant pressure. The overpressure protection analysis is also based on operation of both safety valves and assumes that the valves open at the high range of the setting (2500 psig system design pressure plus 3%). These valves must accommodate pressurizer insurges that could occur during a startup, rod withdrawal, ejected rod, or loss of main feedwater. The startup accident establishes the minimum safety valve capacity. The startup accident is assumed to occur at < 15% power. Single failure of a safety valve is neither assumed in the accident analysis nor required to be addressed by the ASME Code. Compliance with this Specification is required to ensure that the accident analysis and design basis calculations remain valid.

Pressurizer safety valves satisfy Criterion 3 of 10 CFR 50.36 (Ref. 3).

LCO

The two pressurizer safety valves are set to open at the RCS design pressure (2500 psig) and within the ASME specified tolerance to avoid exceeding the maximum RCS design pressure SL, to maintain accident analysis assumptions and to comply with ASME Code requirements. The valves will be tested per ASME Section XI requirements and returned to service with as-left setpoints of 2500 psig ± 1%. The upper and lower pressure tolerance limits are based on the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Article 9, Summer 1967, which limit the rise in pressure within the vessel which they protect, to 10% above the design pressure. Inoperability of one or both valves could result in exceeding the SL if a transient were to occur.

The consequences of exceeding the ASME pressure limit could include damage to one or more RCS components, increased leakage, or additional stress analysis being required prior to resumption of reactor operation.

APPLICABILITY

In MODES 1, 2, and portions of MODE 3 above the LTOP cut in temperature, OPERABILITY of two valves is required because the combined capacity is required to keep reactor coolant pressure below 110% of its design value during certain accidents. Portions of MODE 3 are conservatively included, although the listed accidents may not require both safety valves for protection.

BASES

APPLICABILITY (continued)

The LCO is not applicable in MODE 3 when any RCS cold leg temperature is ≤ 325°F, MODE 4 and MODE 5 because LTOP protection is provided. Overpressure protection is not required in MODE 6 with the reactor vessel head detensioned.

The Note allows entry into MODE 3 with the lift settings outside the LCO limits. This permits testing and examination of the safety valves at high pressure and temperature near their normal operating range, but only after the valves have had a preliminary cold setting. The cold setting gives assurance that the valves are OPERABLE near their design condition. Only one valve at a time will be removed from service for testing. The 36 hour exception is based on an 18 hour outage time for each of the two valves. The 18 hour period is derived from operating experience that hot testing can be performed in this time frame.

ACTIONS

A.1

With one pressurizer safety valve inoperable, restoration must take place within 15 minutes. The Completion Time of 15 minutes reflects the importance of maintaining the RCS overpressure protection system. An inoperable safety valve coincident with an RCS overpressure event could challenge the integrity of the RCPB.

B.1 and B.2

If the Required Action cannot be met within the required Completion Time or if both pressurizer safety valves are inoperable, 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 12 hours and to MODE 3 with any RCS cold leg temperature ≤ 325°F within 18 hours. The 12 hours allowed is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging unit systems. Similarly, the 18 hours allowed is reasonable, based on operating experience, to reach MODE 3 with any RCS cold leg temperature ≤ 325°F without challenging unit systems. With any RCS cold leg temperature at or below 325°F, overpressure protection is provided by LTOP. Reducing the RCS temperature to ≤ 325°F reduces the RCS energy (core power and pressure), lowers the potential for large pressurizer insurges, and thereby removes the need for overpressure protection by two pressurizer safety valves.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.10.1

SRs are specified in the Inservice Testing Program. Pressurizer safety valves are to be tested in accordance with the requirements of Section XI of the ASME Code (Ref. 2), which provides the activities and the Frequency necessary to satisfy the SRs. No additional requirements are specified.

The pressurizer safety valves setpoint is ± 3% for OPERABILITY; however, the valves are reset to ±1% during the Surveillance to allow for drift. These values include instrument uncertainties.

REFERENCES

1. ASME, Boiler and Pressure Vessel Code, Section III.

2. ASME, Boiler and Pressure Vessel Code, Section XI.

3. 10 CFR 50.36.

## B 3.4 REACTOR COOLANT SYSTEM (RCS)

### B 3.4.11 RCS Specific Activity

BASES

BACKGROUND

The Code of Federal Regulations, 10 CFR 100 (Ref. 1), specifies the maximum dose to the whole body and the thyroid an individual at the site boundary can receive for 2 hours during an accident. The limits on specific activity ensure that the doses are held to within the 10 CFR 100 limits during analyzed transients and accidents.

The RCS specific activity LCO limits the allowable concentration level of radionuclides in the reactor coolant. The LCO limits are established to minimize the offsite radioactivity dose consequences in the event of a steam generator tube rupture (SGTR) accident.

The LCO contains specific activity limits for both DOSE EQUIVALENT I-131 and gross specific activity. The allowable levels are intended to limit the 2 hour dose at the site boundary to within the 10 CFR 100 dose guideline limits.

Analysis shows the potential offsite dose levels for an SGTR accident are within the 10 CFR 100 dose guideline limits (Ref. 1).

APPLICABLE SAFETY ANALYSES

The LCO limits on the specific activity of the reactor coolant ensure that the resulting 2 hour doses at the site boundary will not exceed the 10 CFR 100 dose guideline limits following an SGTR or a steam line break (SLB) accident. The SLB safety analysis (Ref. 2) assumptions bound the specific activity of the reactor coolant at the LCO limits and a total existing reactor coolant steam generator (SG) tube leakage rate of 300 gpd. However, the 300 gpd leakage has a negligible effect on the consequences of a SLB. The analysis also assumes a reactor trip and a turbine trip as a result of the SLB event.

The analysis results for the SGTR accident are significantly impacted by the acceptance limits for RCS specific activity. Reference to this analysis is used to assess changes to the facility that could affect RCS specific activity as they relate to the acceptance limits.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The safety analysis shows the radiological consequences of an SGTR accident are within the Reference 1 dose guideline limits. Operation with iodine specific activity levels greater than the LCO limit is permissible, if the activity levels do not exceed the limits shown in Figure 3.4.11-1, in the applicable Specification, for more than 48 hours.

The remainder of the above limit permissible iodine levels shown in Figure 3.4.11-1 are acceptable because of the low probability of an SGTR accident occurring during the established 48 hour time limit. The occurrence of an SGTR accident at these permissible levels could increase the site boundary dose levels, but still be within 10 CFR 100 dose guideline limits.

RCS Specific Activity satisfies Criterion 2 of 10 CFR 50.36 (Ref. 3).

LCO

The specific iodine activity is limited to 1.0 μCi/gm DOSE EQUIVALENT I-131, and the gross specific activity in the primary coolant is limited to the number of μCi/gm equal to 100 divided by $\bar{E}$ ($\bar{E}$ is the average (mean) beta and gamma energies per disintegration, in MeV, weighted in proportion to the measured activity of the radionuclides in reactor coolant samples). The limit on DOSE EQUIVALENT I-131 ensures the 2 hour thyroid dose to an individual at the site boundary during an accident will be within the allowed thyroid dose. The limit on gross specific activity ensures the 2 hour whole body dose to an individual at the site boundary during an accident will be within the allowed whole body dose.

The SGTR accident analysis (Ref. 2) shows that the 2 hour site boundary dose levels are within acceptable limits. Violation of the LCO may result in reactor coolant radioactivity levels that could, in the event of an SGTR, lead to site boundary doses that exceed the 10 CFR 100 dose guideline limits.

APPLICABILITY

In MODES 1 and 2, and in MODE 3 with RCS average temperature ≥ 500°F, operation within the LCO limits for DOSE EQUIVALENT I-131 and gross specific activity are necessary to contain the potential consequences of an SGTR to within the acceptable site boundary dose values.

For operation in MODE 3 with RCS average temperature < 500°F, and in MODES 4 and 5, the release of radioactivity in the event of an SGTR is unlikely since the saturation pressure of the reactor coolant is below the lift pressure settings of the main steam relief valves.

BASES (continued)

ACTIONS

A.1 and A.2

With the DOSE EQUIVALENT I-131 greater than the LCO limit, samples at intervals of 4 hours must be taken to demonstrate the limits of Figure 3.4.11-1 are not exceeded. The Completion Time of 4 hours is required to obtain and analyze a sample. Sampling must continue for trending.

The DOSE EQUIVALENT I-131 must be restored to limits within 48 hours. The Completion Time of 48 hours is required, if the limit violation resulted from normal iodine spiking.

A Note to RA A.1 and A.2 excludes the MODE change restriction of LCO 3.0.4. This exception allows entry into the applicable MODE(S) while relying on the ACTIONS even though the ACTIONS may eventually require unit shutdown. This exception is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is limiting due to exceeding this limit, and the ability to restore transient specific activity excursions while the unit remains at, or proceeds to power operation.

B 1

If a Required Action and associated Completion Time of Condition A are not met or if the DOSE EQUIVALENT I-131 is in the unacceptable region of Figure 3.4.11-1, the reactor must be brought to MODE 3 with RCS average temperature < 500°F within 12 hours. The Completion Time of 12 hours is required to get to MODE 3 below 500°F without challenging reactor emergency systems.

C.1

With the gross specific activity in excess of the allowed limit, the unit must be placed in a MODE in which the requirement does not apply.

The allowed Completion Time of 12 hours to reach MODE 3 and RCS average temperature < 500°F lowers the saturation pressure of the reactor coolant below the setpoints of the main steam safety valves, and prevents venting the SG to the environment in an SGTR event. The Completion Time of 12 hours is required to reach MODE 3 with RCS average temperature ≤ 500°F from full power conditions in an orderly manner and without challenging reactor emergency systems.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.11.1

SR 3.4.11.1 requires performing a gamma isotopic analysis as a measure of the gross specific activity of the reactor coolant at least once per 7 days. While basically a quantitative measure of radionuclides with half lives longer than 30 minutes, excluding iodines, this measurement is the sum of the degassed gamma activities and the gaseous gamma activities in the sample taken. This Surveillance provides an indication of any increase in gross specific activity.

Trending the results of this Surveillance allows proper remedial action to be taken before reaching the LCO limit under normal operating conditions. The Surveillance is applicable in MODES 1 and 2, and in MODE 3 with RCS average temperature at least 500°F. The 7 day Frequency considers the unlikelihood of a gross fuel failure during that time period.

SR 3.4.11.2

This Surveillance is performed in MODE 1 only to ensure the iodine remains within limit during normal operation and following fast power changes when fuel failure is more apt to occur. The 14 day Frequency is adequate to trend changes in the iodine activity level considering gross specific activity is monitored every 7 days. The Frequency, between 2 and 6 hours after a power change of ≥ 15% RTP within a 1 hour period, is established because the iodine levels peak during this time following fuel failure; samples at other times would provide inaccurate results.

SR 3.4.11.3

SR 3.4.11.3 requires radiochemical analysis for $\bar{E}$ determination every 184 days (6 months) with the unit operating in MODE 1 equilibrium conditions. The $\bar{E}$ determination directly relates to the LCO and is required to verify unit operation within the specific gross activity LCO limit. The analysis for $\bar{E}$ is a measurement of the average energies per disintegration for isotopes with half lives longer than 30 minutes, excluding iodines. The Frequency of 184 days recognizes $\bar{E}$ does not change rapidly.

This SR has been modified by a Note that requires sampling to be performed 31 days after a minimum of 2 EFPD and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for at least 48 hours. This ensures the radioactive materials are at equilibrium so the analysis for $\bar{E}$ is representative and not skewed by a crud burst or other similar abnormal event.

BASES (continued)

REFERENCES

1. 10 CFR 100.11.
2. UFSAR, Section 15.9.
3. 10 CFR 50.36.

## B 3.4 REACTOR COOLANT SYSTEM (RCS)

### B 3.4.12 Low Temperature Overpressure Protection (LTOP) System

BASES

BACKGROUND

The LTOP System limits RCS pressure at low temperatures so the integrity of the reactor coolant pressure boundary (RCPB) is not compromised by violating the pressure and temperature (P/T) requirements of 10 CFR 50, Appendix G (Ref. 1). The reactor vessel is the limiting RCPB component for providing such protection. LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," provides the allowable combinations for operational pressure and temperature during cooldown, shutdown, and heatup to keep from violating the Reference 1 limits.

The reactor vessel material is less ductile at reduced temperatures than at normal operating temperature. Also, as vessel neutron irradiation accumulates, the material becomes less resistant to pressure stress at low temperatures (Ref. 2). RCS pressure must be maintained low when temperature is low and may be increased only as temperature is increased.

Operational maneuvering during cooldown, heatup, or related anticipated transients must be controlled to not violate LCO 3.4.3. Exceeding these limits could lead to brittle fracture of the reactor vessel. LCO 3.4.3 presents requirements for administrative control of RCS pressure and temperature to prevent exceeding the P/T limits.

This LCO provides RCS overpressure protection in the applicable MODES by ensuring an adequate pressure relief capacity and a limit on coolant addition capability. The pressure relief capacity requires the power operated relief valve (PORV) lift setpoint to be reduced and administrative controls implemented which assure ≥ 10 minutes available for operator action to mitigate an LTOP event. The administrative controls include limits on pressurizer level, limits on RCS pressure when RCS temperature is < 325°F, limits on RCS makeup flow, the number of available pressurizer heater banks, requirements for alarms and restrictions upon use of the High Pressure Nitrogen System.

The LTOP approach to protecting the vessel by limiting coolant addition capability requires controls upon RCS makeup flow, the number of available pressurizer heater banks, and requires deactivating HPI, and isolating the core flood tanks (CFTs).

Should one or more HPI pumps inject on an HPI actuation (HPI-ES) or a CFT discharge to the RCS, the pressurizer level and PORV may not prevent overpressurizing the RCS.

BASES

BACKGROUND (continued)

The administrative controls upon pressurizer level provides a compressible vapor space or cushion (either steam or nitrogen) that can accommodate a coolant insurge and prevent a rapid pressure increase, allowing the operator time to stop the increase. The PORV, with reduced lift setting, is the overpressure protection device that acts as backup to the operator in terminating an increasing pressure event.

With HPI-ES deactivated, the ability to provide RCS coolant addition is restricted. To balance the possible need for coolant addition, the LCO does not require the makeup system to be deactivated. Due to the lower pressures associated with the LTOP MODES and the expected decay heat levels, the makeup system can provide flow with the HPI pumps providing RCS makeup through the makeup control valve.

PORV Requirements

As required for LTOP, the PORV is signaled to open if the RCS pressure approaches a limit set in the LTOP actuation circuit. The LTOP actuation circuit monitors RCS pressure and determines when an overpressure condition is approached. When the monitored pressure meets or exceeds the setting, the PORV is signaled to open. Maintaining the setpoint within the limits of the LCO ensures the Reference 1 limits will be met in any event analyzed for LTOP.

When a PORV is opened in an increasing pressure transient, the release of coolant causes the pressure increase to slow and reverse. As the PORV releases steam, the RCS pressure decreases until a reset pressure is reached and the valve is signaled to close. The pressure continues to decrease below the reset pressure as the valve closes.

Administrative Control Requirements

Administrative controls are necessary to assure the operator has at least ten minutes available to mitigate the most limiting LTOP event. These administrative controls include the following:

1) Limits on RCS pressure based on RCS temperature;

2) Limits upon pressurizer level;

3) Limits upon makeup flow capability;

4) OPERABLE Alarms;

5) Controls upon use of the High Pressure Nitrogen System; and

6) Restricting the number of available pressurizer heater banks.

BASES

BACKGROUND

Administrative Control Requirements (continued)

Limiting RCS pressure based on RCS temperature provides a minimum margin to the RCS P/T limit. Restricting RCS makeup flow capability and pressurizer level and controls on the use of high pressure nitrogen limit the pressurization rate during an LTOP event. Restricting the number of available pressurizer heater banks limits the pressurization rate during an LTOP event. Alarms ensure early operator recognition of the occurrence of an LTOP event. The combination of minimum margin to the limit, limited pressurization rate and OPERABLE alarms ensure ten minutes are available for operator action to mitigate an LTOP event.

APPLICABLE SAFETY ANALYSES

Safety analyses (Ref. 3) demonstrate that the reactor vessel can be adequately protected against overpressurization transients during shutdown. In MODES 1, 2, and in MODE 3 with RCS temperature exceeding 325°F, the pressurizer safety valves will prevent RCS pressure from exceeding the Reference 1 limits. At nominally 325°F and below, overpressure prevention falls to an OPERABLE PORV, a restricted coolant level in the pressurizer and other administrative controls.

The actual temperature at which the pressure in the P/T limit curve falls below the pressurizer safety valve setpoint increases as vessel material toughness decreases due to neutron embrittlement. Each time the P/T limit curves are revised, the LTOP System will be re-evaluated to ensure that its functional requirements can still be met with the PORV and pressurizer level/administrative controls method.

Transients that are capable of overpressurizing the RCS have been identified and evaluated. These transients relate to either mass input or heat input: actuating the HPI System, discharging the CFTs, energizing the pressurizer heaters, failing the makeup control valve open, losing decay heat removal, starting a reactor coolant pump (RCP) with a large temperature mismatch between the primary and secondary coolant systems, and adding nitrogen to the pressurizer. LTOP limits and restrictions take into account the presence of nitrogen and/or air in the RCS during LTOP conditions.

HPI actuation and CFT discharge are the transients that may result in exceeding P/T limits within < 10 minutes in which time no operator action is assumed to take place. Starting an RCP and adding nitrogen to the pressurizer are self limiting events. In the rest, operator action after that time precludes overpressurization. The analyses demonstrate that the time allowed for operator action is adequate, or the events are self limiting and do not exceed P/T limits.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The following controls are required during the LTOP MODES to ensure that transients do not occur, which either of the LTOP overpressure protection means cannot handle:

a. Limiting RCS makeup flow capability;

b. Deactivating HPI-ES;

c. Immobilizing CFT discharge isolation valves in their closed positions; and

d. Limiting the number of available pressurizer heater banks.

The Reference 3 analyses demonstrate the PORV can maintain RCS pressure below limits when both makeup flow capability and the number of available pressurizer heater banks is restricted. Consequently, the administrative controls require makeup flow capability and the number of available pressurizer heater banks to be limited in the LTOP MODES.

Since the PORV cannot protect the reactor vessel for engineered safeguards actuation of one or more HPI pumps, or discharging the CFTs, the LCO also requires the HPI-ES actuation circuits be deactivated and the CFTs isolated. The isolated CFTs must have their discharge valves closed and the valve power breakers fixed in their open positions.

Fracture mechanics analyses established the temperature of LTOP Applicability at 325°F. Above this temperature, the pressurizer safety valves provide the reactor vessel pressure protection. The vessel materials were assumed to have a neutron irradiation accumulation equal to 33 effective full power years (EFPYs) of operation for Units 1, 2, and 3.

This LCO will deactivate the HPI-ES actuation when the RCS temperature is ≤ 325°F.

Reference 3 contains the acceptance limits that satisfy the LTOP requirements. Any change to the RCS must be evaluated against these analyses to determine the impact of the change on the LTOP acceptance limits.

PORV Performance

The fracture mechanics analyses show that the vessel is protected when the PORV is set to open at ≤ 535 psig. The setpoint is derived by modeling the performance of the LTOP system for different LTOP events. The PORV setpoint at or below the derived limit ensures the Reference 1 limits will be met.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The PORV setpoint is re-evaluated for compliance when the revised P/T limits conflict with the LTOP analysis limits. The P/T limits are periodically modified as the reactor vessel material toughness decreases due to embrittlement induced by neutron irradiation. Revised P/T limits are determined using neutron fluence projections and the results of examinations of the reactor vessel material irradiation surveillance specimens. The Bases for LCO 3.4.3 discuss these examinations.

The PORV is considered an active component. Therefore, its failure represents the worst case LTOP single active failure.

Administrative Controls Performance

Limiting RCS pressure when RCS temperature is < 325°F provides a minimum margin to the RCS P/T limit. Restricting RCS makeup flow capability, the number of available pressurizer heater banks, pressurizer level, and controls on the use of high pressure nitrogen limit the pressurization rate during an LTOP event. Alarms ensure early operator recognition of the occurrence of an incipient LTOP event. The combination of minimum margin to the limit, limited pressurization rate and OPERABLE alarms ensure ten minutes are available for operator action to mitigate an LTOP event.

RCS Vent Requirements for Testing

With the RCS depressurized, analyses show:

a. For HPI System testing, a vent of ≥ 3.6 square inches is capable of mitigating the transient resulting from HPI-ES actuation testing in which three HPI pumps inject to the RCS through two injection flow paths.

b. For CFT Discharge Testing, a vent of ≥ 201 square inches is capable of mitigating the transient resulting for discharge of both CFTs to the RCS.

The capacity of vents of these minimum sizes is sufficient to limit the RCS pressure to ≤ 400 psig, which is less than the maximum allowable pressure at minimum RCS temperature.

The RCS vent size will also be re-evaluated for compliance each time P/T limit curves are revised based on the results of the vessel material surveillance.

These vents are passive and not subject to active failure.

The LTOP System satisfies Criterion 2 and Criterion 3 of 10 CFR 50.36 (Ref.6).

BASES

LCO

The LCO requires an LTOP System OPERABLE with a limited coolant input capability and a pressure relief capability. The LCO requires HPI to be deactivated and the CFTs to be isolated. For pressure relief, it requires the pressurizer coolant at or below a maximum level and the PORV OPERABLE with a lift setting ≤ the LTOP limit.

The PORV is OPERABLE when its block valve is open, its lift setpoint is set at ≤ 535 psig and testing has proven its ability to open at that setpoint, and power is available to the two valves and their control circuits.

An RCS vent path capable of mitigating the most limiting LTOP event (except for HPI-ES actuation or CFT discharge) has a minimum equivalent diameter of 1-3/32 inches, which is equal to the inner throat diameter of the PORV.

Implementation of the following administrative controls assure that ≥ 10 minutes are available for operator action to mitigate an LTOP event:

1. RCS pressure:

   < 375 psig when RCS temperature ≤ 220°F

   < 525 psig when RCS temperature > 220°F and ≤ 325°F

2. Pressurizer level is maintained within the following limits:

   a. RCS pressure is > 100 psig:

      ≤ 220 inches when RCS temperature ≤ 325°F

   b. RCS pressure is ≤ 100 psig:

      ≤ 310 inches when RCS temperature ≤ 220°F.

      ≤ 380 inches while filling or draining the RCS when RCS temperature ≤ 160°F and no HPI pumps are running.

      When the RCS pressure is ≤ 100 psig, pressurizer level is normally maintained ≤ 220 inches except for certain RCS evolutions. The specified pressurizer level limits provide assurance that at least 10 minutes is available for operator action during those evolutions. The temperature limits are based on operational limits for the evolutions and are used in the analyses to determine allowable pressurizer levels.

3. Makeup flow is restricted with the HP-120 (makeup control valve) travel stop set to ≤ 98.0 gpm for all three units.

BASES

LCO (continued)

4. Three audible pressurizer level alarms at ≤ 225 inches, ≤ 260 inches, and ≤ 315 inches from the temperature compensated pressurizer level indication.

5. Two audible RCS pressure alarms at 375 psig and 525 psig.

6. High pressure nitrogen system is administratively controlled to prevent inadvertent pressurization of the RCS.

7. Core Flood Tank(s) are isolated as required by the LCO by closing the appropriate isolation valve(s) (either CF-1 and/or CF-2), tagging open the valve breaker(s), and tagging the valve(s) in the closed position.

8. The HPI safety injection flowpaths must be deactivated.

   a. Deactivating Train A of HPI is accomplished by either:

      1) Shutting and deactivating valve HP-26 by tagging open the valve breaker and tagging the valve handwheel in the closed position, shutting valve HP-410 and tagging the valve switch in the closed position.

      2) Deactivating all HPI pumps aligned to HPI train A and tagging the pump breakers open.

   b Deactivating Train B of HPI is accomplished by either:

      1) Shutting and deactivating valve HP-27 by tagging open the valve breaker and tagging the valve handwheel in the closed position, shutting valve HP-409 and tagging the valve switch in the closed position.

      2) Deactivating all HPI pumps aligned to HPI train B and tagging the pump breakers open.

9. Pressurizer heater bank 3 or 4 must be deactivated.

Operational parameters identified in TS 3.4.12 and this TS Bases include allowances for instrument uncertainty.

APPLICABILITY

This LCO is applicable in MODE 3 when any RCS cold leg temperature is ≤ 325°F, and in MODES 4, 5 and 6 when an RCS vent capable of mitigating the most limiting LTOP event is not open. The Applicability

BASES

APPLICABILITY (continued)

temperature of 325°F is established by fracture mechanics analyses. The pressurizer safety valves provide overpressure protection to meet LCO 3.4.3 P/T limits above 325°F. With the vessel head off, overpressurization is not possible. With an RCS vent capable of mitigating the most limiting LTOP event open, an LTOP event (including HPI-ES actuation or CFT discharge) is incapable of pressurizing the RCS above the RCS P/T limits.

A RCS vent ≥ 3.6 square inches is capable of mitigating a HPI-ES actuation of three pumps through two flow paths to the RCS. A RCS vent ≥ 201 square inches is capable of mitigating a discharge of both CFTs.

LCO 3.4.3 provides the operational P/T limits for all MODES. LCO 3.4.10, "Pressurizer Safety Valves," requires the pressurizer safety valves OPERABLE to provide overpressure protection during MODES 1, 2, and 3 above 325°F.

The Applicability is modified by two Notes. Note 1 states that CFT isolation is only required when the CFT pressure is more than or equal to the maximum RCS pressure for the existing RCS temperature, as allowed in LCO 3.4.3. This Note permits the CFT discharge valve surveillance performed only under these pressure and temperature conditions.

Note 2 permits the PORV to be inoperable when no HPI pumps are running and RCS pressure is < 100 psig. PORV operability is not required when RCS pressure is < 100 psig and HPI pumps are not operating since credible LTOP events progress relatively slowly, thus giving the operator ample time to respond.

ACTIONS

A.1

With the HPI activated, immediate actions are required to deactivate HPI. Emphasis is on immediate deactivation because inadvertent injection with one or more HPI pump OPERABLE is the event of greatest significance, since these events cause the greatest pressure increase in the shortest time.

The immediate Completion Times reflect the urgency of quickly proceeding with the Required Actions.

B.1, C.1, and C.2

An unisolated CFT requires isolation within 1 hour only when the CFT pressure is at or more than the maximum RCS pressure for the existing temperature allowed in LCO 3.4.3.

BASES

ACTIONS

B.1, C.1, and C.2 (continued)

If isolation is needed and cannot be accomplished in 1 hour, Required Action C.1 and Required Action C.2 provide two options, either of which must be performed in 12 hours. By placing the unit in MODE 4 with the RCS temperature > 200°F, the CFT pressure of 650 psig cannot exceed the LTOP limits if both tanks are fully injected. Depressurizing the CFTs below the LTOP limit of 373 psig also prevents exceeding the LTOP limits in the same event.

The Completion Times are based on operating experience that these activities can be accomplished in these time periods and on engineering judgement indicating that a limiting LTOP event is not likely in the allowed times.

D.1, E.1, and E.2

With the PORV inoperable, overpressure relieving capability is lost, and restoration of the PORV within 1 hour is required.

If restoration cannot be completed within 1 hour, either Required Action E.1 or Required Action E.2 must be performed. Required Action E.1 requires increasing RCS temperature within 23 hours to exit the Applicability of the specification. With RCS temperature > 325°F, the CFTs are not required to be isolated. Required Action E.2 requires the RCS be depressurized to less than 100 psig within 35 hours. With reactor pressure < 100 psig more time is available for operator action to mitigate an LTOP event.

These Completion Times also consider these activities can be accomplished in these time periods. A limiting LTOP event is not likely in these times.

F.1 and G.1

With Administrative Controls that assure ≥ 10 minutes are available to mitigate the consequences of an event not implemented, the capability for operator action to mitigate an LTOP event may be lost. In this circumstance, compensatory measures must be established to monitor for initiation of an LTOP event. Establishing a dedicated operator within 4 hours to monitor for initiation of an LTOP event is sufficient to compensate for inoperability of makeup flow restrictions, having too many pressurizer heater banks available, inoperability of required alarms, or deviation from pressure, temperature or level limits. Establishing a dedicated operator is not sufficient to compensate for not deactivating HPI or isolating CFTs. If the Required Action and associated Completion Time of Condition F is not met, the RCS must be depressurized and an

BASES

ACTIONS

F.1 and G.1 (continued)

RCS vent path capable of mitigating the most limiting LTOP event must be established within 12 hours. These Completion Times also consider that these activities can be accomplished in these time periods. A limiting LTOP event is not likely in these periods.

H.1 and H.2

With administrative controls which assure ≥ 10 minutes are available to mitigate the consequences of an LTOP event not implemented and the PORV inoperable; or the LTOP System inoperable for any reason other than cited in Condition A through G, the system must be restored to OPERABLE status within one hour. When this is not possible, Required Action H.2 requires the RCS depressurized and vented within 12 hours.

One or more vents may be used. A vent path capable of mitigating the most limiting LTOP event is specified. Because makeup may be required, the vent size accommodates inadvertent full makeup system operation. Such a vent keeps the pressure from full flow of the makeup pump(s) with a wide open makeup control valve within the LCO limit.

The Completion Time is based on operating experience that these activity can be accomplished in this time period and on engineering judgement indicating that a limiting LTOP transient is not likely in this time.

SURVEILLANCE REQUIREMENTS

SR 3.4.12.1 and SR 3.4.12.2

Verifications must be performed that HPI is deactivated, and the CFTs are isolated. These Surveillances ensure the minimum coolant input capability will not create an RCS overpressure condition to challenge the LTOP System. The Surveillances are required at 12 hour intervals. The 12 hour intervals are shown by operating practice to be sufficient to regularly assess conditions for potential degradation and verify operation within the safety analysis.

SR 3.4.12.3

Verification that the pressurizer level is less than the volume necessary to assure ≥ 10 minutes are available for operator action to mitigate an LTOP event by observing control room or other indications ensures a cushion of sufficient size is available to reduce the rate of pressure increase from potential transients.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.12.3 (continued)

The 30 minute Surveillance Frequency during heatup and cooldown must be performed for the LCO Applicability period when temperature changes can cause pressurizer level variations. This Frequency may be discontinued when the ends of these conditions are satisfied, as defined in plant procedures. Thereafter, the Surveillance is required at 12 hour intervals.

These Frequencies are shown by operating practice sufficient to regularly assess indications of potential degradation and verify operation within the safety analysis.

SR 3.4.12.4

Verification that the PORV block valve is open ensures a flow path to the PORV. This is required at 12 hour intervals.

The interval has been shown by operating practice to be sufficient to regularly assess conditions for potential degradation and verify operation is within the safety analysis.

SR 3.4.12.5

A CHANNEL FUNCTIONAL TEST is required within 12 hours after decreasing RCS temperature to ≤ 325°F and every 31 days thereafter to ensure the setpoint is proper for using the PORV for LTOP. PORV actuation is not needed, as it could depressurize the RCS.

The 12 hour Frequency considers the unlikelihood of a low temperature overpressure event during the time. The 31 day Frequency is based on industry accepted practice and is acceptable by experience with equipment reliability.

SR 3.4.12.6

Verification that administrative controls, other than limits for pressurizer level, that assure ≥ 10 minutes are available for operator action to mitigate the consequences of an LTOP event are implemented is necessary every 12 hours. This verification consists of a combination of administrative checks for alarm availability, verification that pressurizer heater bank 3 or 4 is deactivated, appropriate restrictions on pressurizer level, controls for High Pressure Nitrogen, etc., as well as visual confirmation using available indications that associated physical parameters are within limits.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.12.6 (continued)

The Frequency is shown by operating practice sufficient to regularly assess indications of potential degradation and verify operation within the safety analysis.

SR 3.4.12.7

The performance of a CHANNEL CALIBRATION is required every 18 months. The CHANNEL CALIBRATION for the LTOP setpoint ensures that the PORV will be actuated at the appropriate RCS pressure by verifying the accuracy of the instrument string. The calibration can only be performed in shutdown.

The Frequency considers a typical refueling cycle and industry accepted practice.

REFERENCES

1. 10 CFR 50, Appendix G.
2. Generic Letter 88-11.
3. UFSAR, 5.2.3.7.
4. 10 CFR 50.46.
5. 10 CFR 50, Appendix K.
6. 10 CFR 50.36.

## B 3.4 REACTOR COOLANT SYSTEM (RCS)

## B 3.4.13 RCS Operational LEAKAGE

BASES

BACKGROUND

Components that contain or transport the coolant to or from the reactor core make up the RCS. Component joints are made by welding, bolting, rolling, or pressure loading, and valves isolate connecting systems from the RCS.

During unit life, the joint and valve interfaces can produce varying amounts of reactor coolant LEAKAGE, through either normal operational wear or mechanical deterioration. The purpose of the RCS Operational LEAKAGE LCO is to limit system operation in the presence of LEAKAGE from these sources to amounts that do not compromise safety. This LCO specifies the types and amounts of LEAKAGE.

The safety significance of RCS LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring reactor coolant LEAKAGE into the containment area are necessary. Separating the identified LEAKAGE from the unidentified LEAKAGE is necessary to provide quantitative information to the operators, allowing them to take corrective action should a leak occur detrimental to the safety of the facility and the public.

This LCO deals with protection of the reactor coolant pressure boundary (RCPB) from degradation and the core from inadequate cooling, in addition to preventing the accident analysis radiation release assumptions from being exceeded. The consequences of violating this LCO include the possibility of a loss of coolant accident (LOCA). However, the ability to monitor leakage provides advance warning to permit unit shutdown before a LOCA occurs. This advantage has been shown by "leak before break" studies.

APPLICABLE SAFETY ANALYSES

Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event. The steam line break (SLB) and Loss of Load Safety analyses assume total primary to secondary LEAKAGE greater than 300 gallon per day as the initial condition.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Primary to secondary LEAKAGE is a factor in the dose releases outside containment resulting from a SLB accident. To a lesser extent, other accidents or transients involve secondary steam release to the atmosphere, such as a steam generator tube rupture (SGTR). The leakage contaminates the secondary fluid and can be released to the environment.

The safety analysis assumptions for the SLB accident bounds 300 gallon per day primary to secondary LEAKAGE in one generator as an initial condition. The dose consequences resulting from the SLB accident are within the limits defined in 10 CFR 100.

RCS operational LEAKAGE satisfies Criterion 2 of 10 CFR 50.36 (Ref.3).

LCO

RCS LEAKAGE includes leakage from connected systems up to and including the second normally closed valve for systems which do not penetrate containment and the outermost isolation valve for systems which penetrate containment. Loss of reactor coolant through reactor coolant pump seals and system valves to connecting systems which vent to the gas vent header and from which coolant can be returned to the RCS shall not be considered as RCS LEAKAGE.

RCS operational LEAKAGE shall be limited to:

a. Pressure Boundary LEAKAGE

No pressure boundary LEAKAGE is allowed, being indicative of material deterioration. LEAKAGE of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher LEAKAGE. Violation of this LCO could result in continued degradation of the RCPB. LEAKAGE past seals, gaskets, and steam generator tubes is not pressure boundary LEAKAGE.

b. Unidentified LEAKAGE

One gallon per minute (gpm) of unidentified LEAKAGE is allowed as a reasonable minimum detectable amount that the containment air monitoring and containment sump level monitoring equipment can detect within a reasonable time period. Violation of this LCO could result in continued degradation of the RCPB, if the LEAKAGE is from the pressure boundary.

BASES

LCO (continued)

c. Identified LEAKAGE

Up to 10 gpm of identified LEAKAGE is considered allowable because LEAKAGE is from known sources that do not interfere with detection of unidentified LEAKAGE and is well within the capability of the RCS makeup system. Identified LEAKAGE includes LEAKAGE to the containment from specifically known and located sources, but does not include pressure boundary LEAKAGE or controlled reactor coolant pump (RCP) seal leakoff (a normal function not considered LEAKAGE). Violation of this LCO could result in continued degradation of a component or system.

d. Primary to Secondary LEAKAGE through All Steam Generators (SGs)

Total primary to secondary LEAKAGE amounting to 300 gallon per day through all SGs produces acceptable offsite doses in the SLB accident analysis. Violation of this LCO could exceed the offsite dose limits for this accident. Primary to secondary LEAKAGE must be included in the total allowable limit for identified LEAKAGE.

e. Primary to Secondary LEAKAGE through Any One SG

The 150 gallon per day limit on one SG is equivalent to a total of 300 gallon per day primary to secondary LEAKAGE allocated equally between the two generators.

APPLICABILITY

In MODES 1, 2, 3, and 4, the potential for RCPB LEAKAGE is greatest when the RCS is pressurized.

In MODES 5 and 6, LEAKAGE limits are not required because the reactor coolant pressure is far lower, resulting in lower stresses and reduced potentials for LEAKAGE.

LCO 3.4.14, "RCS Pressure Isolation Valve (PIV) Leakage," measures leakage through each individual PIV and can impact this LCO. Of the two PIVs in series in each isolated line, leakage measured through one PIV does not result in RCS LEAKAGE when the other is leaktight. If both valves leak and result in a loss of mass from the RCS, the loss must be included in the allowable identified LEAKAGE.

BASES (continued)

ACTIONS

A.1

If unidentified LEAKAGE, identified LEAKAGE, or primary to secondary LEAKAGE are in excess of the LCO limits, the LEAKAGE must be reduced to within limits within 4 hours. This Completion Time allows time to verify leakage rates and either identify unidentified LEAKAGE or reduce LEAKAGE to within limits before the reactor must be shut down. This action is necessary to prevent further deterioration of the RCPB.

B.1 and B.2

If any pressure boundary LEAKAGE exists or if unidentified, identified, or primary to secondary LEAKAGE cannot be reduced to within limits within 4 hours, the reactor must be brought to lower pressure conditions to reduce the severity of the LEAKAGE and its potential consequences. The reactor must be brought to MODE 3 within 12 hours and MODE 5 within 36 hours. This action reduces the LEAKAGE and also reduces the factors that tend to degrade the pressure boundary.

The Completion Times allowed are reasonable, based on operating experience, to reach the required conditions from full power conditions in an orderly manner and without challenging unit systems. In MODE 5, the pressure stresses acting on the RCPB are much lower and further deterioration is much less likely.

SURVEILLANCE REQUIREMENTS

SR 3.4.13.1

Evaluation of RCS LEAKAGE ensures identified and unidentified leakage is maintained within the associated LCO limits and ensures that the integrity of the RCPB is maintained. Identified and unidentified LEAKAGE is determined by performance of an RCS water inventory balance. Primary to secondary LEAKAGE is measured by effluent monitoring within the secondary systems or comparison of primary and secondary radioisotope concentrations. These methods provide the required leakage detection sensitivity to ensure leakage is within limits.

The RCS water inventory balance must be performed with the reactor at steady state operating conditions and near operating pressure. Therefore, a Note is added allowing that this SR is not required to be performed until 12 hours after establishing steady state operation. This 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.13.1 (continued)

Steady state operation is required to perform a proper water inventory balance since calculations during maneuvering are not useful. For RCS operational LEAKAGE determination by water inventory balance, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP pump seal injection and return flows.

An early warning of LEAKAGE is provided by the automatic systems that monitor the containment atmosphere radioactivity and the containment sump level.

These leakage detection systems are specified in LCO 3.4.15, "RCS Leakage Detection Instrumentation."

The 72 hour Frequency is a reasonable interval to trend LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents.

SR 3.4.13.2

This SR provides the means necessary to determine SG OPERABILITY in an operational MODE. The requirement to demonstrate SG tube integrity in accordance with the Steam Generator Tube Surveillance Program emphasizes the importance of SG tube integrity, even though this Surveillance cannot be performed at normal operating conditions.

REFERENCES

1. UFSAR, Section 3.1.

2. UFSAR, Chapter 15.

3. 10 CFR 50.36.

## B 3.4 REACTOR COOLANT SYSTEM (RCS)

## B 3.4.14 RCS Pressure Isolation Valve (PIV) Leakage

BASES

BACKGROUND

10 CFR 50.2 (Ref. 1), 10 CFR 50.55a(c) (Ref. 2), and Ref. 3 define RCS PIVs as any two normally closed valves in series within the RCS pressure boundary that separate the high pressure RCS from an attached low pressure system. During their lives, these valves can produce varying amounts of reactor coolant leakage through either normal operational wear or mechanical deterioration. The RCS PIV Leakage LCO allows RCS high pressure operation when leakage through these valves exists in amounts that do not compromise safety.

The PIV leakage limit applies to each individual valve. Leakage through both series PIVs in a line must be included as part of the identified LEAKAGE, governed by LCO 3.4.13, "RCS Operational LEAKAGE." This is true during operation only when the loss of RCS mass through two series valves is determined by a water inventory balance (SR 3.4.13.1). A known component of the identified LEAKAGE before operation begins is the least of the two individual leakage rates determined for leaking series PIVs during the required surveillance testing; leakage measured through one PIV in a line is not RCS operational LEAKAGE if the other is leaktight.

Although this specification provides a limit on allowable PIV leakage rate, its main purpose is to prevent overpressure failure of the low pressure portions of connecting systems. The leakage limit is an indication that the PIVs between the RCS and the connecting systems are degraded or degrading. PIV leakage could lead to overpressure of the low pressure piping or components. Failure consequences could be a loss of coolant accident (LOCA) outside of containment, an unanalyzed accident that could degrade the ability for low pressure injection.

The basis for this LCO is the 1975 NRC "Reactor Safety Study" (Ref. 4) that identified potential intersystem LOCAs as a significant contributor to the risk of core melt.

A subsequent study (Ref. 5) evaluated various PIV configurations to determine the probability of intersystem LOCAs.

PIVs are provided to isolate the RCS from the Low Pressure Injection (LPI) System.

BASES

BACKGROUND (continued)

Violation of this LCO could result in continued degradation of a PIV, which could lead to overpressurization of a low pressure system and the loss of the integrity of a fission product barrier.

APPLICABLE SAFETY ANALYSES

Reference 4 identified potential intersystem LOCAs as a significant contributor to the risk of core melt. The dominant accident sequence in the intersystem LOCA category is the failure of the low pressure portion of the LPI System outside of containment. The accident is the result of a postulated failure of the PIVs, which are part of the reactor coolant pressure boundary (RCPB), and the subsequent pressurization of the LPI System downstream of the PIVs from the RCS. Because the low pressure portion of the LPI System is designed for pressures significantly less than RCS pressure, overpressurization failure of the LPI low pressure line would result in a LOCA outside containment and subsequent risk of core melt.

Reference 5 evaluated various PIV configurations, leakage testing of the valves, and operational changes to determine the effect on the probability of intersystem LOCAs. This study concluded that periodic leakage testing of the PIVs can substantially reduce the probability of an intersystem LOCA.

RCS PIV leakage satisfies Criterion 2 of 10 CFR 50.36 (Ref. 6).

LCO

RCS PIV leakage is identified LEAKAGE into closed low pressure systems connected to the RCS. PIV leakage is usually on the order of drops per minute. Leakage that increases significantly suggests that something is operationally wrong and corrective action must be taken.

The PIV leakage limit for specified valves is 0.5 gpm per nominal inch of valve size with a maximum limit of 5 gpm. A study concluded a leakage rate limit based on valve size was superior to a single allowable value.

Reference 7 permits leakage testing at a lower pressure differential than between the specified maximum RCS pressure and the normal pressure of the connected system during RCS operation (the maximum pressure differential) in those types of valves in which the higher service pressure will tend to diminish the overall leakage channel opening. In such cases, the observed rate may be adjusted to the maximum pressure differential by assuming leakage is directly proportional to the pressure differential to the one half power.

BASES (continued)

APPLICABILITY

In MODES 1, 2, 3, and 4, this LCO applies because the PIV leakage potential is greatest when the RCS is pressurized. In MODE 4, valves in the DHR flow path are not required to meet the requirements of this LCO when in, or during the transition to or from, the DHR mode of operation.

In MODES 5 and 6, leakage limits are not provided because the lower reactor coolant pressure results in a reduced potential for leakage and for a LOCA outside the containment.

ACTIONS

The ACTIONS are modified by two Notes. Note 1 is added to provide clarification that each flow path allows separate entry into a Condition. This is allowed based upon the functional independence of the flow path. Note 2 requires an evaluation of affected systems if a PIV is inoperable. The leakage may have affected system OPERABILITY, or isolation of a leaking flow path with an alternate valve may have degraded the ability of the interconnected system to perform its safety function.

A.1 and A.2

The flow path with leakage must be isolated by two valves. Required Actions A.1 and A.2 are modified by a Note that the valves used for isolation must meet the same leakage requirements as the PIVs and must be on the RCS pressure boundary or the high pressure portion of the system.

Required Action A.1 requires that the isolation with one valve must be performed within 4 hours. Four hours provides time to reduce leakage in excess of the allowable limit and to isolate the affected system if leakage cannot be reduced. The 4 hours allows the actions and restricts the operation with leaking isolation valves.

Required Action A.2 specifies that the double isolation barrier of two valves be restored by closing some other valve qualified for isolation. The 72 hour time after exceeding the limit considers the time required to complete the Action and the low probability of a second valve failing during this time period.

B.1 and B.2

If Required Actions and associated Completion Times are not met, the unit must be brought to a MODE in which the requirement does not apply.

BASES

ACTIONS

B.1 and B.2 (continued)

To achieve this status, the unit must be brought to MODE 3 within 12 hours and to MODE 5 within 36 hours. This Required Action may reduce the leakage and also reduces the potential for a LOCA outside the containment. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1

Performance of leakage testing on each required RCS PIV or isolation valve used to satisfy Required Action A.1 or A.2 is required to verify that leakage is below the specified limit and to identify each leaking valve. The leakage limit of 0.5 gpm per inch of nominal valve diameter up to 5 gpm maximum applies to each valve. Leakage testing requires a stable pressure condition.

For the two PIVs in series, the leakage requirement applies to each valve individually and not to the combined leakage across both valves. If the PIVs are not individually leakage tested, one valve may have failed completely and not detected if the other valve in series meets the leakage requirement. In this situation, the protection provided by redundant valves would be lost.

Testing is to be performed every 18 months, a typical refueling cycle, if the unit does not go into MODE 5 for at least 7 days. The 18 month Frequency is consistent with 10 CFR 50.55a(g) (Ref. 8) as contained in the Inservice Testing Program, is within frequency allowed by the American Society of Mechanical Engineers (ASME) Code, Section XI (Ref. 7), and is based on the need to perform such surveillances under conditions that apply during an outage and the potential for an unplanned transient if the Surveillance were performed with the unit at power.

The leakage limit is to be met at the RCS pressure associated with MODES 1 and 2. This permits leakage testing at high differential pressures with stable conditions not possible in the MODES with lower pressures.

To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.

SURVEILLANCE REQUIREMENTS

SR [illegible]

Entry into MODES 3 and 4 is allowed to establish the necessary differential pressures and stable conditions to allow for performance of this Surveillance. The Note that allows this provision is complimentary to the Frequency of prior to entry into MODE 2 whenever the unit has been in MODE 5 for 7 days or more, if leakage testing has not been performed in the previous 9 months. In addition, this Surveillance is not required to be performed on the LPI System when the LPI System is aligned to the RCS in the decay heat removal mode of operation. PIVs contained in the DHR flow path must be leakage rate tested after DHR is secured and stable unit conditions and the necessary differential pressures are established. For the purposes of meeting this SR, test activities including contingencies may be performed prior to declaring a PIV inoperable. A PIV will be considered "in testing" until the test procedure is complete, or the test coordinator determines that further test contingencies would not be expected to produce an acceptable result.

REFERENCES


1. 10 CFR [illegible]

2. [illegible]

3. NRC letter to DPC, "Order for Modification of License Concerning Primary Coolant System Pressure Isolation Valves," dated April 20, 1981.

4. NUREG-75/014, Appendix V, October 1975.

5. NUREG-0677, NRC, May 1980.

6. 10 CFR 50.36.

7. ASME, Boiler and Pressure Vessel Code, Section XI.

8. 10 CFR 50.55a(g).




(Bases Change of 12-23-99)

## B 3.4 REACTOR COOLANT SYSTEM (RCS)

### B 3.4.15 RCS Leakage Detection Instrumentation

BASES

BACKGROUND

ONS Design Criteria (Ref. 1) requires means for detecting RCS LEAKAGE.

Leakage detection systems must have the capability to detect significant reactor coolant pressure boundary (RCPB) degradation as soon after occurrence as practical to minimize the potential for propagation to a gross failure. Thus, an early indication or warning signal is necessary to permit proper evaluation of all unidentified LEAKAGE.

Industry practice has shown that water flow changes of 0.5 to 1.0 gpm can readily be detected in contained volumes by monitoring changes in water level, in flow rate, or in the operating frequency of a pump.

The reactor coolant contains radioactivity that, when released to the containment, can be detected by radiation monitoring instrumentation. Reactor coolant radioactivity levels will be low during initial reactor startup and for a few weeks thereafter until activated corrosion products have been formed and fission products appear from fuel element cladding contamination or cladding defects. Instrument sensitivities of $10^{-9}$ $\mu$Ci/cc radioactivity for particulate monitoring and of $10^{-6}$ $\mu$Ci/cc radioactivity for gaseous monitoring are practical for these leakage detection systems. Radioactivity detection systems are included for monitoring both particulate and gaseous activities because of their sensitivities and rapid responses to RCS LEAKAGE.

Air temperature and pressure monitoring methods may also be used to infer unidentified LEAKAGE to the containment. Containment temperature and pressure fluctuate slightly during unit operation, but a rise above the normally indicated range of values may indicate RCS LEAKAGE into the containment. The relevance of temperature and pressure measurements are affected by containment free volume and, for temperature, detector location. Signals from these instruments can be valuable in recognizing rapid and sizable leakage to the containment. Temperature and pressure monitors are not required by this LCO.

BASES (continued)

APPLICABLE SAFETY ANALYSES

The need to evaluate the severity of an alarm or an indication is important to the operators, and the ability to compare and verify with indications from other systems is necessary.

The safety significance of RCS LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring reactor coolant LEAKAGE into the containment area are necessary. Separating the identified LEAKAGE from the unidentified LEAKAGE provides quantitative information to the operators, allowing them to take corrective action should a leak occur detrimental to the safety of the unit and the public.

RCS leakage detection instrumentation satisfies Criterion 1 of 10 CFR 50.36 (Ref. 2).

LCO

One method of protecting against large RCS LEAKAGE derives from the ability of instruments to rapidly detect small leaks. This LCO requires instruments of diverse monitoring principles to be OPERABLE to provide a high degree of confidence that small leaks are detected in time to allow actions to place the unit in a safe condition when RCS LEAKAGE indicates possible RCPB degradation.

The LCO requirements are satisfied when instruments of diverse measurement means are available. Thus, the containment normal sump level indication, in combination with a particulate (RIA-47) or gaseous radioactivity monitor (RIA-49), provides an acceptable minimum.

APPLICABILITY

Because of elevated RCS temperature and pressure in MODES 1, 2, 3, and 4, RCS leakage detection instrumentation is required to be OPERABLE.

In MODE 5 or 6, the temperature is ≤ 200°F and pressure is maintained low or at atmospheric pressure. Since the temperatures and pressures are far lower than those for MODES 1, 2, 3, and 4, the likelihood of leakage and crack propagation is much smaller. Therefore, the requirements of this LCO are not applicable in MODES 5 and 6.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.15.2

SR 3.4.15.2 requires the performance of a CHANNEL FUNCTIONAL TEST of the required containment atmosphere radioactivity monitor. The test ensures that the monitor can perform its function in the desired manner. The test verifies the alarm setpoint and relative accuracy of the instrument string. The Frequency of 92 days considers instrument reliability, and operating experience has shown it proper for detecting degradation.

SR 3.4.15.3 and SR 3.4.15.4

These SRs require the performance of a CHANNEL CALIBRATION for each of the required RCS leakage detection instrumentation channels. The calibration verifies the accuracy of the instrument string, including the instruments located inside containment. The Frequency of 18 months is a typical refueling cycle and considers channel reliability. Industry operating experience has proven this Frequency is acceptable.

REFERENCES

1. UFSAR, Section 3.1.

2. 10 CFR 50.36.

## B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

### B 3.5.1 Core Flood Tanks (CFTs)

BASES

BACKGROUND

The function of the ECCS CFTs is to supply water to the reactor vessel during the blowdown phase of a loss of coolant accident (LOCA), to provide inventory to help accomplish the refill phase that follows thereafter, and to provide Reactor Coolant System (RCS) makeup for a small break LOCA. Two CFTs are provided for these functions.

The blowdown phase of a large break LOCA is the initial period of the transient during which the RCS departs from equilibrium conditions, and heat from fission product decay, hot internals, and the vessel continues to be transferred to the reactor coolant. The blowdown phase of the transient ends when the RCS pressure falls to a value approaching that of the containment atmosphere.

In the refill phase of a LOCA, which follows immediately, reactor coolant inventory has vacated the core through steam flashing and ejection through the break. The core is essentially in adiabatic heatup. The balance of inventory is then available to help fill voids in the lower plenum and reactor vessel downcomer so as to establish a recovery level at the bottom of the core and ongoing reflood of the core with the addition of safety injection water.

The CFTs are pressure vessels partially filled with borated water and pressurized with nitrogen gas. The CFTs are passive components, since no operator or control actions are required for them to perform their function. Internal tank pressure is sufficient to discharge the contents of the CFTs to the RCS if RCS pressure decreases below the CFT pressure. Each CFT is piped separately into the reactor vessel downcomer. The CFT injection lines are also utilized by the Low Pressure Injection (LPI) System. Each CFT is isolated from the RCS by a motor operated isolation valve and two check valves in series. The motor operated isolation valves are normally open, with power removed from the valve motor to prevent inadvertent closure prior to or during an accident.

The CFTs thus form a passive system for injection directly into the reactor vessel. Except for the core flood line break LOCA, a unique accident that also disables a portion of the injection system, both tanks are assumed to

BASES

BACKGROUND (continued)

operate in the safety analyses for Design Basis Events. Because injection is directly into the reactor vessel downcomer, and because it is a passive system not subject to the single active failure criterion, all fluid injection is credited for core cooling.

The CFT gas/water volumes, gas pressure, and outlet pipe size are selected to provide core cooling for a large break LOCA prior to the injection of coolant by the LPI System.

APPLICABLE SAFETY ANALYSES

The CFTs are credited in both the large and small break LOCA analyses (Ref. 1). These accident analyses establish the acceptance limits for the CFTs. In performing the LOCA calculations, conservative assumptions are made concerning the availability of emergency injection flow. The assumption of the loss of offsite power is required by regulations. In the early stages of a LOCA with the loss of offsite power, the CFTs provide the sole source of makeup water to the RCS.

This is because the LPI pumps and high pressure injection (HPI) pumps cannot deliver rated flow until the Keowee Hydro Units start and come to rated speed and valves open.

The limiting large break LOCA is a double ended guillotine cold leg break at the discharge of the reactor coolant pump. During this event, the CFTs discharge to the RCS as soon as RCS pressure decreases below CFT pressure. As a conservative estimate, no credit is taken for HPI for large break LOCAs. LPI is not assumed to occur until 38 seconds after loss of offsite power occurs with full LPI flow not occurring until 15 seconds later, or 53 seconds total. No operator action is assumed during the blowdown stage of a large break LOCA.

The small break LOCA analysis also assumes a time delay after Engineered Safeguards actuation before pumped flow reaches the core. For the larger range of small breaks, the rate of blowdown is such that the increase in fuel clad temperature is terminated by the CFTs, with pumped flow then providing continued cooling. As break size decreases, the CFTs and HPI pumps both play a part in terminating the rise in clad temperature. As break size continues to decrease, the role of the CFTs continues to decrease until the tanks are not required and the HPI pumps become responsible for terminating the temperature increase.

BASES

APPLICABLE SAFETY ANALYSES (continued)

This LCO helps to ensure that the following acceptance criteria for the ECCS, established by 10 CFR 50.46 (Ref. 2), will be met following a LOCA:

a. Maximum fuel element cladding temperature of 2200°F;

b. Maximum cladding oxidation of ≤ 0.17 times the total cladding thickness before oxidation;

c. Maximum hydrogen generation from a zirconium water reaction of ≤ 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react; and

d. Core maintained in a coolable geometry.

Since the CFTs discharge during the blowdown phase of a LOCA, they do not contribute to the long term cooling requirements of 10 CFR 50.46.

The limits for operation with a CFT that is inoperable for any reason other than the boron concentration not being within limits minimize the time that the unit is exposed to a LOCA event occurring along with failure of a CFT, which might result in unacceptable peak cladding temperatures. If a closed isolation valve cannot be opened, or the proper water volume or nitrogen cover pressure cannot be restored, the full capability of one CFT is not available and prompt action is required to place the reactor in a MODE in which this capability is not required.

In addition to LOCA analyses, the CFTs have been assumed to operate to provide borated water for reactivity control for severe overcooling events such as a main steam line break (MSLB).

The CFTs are part of the primary success path that functions or actuates to mitigate an accident that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

The minimum volume requirement for the CFTs ensures that both CFTs can provide adequate inventory to reflood the core (to the hot spot) and downcomer following a LOCA. The downcomer then remains flooded until the HPI and LPI systems start to deliver flow.

The maximum volume limit is based upon the need to maintain adequate gas volume to ensure proper injection, ensure the ability of the CFTs to fully discharge, and limit the maximum amount of boron inventory in the

BASES

APPLICABLE SAFETY ANALYSES (continued)

CFTs. The specified values (1010 $ft^3$ and 1070 $ft^3$) are allowable values. The corresponding CFT levels are 12.56 ft and 13.44 ft (allowable values).

The minimum nitrogen cover pressure requirement of 575 psig (allowable value) ensures that the contained gas volume will generate discharge flow rates during injection that are consistent with those assumed in the safety analysis.

The maximum nitrogen cover pressure limit of 625 psig (allowable value) ensures that the amount of CFT inventory that is discharged while the RCS depressurizes, and is therefore lost through the break, will not be larger than that predicted by the safety analysis.

The maximum allowable boron concentration specified in the COLR for the CFTs ensures that boron precipitation will not occur following a LOCA.

The minimum boron requirement of the COLR is selected to ensure that the reactor will remain subcritical during the reflood stage of a large break LOCA. During a large break LOCA, all CONTROL RODS are assumed not to insert into the core until reflood, and the initial reactor shutdown is accomplished by void formation during blowdown. Sufficient boron concentration must be maintained in the CFTs to prevent a return to criticality during reflood. After reflood, the analysis assumes one half of the CONTROL ROD worth is available.

The CFT isolation valves are not single failure proof; therefore, whenever these valves are open, power shall be removed from them. This precaution ensures that both CFTs are available during an accident. With power supplied to the valves, operator error could result in a valve closure, which would render one CFT unavailable for injection. Both CFTs are required to function in the event of a large break LOCA.

The CFTs satisfy Criterion 3 of 10 CFR 50.36 (Ref. 3).

LCO

The LCO establishes the minimum conditions required to ensure that the CFTs are available to accomplish their core cooling safety function following a LOCA. Both CFTs are required to function in the event of a large break LOCA. If the entire contents of both tanks are not injected during the blowdown phase of a large break LOCA, the ECCS acceptance criteria of 10 CFR 50.46 (Ref. 2) could be violated. For a CFT to be considered OPERABLE, the isolation valve must be fully open, power removed when RCS pressure is above 800 psig and the limits established in the SR for contained volume, boron concentration, and nitrogen cover pressure must be met.

BASES (continued)

APPLICABILITY

In MODES 1 and 2, and in MODE 3 with RCS pressure > 800 psig, the CFT OPERABILITY requirements are based on full power operation. Although cooling requirements may decrease as power decreases, the CFTs are still required to provide core cooling as long as elevated RCS pressures and temperatures exist.

This LCO is only applicable at pressures > 800 psig. At or below 800 psig, the rate of RCS blowdown is such that the safety injection pumps can provide adequate injection to ensure that peak clad temperature remains below the 10 CFR 50.46 (Ref. 2) limit of 2200°F.

In MODE 3 with RCS pressure ≤ 800 psig, and in MODES 4, 5, and 6, the CFT motor operated isolation valves may be closed to isolate the CFTs from the RCS. This allows RCS cooldown and depressurization without discharging the CFTs into the RCS or requiring depressurization of the CFTs.

In addition, LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP)," requires that in MODE 3 when any RCS cold leg temperature is ≤ 325°F, MODE 4, MODE 5, and MODE 6 when a vent path capable of mitigating the most limiting LTOP event is not open, each CFT whose pressure is greater than or equal to the maximum RCS pressure for the existing RCS temperature allowed by the pressure and temperature limit curves provided in LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," be deactivated.

ACTIONS

A.1

If the boron concentration of one CFT is not within limits, it must be returned to within the limits within 72 hours. In this condition, ability to maintain subcriticality may be reduced, but the effects of reduced boron concentration on core subcriticality during reflood are minor. Boiling of the ECCS water in the core during reflood concentrates the boron in the saturated liquid that remains in the core. In addition, the volume of the CFT is still available for injection. Since the boron requirements are based on the average boron concentration of the total volume of two CFTs, the consequences are less severe than they would be if the contents of a CFT were not available for injection. Thus, 72 hours is allowed to return the boron concentration to within limits.

BASES

ACTIONS (continued)

B.1

If one CFT is inoperable for a reason other than boron concentration, the CFT must be returned to OPERABLE status within 1 hour. In this condition it cannot be assumed that the CFT will perform its required function during a LOCA. Due to the severity of the consequences should a LOCA occur in these conditions, the 1 hour Completion Time to open the valve, remove power to the valve, or restore the proper water volume or nitrogen cover pressure ensures that prompt action will be taken to return the inoperable CFT to OPERABLE status. The Completion Time minimizes the time the unit is potentially exposed to a LOCA in these conditions.

C.1 and C.2

If the Required Actions and associated Completion Times of Condition A or B are 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 12 hours and RCS pressure reduced to ≤ 800 psig within 18 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.

D.1

If more than one CFT is inoperable, the unit is in a condition outside the accident analysis; therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.5.1.1

Verification every 12 hours that each CFT isolation valve is fully open ensures that the CFTs are available for injection and ensures timely discovery if a valve should be less than fully open. If an isolation valve is not fully open, the rate of injection to the RCS would be reduced. Although a motor operated valve position should not change with power removed, a closed valve could result in accident analysis assumptions not being met. A 12 hour Frequency is considered reasonable in view of administrative controls that ensure that a mispositioned isolation valve is unlikely.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.1.2 and SR 3.5.1.3

Verification every 12 hours of each CFT's nitrogen cover pressure (≥ 575 psig and ≤ 625 psig) and the borated water volume (≥ 1010 $ft^3$ and ≤ 1070 $ft^3$) is sufficient to ensure adequate injection during a LOCA. A CFT level of ≥ 12.56 ft and ≤ 13.44 ft corresponds to the specified borated water volume. Due to the static design of the CFTs, a 12 hour Frequency usually allows the operator to identify changes before the limits are reached. Operating experience has shown that this Frequency is appropriate for early detection and correction of off normal trends.

SR 3.5.1.4

Surveillance once every 31 days is reasonable to verify that the CFT boron concentration is within the required limits, because the static design of the CFT limits the ways in which the concentration can be changed. The Frequency is adequate to identify changes that could occur from mechanisms such as stratification or inleakage. Verifying CFT boron concentration within 12 hours after an 80 gallon volume increase will identify whether inleakage from the RCS has caused a reduction in boron concentration to below the required limit. The 80 gallon increase represents approximately 1% increase in volume. It is not necessary to verify boron concentration if the added water inventory is from a borated water source that meets CFT boron concentration requirements, such as the boric acid mix tank or the borated water storage tank (BWST). This is consistent with the recommendations of NUREG-1366 (Ref. 4).

SR 3.5.1.5

Verification every 31 days that power is removed from each CFT isolation valve operator ensures that an active failure could not result in the undetected closure of a CFT motor operated isolation valve coincident with a LOCA. Since power is removed under administrative control, the 31 day Frequency will provide adequate assurance that the power is removed.

BASES (continued)

REFERENCES

1. UFSAR, Section 15.14.

2. 10 CFR 50.46.

3. 10 CFR 50.36.

4. NUREG-1366, "Improvements to Technical Specifications Surveillance Requirements," December 1992.

## B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

### B 3.5.2 High Pressure Injection (HPI)

BASES

BACKGROUND

The function of the ECCS is to provide core cooling to ensure that the reactor core is protected after any of the following accidents:

a. Loss of coolant accident (LOCA);

b. Rod ejection accident (REA);

c. Steam generator tube rupture (SGTR); and

d. Main steam line break (MSLB).

There are two phases of ECCS operation: injection and recirculation. In the injection phase, all injection is initially added to the Reactor Coolant System (RCS) via the cold legs or Core Flood Tank (CFT) lines to the reactor vessel. After the borated water storage tank (BWST) has been depleted, the recirculation phase is entered as the suction is transferred to the reactor building sump.

The HPI System consists of two independent trains, each of which splits to discharge into two RCS cold legs, so that there are a total of four HPI injection lines. Each train takes suction from the BWST, and has an automatic suction valve and discharge valve which open upon receipt of an Engineered Safeguards Protective System (ESPS) signal. The two HPI trains are designed and aligned such that they are not both susceptible to any single active failure including the failure on any power operating component to operate or any single failure of electrical equipment. There are three ESPS actuated HPI pumps, each of which can provide flow to either train. At least one pump is normally running providing RCS makeup and seal injection to the reactor coolant pumps. Suction header cross-connect valves are normally open, and discharge header cross-connect valves are normally closed. Additional discharge valves (HPI discharge crossover valves) can be used to bypass the normal discharge valves and assure the ability to feed either train's injection lines from the pump(s) on the other train. A safety grade flow indicator is provided for the flow path associated with each of these four discharge valves. These indicators are required to be OPERABLE to support HPI OPERABILITY and are needed to throttle HPI flow during an accident to assure that runout limits are not exceeded.

BASES

BACKGROUND (continued)

To fulfill HPI ECCS heat removal requirements during a small break LOCA with the reactor above 350°F, one HPI pump is assumed to inject immediately through one HPI train upon ESPS actuation. If THERMAL POWER is above 60% RTP, there are additional HPI System heat removal requirements to mitigate the consequences of certain small break LOCAs. Three HPI pumps must be OPERABLE to ensure adequate cooling in response to the design basis RCP discharge small break LOCA. If one HPI train fails to actuate, and the break location is such that full flow from only one of the two injection lines of the other HPI train actually reaches the reactor, at least one HPI pump is assumed to provide flow through the automatically actuating train and injection through the other HPI train must occur within 10 minutes.

A suction header supplies water from the BWST or the reactor building sump (via the LPI-HPI flow path) to the HPI pumps. HPI discharges into each of the four RCS cold legs between the reactor coolant pump and the reactor vessel. There is one flow limiting orifice in each of the four injection headers that connect to the RCS cold legs. If a pipe break were to occur in an HPI line between the last check valve and the RCS, the orifice in the broken line would limit the HPI flow lost through the break and increase the flow supplied to the reactor vessel via the other line supplied by the HPI header.

The HPI pumps are capable of discharging to the RCS at an RCS pressure above the opening setpoint of the pressurizer safety valves. The HPI pumps cannot take suction directly from the sump. If the BWST is emptied and HPI is still needed, a cross connect from the discharge side of the LPI pump to the suction of the HPI pumps would be opened. This is known as "piggy backing" HPI to LPI and enables continued HPI to the RCS.

The HPI System also functions to supply borated water to the reactor core following increased heat removal events, such as MSLBs.

During a small break LOCA, the HPI System supplies makeup water to the reactor vessel via the RCS cold legs. The HPI System is actuated upon receipt of an ESPS signal. If offsite power is available, the safeguard loads start immediately. If offsite power is not available, the Engineered Safeguards (ES) buses are connected to the Keowee Hydro Units. The time delay associated with Keowee Hydro Unit startup, HPI valve opening, and pump starting determines the time required before pumped flow is available to the core following a LOCA.

BASES

BACKGROUND (continued)

The HPI and LPI (LCO 3.5.3, "Low Pressure Injection (LPI)") components, along with the passive CFTs and the BWST covered in LCO 3.5.1, "Core Flood Tanks (CFTs)," and LCO 3.5.4, "Borated Water Storage Tank (BWST)," provide the cooling water necessary to meet 10 CFR 50.46 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The LCO helps to ensure that the following acceptance criteria for the ECCS, established by 10 CFR 50.46 (Ref. 1), will be met following a LOCA:

a. Maximum fuel element cladding temperature is ≤ 2200°F;

b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation;

c. Maximum hydrogen generation from a zirconium water reaction is ≤ 0.01 times the hypothetical amount generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react; and

d. Core is maintained in a coolable geometry.

The HPI System is credited in the small break LOCA analysis (Ref. 2). This analysis establishes the minimum required flow and discharge head requirements at the design point for the HPI pumps, as well as the minimum required response time for their actuation. The SGTR and MSLB analyses also credit the HPI pumps but are not limiting in their design.

One HPI pump injecting down one train provides sufficient flow to mitigate most small break LOCAs. However, for cold leg breaks located on the discharge of the reactor coolant pumps, some HPI injection will be lost out the break. For this reason, operator actions are credited to cross-connect the HPI trains when flow in one train is insufficient. The safety analyses have determined that the capacity of one HPI train is sufficient to mitigate a small break LOCA on the discharge of the reactor coolant pumps if THERMAL POWER is ≤ 60% RTP. For THERMAL POWER levels > 60 RTP%, the additional HPI flow obtained by cross-connecting the HPI trains and the second HPI pump is necessary to mitigate the reactor coolant pump discharge break small break LOCA.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Hydraulic separation on the suction side of the HPI pumps could cause a loss of redundancy. With any one of the normally open suction header cross-connect valves closed, a failure of an automatic suction valve to open during an accident could cause two pumps to lose suction. For HPI OPERABILITY above 60% RTP, the suction header cross-connect valves must remain open. However, with THERMAL POWER ≤ 60% RTP, cross-connection is not required since the accident analysis requirements are met with one HPI pump injecting through a single train.

The safety analyses show that the HPI pump(s) will deliver sufficient water for a small break LOCA and provide sufficient boron to maintain the core subcritical.

In the small break LOCA analyses, only one HPI train is credited after actuation of the ESPS signal. For a large break LOCA, HPI is not credited at all.

The HPI trains satisfy Criterion 3 of 10 CFR 50.36 (Ref. 3).

LCO

In MODES 1 and 2, and MODE 3 with RCS temperature > 350°F, two independent HPI trains and two independent LPI-HPI flow paths are required to ensure that at least one HPI train is available, assuming a single failure in the other train. Additionally, individual components within the HPI trains may be called upon to mitigate the consequences of other transients and accidents. Each HPI train includes the piping, instruments, pumps, valves, and controls to ensure an OPERABLE flow path capable of taking suction from the BWST upon an ESPS signal. The safety grade flow indicator associated with the normal discharge valve is required to be OPERABLE to support the associated HPI train's automatic OPERABILITY. Each LPI-HPI flow path includes the piping, instruments, pumps, valves and controls to ensure the capability to manually transfer suction to the reactor building sump (LPI-HPI flow path).

During an event requiring HPI actuation, a flow path is provided to ensure an abundant supply of water from the BWST to the RCS via the HPI pumps and their respective discharge flow paths to each of the four cold leg injection nozzles and the reactor vessel. In the long term, this flow path may be manually transferred to take its supply from the reactor building sump and to supply borated water to the RCS via the LPI-HPI flow path (piggy-back mode).

BASES

LCO (continued)

The flow path for each HPI train must maintain its designed independence to ensure that no single active failure can disable both HPI trains.

The LCO is modified by a Note that requires three HPI pumps and the HPI discharge crossover valves (HP-409 and HP-410) to be OPERABLE and the suction header to be cross-connected when THERMAL POWER is > 60% RTP. The safety grade flow indicator associated with a HPI discharge crossover valve is required to be OPERABLE to support HPI discharge crossover valve OPERABILITY. The Note modifies the pump and valve OPERABILITY and valve alignment requirements to provide additional requirements assumed by the safety analyses at power levels > 60% RTP.

APPLICABILITY

In MODES 1 and 2, and MODE 3 with RCS temperature > 350°F, the HPI train OPERABILITY requirements for the small break LOCA are based on full power operation. Although reduced power would not require the same level of performance, the accident analysis does not provide for reduced cooling requirements in the lower MODES. The HPI pump performance is based on the small break LOCA, which establishes the pump performance curve. MODE 2 and MODE 3 with RCS temperature > 350°F requirements are bounded by the MODE 1 analysis.

In MODE 3 with RCS temperature ≤ 350°F and in MODE 4, the probability of an event requiring HPI actuation is significantly lessened. In this operating condition, the low probability of an event requiring HPI actuation and the availability of the LPI System provide reasonable assurance that the safety injection function is preserved.

In MODES 5 and 6, unit conditions are such that the probability of an event requiring HPI injection is extremely low. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops–MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops–MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.4, "Decay Heat Removal (DHR) and Coolant Circulation–High Water Level," and LCO 3.9.5, "Decay Heat Removal (DHR) and Coolant Circulation–Low Water Level."

ACTIONS

A.1, A.2 and A.3

With one required HPI pump inoperable, one or more HPI discharge crossover valve(s) inoperable, or the HPI suction header not cross-connected when required with THERMAL POWER > 60% RTP, the HPI pump and discharge crossover valve(s) must be restored to OPERABLE

BASES

ACTIONS

A.1, A.2 and A.3 (continued)

status and the HPI suction header must be cross-connected within 72 hours. The HPI System continues to be capable of mitigating an accident, barring a single failure. The 72 hour Completion Time is based on NRC recommendations (Ref. 4) that are based on a risk evaluation and is a reasonable time for many repairs.

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 or condition in which the LCO does not apply. To achieve this status, THERMAL POWER of the unit must be reduced to ≤ 60% RTP within 12 hours. The 12 hour Completion Time is reasonable, based on operating experience, to reach the required unit condition from full power conditions in an orderly manner and without challenging unit systems.

C.1

With one LPI-HPI flow path inoperable, the inoperable LPI-HPI flow path must be restored to OPERABLE status within 72 hours. The HPI System continues to be capable of mitigating an accident, barring a single failure. The 72 hour Completion Time is justified because there is a limited range of break sizes, and therefore a lower probability for a small break LOCA which would required piggy back operation.

D.1 and D.2

With one HPI train incapable of being automatically actuated but capable of being manually actuated with THERMAL POWER > 60% RTP, the automatic capability must be restored within 24 hours. With one HPI train inoperable with THERMAL POWER ≤ 60% RTP, the inoperable HPI train must be restored to OPERABLE status within 24 hours. The HPI System continues to be capable of mitigating an accident, barring a single failure. The 24 hour Completion Time is appropriate based on engineering judgment, taking into consideration the time required to complete the required action.

BASES

ACTIONS (continued)

E.1

If the Required Actions and the associated Completion Times of Condition C or D are 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 12 hours and the RCS temperature reduced to 350°F within 60 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.5.2.1

Verifying the correct alignment for manual and non-automatic power operated valves in the HPI flow paths provides assurance that the proper flow paths will exist for HPI operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves were verified to be in the correct position prior to locking, sealing, or securing. Similarly, this SR does not apply to automatic valves since automatic valves actuate to their required position upon an accident signal. This Surveillance does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. The 31 day Frequency is appropriate because the valves are operated under administrative control. This Frequency has been shown to be acceptable through operating experience.

SR 3.5.2.2

With the exception of the HPI pump operating to provide normal makeup, the other two HPI pumps are normally in a standby, nonoperating mode. As such, the emergency injection flow path piping has the potential to develop voids and pockets of entrained gases. Venting the HPI pump casings periodically reduces the potential that such voids and pockets of entrained gases can adversely affect operation of the HPI System. This will also minimize the potential for water hammer, pump cavitation, and pumping of noncondensible gas (e.g., air, nitrogen, or hydrogen) into the reactor vessel following an ESPS signal. This Surveillance is modified by a Note that indicates it is not applicable to operating HPI pump(s) providing normal makeup. The 31 day Frequency takes into consideration the gradual nature of gas accumulation in the HPI piping and the existence of procedural controls governing system operation.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.2.3

Periodic surveillance testing of HPI pumps to detect gross degradation caused by impeller structural damage or other hydraulic component problems is required by Section XI of the ASME Code (Ref. 5). SRs are specified in the Inservice Testing Program, which encompasses Section XI of the ASME Code.

SR 3.5.2.4 and SR 3.5.2.5

These SRs demonstrate that each automatic HPI valve actuates to the required position on an actual or simulated ESPS signal and that each HPI pump starts on receipt of an actual or simulated ESPS signal. This SR is not required for valves that are locked, sealed, or otherwise secured in position under administrative controls. The test will be considered satisfactory if control board indication verifies that all components have responded to the ESPS actuation signal properly (all appropriate ESPS actuated pump breakers have opened or closed and all ESPS actuated valves have completed their travel). The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The 18 month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment. The actuation logic is tested as part of the ESPS testing, and equipment performance is monitored as part of the Inservice Testing Program.

SR 3.5.2.6

Periodic inspections of the reactor building sump suction inlet (for LPI-HPI flow path) ensure that it is unrestricted and stays in proper operating condition. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage, on the need to preserve access to the location, and on the potential for an unplanned transient if the Surveillance were performed with the reactor at power. This Frequency has been found to be sufficient to detect abnormal degradation and has been confirmed by operating experience.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.2.7

The function of the LPI discharge valve (LP-15, LP-16) to the LPI-HPI flow path is to open and allow a cross-connection from the discharge side of an LPI pump to the suction of the HPI pumps. Manually cycling each valve open demonstrates the ability to fulfill this function. This test is performed on an 18 month Frequency. Operating experience has shown that these components usually pass the Surveillance when performed at the this Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

REFERENCES

1. 10 CFR 50.46.

2. UFSAR, Section 15.14.3.3.6.

3. 10 CFR 50.36.

4. NRC Memorandum to V. Stello, Jr., from R.L. Baer, "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975.

5. ASME, Boiler and Pressure Vessel Code, Section XI, Inservice Inspection, Article IWV-3400.

## B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

### B 3.5.3 Low Pressure Injection (LPI)

BASES

BACKGROUND

The function of the ECCS is to provide core cooling to ensure that the reactor core is protected after any of the following accidents:

a. Loss of coolant accident (LOCA);

b. Rod ejection accident (REA);

c. Steam generator tube rupture (SGTR); and

d. Main steam line break (MSLB).

There are two phases of ECCS operation: injection and recirculation. In the injection phase, all injection is initially added to the Reactor Coolant System (RCS) via the cold legs or Core Flood Tank (CFT) lines to the reactor vessel. After the borated water storage tank (BWST) has been depleted, the recirculation phase is entered as the suction is transferred to the reactor building sump.

Two redundant low pressure injection (LPI) trains are provided. The LPI trains consist of piping, valves, instruments, controls, heat exchangers, and pumps, such that water from the borated water storage tank (BWST) can be injected into the Reactor Coolant System (RCS). Safety grade flow instrumentation is required to support OPERABILITY of the LPI trains to preclude NPSH or runout problems. In MODES 1, 2 and 3, both trains of LPI must be OPERABLE. This ensures that 100% of the core cooling requirements can be provided even in the event of a single active failure. The LPI discharge header crossover valves must be manually OPERABLE in MODE 1, 2, and 3 to assure abundant, long term core cooling. Only one LPI train is required for MODE 4.

A suction header supplies water from the BWST or the reactor building sump to the LPI pumps. LPI discharges into each of the two core flood nozzles on the reactor vessel that discharge into the vessel downcomer area.

The LPI pumps are capable of discharging to the RCS at an RCS pressure of approximately 200 psia. When the BWST has been nearly emptied, the suction for the LPI pumps is manually transferred to the reactor building sump.

BASES

BACKGROUND (continued)

In the long term cooling period, flow paths in the LPI System are established to preclude the possibility of boric acid in the core region reaching an unacceptably high concentration. Two gravity flow paths are available by means of a drain line from the hot leg to the Reactor Building sump which draws coolant from the top of the core, thereby inducing core circulation. The system is designed with redundant drain lines.

During a large break LOCA, RCS pressure will rapidly decrease. The LPI System is actuated upon receipt of an ESPS signal. If offsite power is available, the safeguard loads start immediately. If offsite power is not available, the Engineered Safeguards (ES) buses are connected to the Keowee Hydro Units. The time delay (38 seconds) associated with Keowee Hydro Unit startup and pump starting determines the time required before pumped flow is available to the core following a LOCA. Full LPI flow is not available until the LPI valve strokes full open.

The LPI and HPI (LCO 3.5.2, "High Pressure Injection (HPI)"), along with the passive CFTs and the BWST covered in LCO 3.5.1, "Core Flood Tanks (CFTs)," and LCO 3.5.4, "Borated Water Storage Tank (BWST)," provide the cooling water necessary to meet 10 CFR 50.46 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The LCO helps to ensure that the following acceptance criteria for the ECCS, established by 10 CFR 50.46 (Ref. 1), will be met following a LOCA:

a. Maximum fuel element cladding temperature is ≤ 2200°F;

b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation;

c. Maximum hydrogen generation from a zirconium water reaction is ≤ 0.01 times the hypothetical amount generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react;

d. Core is maintained in a coolable geometry; and

e. Adequate long term core cooling capability is maintained.

The LCO also helps ensure that reactor building temperature limits are met.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The LPI System is assumed to provide injection in the large break LOCA analysis at full power (Ref. 2). This analysis establishes a minimum required flow for the LPI pumps, as well as the minimum required response time for their actuation.

The large break LOCA event assumes a loss of offsite power and a single failure (loss of the CT-4 transformer). For analysis purposes, the loss of offsite power assumption may be conservatively inconsistent with the assumed operation of some equipment, such as reactor coolant pumps (Ref. 3). During the blowdown stage of a LOCA, the RCS depressurizes as primary coolant is ejected through the break into the reactor building. The nuclear reaction is terminated by moderator voiding during large breaks. Following depressurization, emergency cooling water is injected into the reactor vessel core flood nozzles, then flows into the downcomer, fills the lower plenum, and refloods the core.

In the event of a Core Flood line break which results in a LOCA, with a concurrent single failure on the unaffected LPI train opposite the Core Flood break, the LPI discharge header crossover valves (LP-9 and LP-10) must be capable of being manually opened. The LPI cooler outlet throttle valves and LPI header isolation valves must be capable of being manually opened to provide assurance that flow can be established in a timely manner even if the capability to operate them from the control room is lost. These manual actions will allow cross-connection of the LPI pump discharge to the intact LPI/Core Flood tank header to provide abundant emergency core cooling.

The safety analyses show that an LPI train will deliver sufficient water to match decay heat boiloff rates for a large break LOCA.

In the large break LOCA analyses, full LPI is not credited until 53 seconds after actuation of the ESPS signal. This is based on a loss of offsite power and the associated time delays in Keowee Hydro Unit startup, valve opening and pump start. Further, LPI flow is not credited until RCS pressure drops below the pump's shutoff head. For a large break LOCA, HPI is not credited at all.

The LPI trains satisfy Criterion 3 of 10 CFR 50.36 (Ref. 4).

LCO

In MODES 1, 2, and 3, two independent (and redundant) LPI trains are required to ensure that at least one LPI train is available, assuming a single failure in the other train. Additionally, individual components within the LPI trains may be called upon to mitigate the consequences of other transients

BASES

LCO (continued)

and accidents. Each LPI train includes the piping, instruments, pumps, valves, heat exchangers and controls to ensure an OPERABLE flow path capable of taking suction from the BWST upon an ES signal and the capability to manually (remotely) transfer suction to the reactor building sump. The safety grade flow indicator associated with an LPI train is required to be OPERABLE to support LPI train OPERABILITY. The safety grade flow indicator associated with LPSW flow to an LPI cooler is required to be OPERABLE to support LPI train OPERABILITY.

In MODE 4, one of the two LPI trains is required to ensure sufficient LPI flow is available to the core.

During an event requiring LPI injection, a flow path is required to provide an abundant supply of water from the BWST to the RCS, via the LPI pumps and their respective supply headers, to the reactor vessel. In the long term, this flow path may be switched to take its supply from the reactor building sump.

This LCO is modified by three Notes. Note 1 changes the LCO requirement when in MODE 4 for the number of OPERABLE trains from two to one. Note 2 allows an LPI train to be considered OPERABLE during alignment, when aligned or when operating for decay heat removal if capable of being manually (remotely) realigned to the LPI mode of operation. This provision is necessary because of the dual requirements of the components that comprise the LPI and decay heat removal modes of the LPI System. Note 3 requires the LPI discharge header crossover valves (LP-9 and LP-10) to be OPERABLE in MODES 1, 2, and 3.

The flow path for each train must maintain its designed independence to ensure that no single failure can disable both LPI trains. If both LPI discharge header crossover valves (LP-9 and LP-10) are simultaneously open then only one LPI train is considered OPERABLE.

APPLICABILITY

In MODES 1, 2 and 3, the LPI train OPERABILITY requirements for the Design Basis Accident, a large break LOCA, are based on full power operation. The LPI discharge crossover valve OPERABILITY requirements for CFT line break is based on full power operation. Although reduced power would not require the same level of performance, the accident analysis does not provide for reduced cooling requirements in the lower MODES.

In MODE 4, one OPERABLE LPI train is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

BASES

APPLICABILITY (continued)

In MODES 5 and 6, unit conditions are such that the probability of an event requiring LPI injection is extremely low. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops–MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops–MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.4, "DHR and Coolant Circulation–High Water Level," and LCO 3.9.5, "DHR and Coolant Circulation–Low Water Level."

ACTIONS

A.1

With one LPI train inoperable in MODES 1, 2 or 3, the inoperable train must be returned to OPERABLE status within 72 hours. The 72 hour Completion Time is based on NRC recommendations (Ref. 5) that are based on a risk evaluation and is a reasonable time for many repairs. This reliability analysis has shown the risk of having one LPI train inoperable to be sufficiently low to justify continued operation for 72 hours.

B.1

With one or more LPI discharge crossover valves inoperable, the inoperable valve(s) must be returned to OPERABLE status within 72 hours. The 72 hour Completion Time is based on NRC recommendations (Ref. 5) that are based on a risk evaluation and is a reasonable time for many repairs.

C.1

If the Required Action and associated Completion Time of Condition A or B are 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 12 hours and MODE 4 within 60 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.

D.1

With one required LPI train inoperable in MODE 4, the unit is not prepared to respond to an event requiring low pressure injection and may not be prepared to continue cooldown using the LPI pumps and LPI heat exchangers. The Completion Time of immediately, which would initiate

BASES

ACTIONS

D.1 (continued)

action to restore at least one LPI train to OPERABLE status, ensures that prompt action is taken to restore the required LPI capacity. Normally, in MODE 4, reactor decay heat must be removed by a decay heat removal (DHR) loop operating with suction from the RCS. If no LPI train is OPERABLE for this function, reactor decay heat must be removed by some alternate method, such as use of the steam generator(s).

The alternate means of heat removal must continue until one of the inoperable LPI trains can be restored to operation so that continuation of decay heat removal (DHR) is provided.

With the LPI pumps (including the non ES pump) and LPI heat exchangers inoperable, it would be unwise to require the unit to go to MODE 5, where the only available heat removal system is the LPI trains operating in the DHR mode. Therefore, the appropriate action is to initiate measures to restore one LPI train and to continue the actions until the subsystem is restored to OPERABLE status.

D.2

Required Action D.2 requires that the unit be placed in MODE 5 within 24 hours. This Required Action is modified by a Note that states that the Required Action is only required to be performed if a DHR loop is OPERABLE. This Required Action provides for those circumstances where the LPI trains may be inoperable but otherwise capable of providing the necessary decay heat removal. Under this circumstance, the prudent action is to remove the unit from the Applicability of the LCO and place the unit in a stable condition in MODE 5. The Completion Time of 24 hours is reasonable, based on operating experience, to reach MODE 5 in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.5.3.1

Verifying the correct alignment for manual and non-automatic power operated valves in the LPI flow paths provides assurance that the proper flow paths will exist for LPI operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves were verified to be in the correct position prior to locking, sealing, or securing. Similarly, this SR does not apply to automatic valves since

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.3.1 (continued)

automatic valves actuate to their required position upon an accident signal. This Surveillance does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. The 31 day Frequency is appropriate because the valves are operated under administrative control, and an inoperable valve position would only affect a single train. This Frequency has been shown to be acceptable through operating experience.

When in MODE 4 an LPI train may be considered OPERABLE during alignment, when aligned or when operating for decay heat removal if capable of being manually realigned to the LPI mode of operation. Therefore, for this condition, the SR verifies that LPI is capable of being manually realigned to the LPI mode of operation.

SR 3.5.3.2

With the exception of systems in operation, the LPI pumps are normally in a standby, non-operating mode. As such, the flow path piping has the potential to develop voids and pockets of entrained gases. Venting the LPI pump casings periodically reduces the potential that such voids and pockets of entrained gases can adversely affect operation of the LPI System. This will also minimize the potential for water hammer, pump cavitation, and pumping of noncondensible gas (e.g., air, nitrogen, or hydrogen) into the reactor vessel following an ESPS signal or during shutdown cooling. This Surveillance is modified by a Note that indicates it is not applicable to operating LPI pump(s). The 31 day Frequency takes into consideration the gradual nature of gas accumulation in the LPI piping and the existence of procedural controls governing system operation.

SR 3.5.3.3

Periodic surveillance testing of LPI pumps to detect gross degradation caused by impeller structural damage or other hydraulic component problems is required by Section XI of the ASME Code (Ref. 6). SRs are specified in the Inservice Testing Program, which encompasses Section XI of the ASME Code.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.3.4 and SR 3.5.3.5

These SRs demonstrate that each automatic LPI valve actuates to the required position on an actual or simulated ESPS signal and that each LPI pump starts on receipt of an actual or simulated ESPS signal. This SR is not required for valves that are locked, sealed, or otherwise secured in position under administrative controls. The test will be considered satisfactory if control board indication verifies that all components have responded to the ESPS actuation signal properly (all appropriate ESPS actuated pump breakers have opened or closed and all ESPS actuated valves have completed their travel). The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The 18 month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment. The actuation logic is tested as part of the ESPS testing, and equipment performance is monitored as part of the Inservice Testing Program.

SR 3.5.3.6

Periodic inspections of the reactor building sump suction inlet ensure that it is unrestricted and stays in proper operating condition. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage, on the need to preserve access to the location, and on the potential for an unplanned transient if the Surveillance were performed with the reactor at power. This Frequency has been found to be sufficient to detect abnormal degradation and has been confirmed by operating experience.

SR 3.5.3.7

The function of the LPI discharge header crossover valves (LP-9, LP-10) is to open and allow a cross-connection between LPI trains. The LPI cooler outlet throttle valves (LP-12, LP-14) and LPI header isolation valves (LP-17, LP-18) must be capable of being manually opened to provide assurance that flow can be established in a timely manner even if the capability to operate them from the control room is lost. Manually cycling each valve open demonstrates the ability to fulfill this function. This test is performed on an 18 month Frequency. Operating experience has shown that these components usually pass the Surveillance when performed at the this Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

BASES (continued)

REFERENCES

1. 10 CFR 50.46.

2. UFSAR, Section 15.14.3.3.6.

3. UFSAR, Section 15.14.3.3.5.

4. 10 CFR 50.36.

5. NRC Memorandum to V. Stello, Jr., from R.L. Baer, "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975.

6. ASME, Boiler and Pressure Vessel Code, Section XI, Inservice Inspection, Article IWV-3400.

## B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

### B 3.5.4 Borated Water Storage Tank (BWST)

BASES

BACKGROUND

The BWST supports the ECCS and the Reactor Building Spray System by providing a source of borated water for ECCS and reactor building spray pump operation. In addition, the BWST supplies borated water to the refueling canal for refueling operations.

A normally open, motor operated isolation valve is provided in each LPI line to allow the operator to isolate the BWST from the LPI System after the LPI pump suction has been transferred to the reactor building sump following depletion of the BWST during a loss of coolant accident (LOCA). Use of a single BWST to supply both ECCS trains is acceptable because the BWST is a passive component, and passive failures are not assumed to occur coincidentally with a LOCA.

This LCO ensures that:

a. The BWST contains sufficient borated water to support the ECCS during the injection phase;

b. Sufficient water volume exists in the reactor building sump to support continued operation of the ECCS and reactor building spray pumps at the time of transfer to the recirculation mode of cooling; and

c. The reactor remains subcritical following a LOCA and returns subcritical following a MSLB once borated water from the ECCS reaches the core.

Insufficient water inventory in the BWST could result in insufficient cooling capacity by the ECCS when the transfer to the recirculation mode occurs.

Improper boron concentrations could result in a reduction of SDM or excessive boric acid precipitation in the core following a LOCA, as well as excessive caustic stress corrosion of mechanical components and systems inside containment.

BASES (continued)

APPLICABLE SAFETY ANALYSES

During accident conditions, the BWST provides a source of borated water to the high pressure injection (HPI), low pressure injection (LPI), and reactor building spray pumps. As such, it provides reactor building cooling and depressurization, core cooling, and replacement inventory and is a source of negative reactivity for reactor shutdown. The design basis transients and applicable safety analyses concerning each of these systems are discussed in the Applicable Safety Analyses section of Bases B 3.5.2, "High Pressure Injection (HPI)," B 3.5.3, "Low Pressure Injection (LPI)," and B 3.6.5, "Reactor Building Spray and Cooling Systems." These analyses are used to assess changes to the BWST in order to evaluate their effects in relation to the acceptance limits.

The limit on volume of ≥ 350,000 gallons (46.0 ft) is based on several factors. Sufficient deliverable volume must be available to provide at least 20 minutes of full flow of all LPI pumps prior to the transfer to the reactor building sump for recirculation. Twenty minutes gives the operator adequate time to prepare for switchover to reactor building sump recirculation.

A second factor that affects the minimum required BWST volume is the ability to support continued LPI pump operation after the manual transfer to recirculation occurs. When LPI pump suction is transferred to the sump, there must be sufficient water in the sump to ensure adequate net positive suction head (NPSH) for the LPI and reactor building spray pumps. The amount of water that enters the sump from the BWST and other sources is one of the input assumptions of the NPSH calculation. Since the BWST is the main source that contributes to the amount of water in the sump following a LOCA, the calculation does not take credit for more than the minimum volume of usable water from the BWST.

The maximum volume of water in the BWST is limited by design and ensures the solution in the sump following a LOCA is within a specified pH range that will minimize the evolution of iodine and the effect of chloride and caustic stress corrosion cracking on the mechanical systems and components.

The volume range ensures that refueling requirements are met and that the capacity of the BWST is not exceeded. Note that the volume limits refer to total, rather than usable, volume required to be in the BWST; a certain amount of water is unusable because of tank discharge line location and other physical characteristics.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The limit for minimum boron concentration of the COLR was established to ensure that, following a LOCA, with a minimum BWST level, the reactor will remain subcritical in the cold condition following mixing of the BWST and Reactor Coolant System (RCS) water volumes. Large break LOCAs assume that all CONTROL RODS remain withdrawn from the core until reflood. At this time, the analysis assumes one half of the CONTROL ROD worth is available.

The minimum and maximum concentration limits both ensure that the long term solution in the sump following a LOCA is within a specified pH range that will minimize the evolution of iodine and the effect of chloride and caustic stress corrosion cracking on the mechanical systems and components.

The maximum limit for boron concentration in the BWST of the COLR is also based on the potential for boron precipitation in the core during the long term cooling period following a LOCA. For a cold leg break, the core dissipates heat by pool boiling. Because of this boiling phenomenon in the core, the boric acid concentration will increase in this region. If allowed to proceed in this manner, a point may be reached where boron precipitation will occur in the core. Post LOCA emergency procedures direct the operator to establish dilution flow paths in the LPI System to prevent this condition by establishing a forced flow path through the core regardless of break location. These procedures are based on the minimum time in which precipitation could occur, assuming that maximum boron concentrations exist in the borated water sources used for injection following a LOCA.

Boron concentrations in the BWST in excess of the limit could result in precipitation earlier than assumed in the analysis.

The 45°F lower limit on the temperature of the solution in the BWST was established to ensure that the solution will not freeze. This temperature also helps prevent boron precipitation and ensures that water injection in the reactor vessel will not be colder than the lowest temperature assumed in reactor vessel stress analysis. The 115°F upper limit on the temperature of the BWST contents is consistent with the maximum injection water temperature assumed in the accident analysis.

The numerical values of the parameters stated in the SRs are actual values and do not include allowance for instrument errors.

The BWST satisfies Criterion 3 of 10 CFR 50.36 (Ref. 1).

BASES

LCO

The BWST exists to ensure that an adequate supply of borated water is available to cool and depressurize the reactor building in the event of an accident; to cool and cover the core in the event of a LOCA, thereby ensuring the reactor remains subcritical following an accident; and to ensure an adequate level exists in the reactor building sump to support ECCS and reactor building spray pump operation in the recirculation MODE. To be considered OPERABLE, the BWST must meet the limits for water volume, boron concentration, and temperature established in the SRs.

APPLICABILITY

In MODES 1, 2, 3, and 4, the BWST OPERABILITY requirements are dictated by the ECCS and Reactor Building Spray System OPERABILITY requirements. Since all or portions of the ECCS and Reactor Building Spray System must be OPERABLE in MODES 1, 2, 3, and 4, the BWST must be OPERABLE to support their operation.

Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops – MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops – MODE 5, Loops Not Filled," respectively. MODE 6 core cooling requirements are addressed by LCO 3.9.4, "DHR and Coolant Circulation – High Water Level," and LCO 3.9.5, "DHR and Coolant Circulation – Low Water Level."

ACTIONS

A.1

With either the BWST boron concentration or borated water temperature not within limits, the condition must be corrected within 8 hours. In this condition, the ECCS cannot perform its design functions. Therefore, prompt action must be taken to restore the tank to OPERABLE status or to place the unit in a MODE in which these systems are not required. The 8 hour limit to restore the temperature or boron concentration to within limits was developed considering the time required to change boron concentration or temperature and assuming that the contents of the tank are still available for injection.

B.1

With the BWST inoperable for reasons other than Condition A (e.g., water volume), the BWST must be restored to OPERABLE status within 1 hour. In this condition, neither the ECCS nor the Reactor Building Spray System can perform its design functions. Therefore, prompt action must be taken

BASES

ACTIONS

B.1 (continued)

to restore the BWST to OPERABLE status or to place the unit in a MODE in which the BWST is not required. The allowed Completion Time of 1 hour to restore the BWST to OPERABLE status is based on this condition simultaneously affecting multiple redundant trains.

C.1 and C.2

If the Required Action and associated Completion Time are 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 12 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.5.4.1

Verification every 24 hours that the BWST water temperature is within the specified temperature band ensures that the fluid will not freeze and that the fluid temperature entering the reactor vessel will not be colder than assumed in the reactor vessel stress analysis; and the fluid temperature entering the reactor vessel will not be hotter than assumed in the LOCA analysis. The 24 hour Frequency is sufficient to identify a temperature change that would approach either temperature limit and has been shown to be acceptable through operating experience.

The SR is modified by a Note that requires the Surveillance to be performed only when ambient air temperatures are outside the operating temperature limits of the BWST. With ambient temperature within this band, the BWST temperature should not exceed the limits.

SR 3.5.4.2

Verification every 7 days that the BWST contained volume is ≥ 350,000 gallons (46.0 ft.) ensures that a sufficient initial supply is available for injection and to support continued ECCS pump operation on recirculation. Since the BWST volume is normally stable, a 7 day Frequency has been shown to be appropriate through operating experience.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.4.3

Verification every 7 days that the boron concentration of the BWST fluid is within the required band ensures that the reactor will remain subcritical following a LOCA. Since the BWST volume is normally stable, a 7 day sampling Frequency is appropriate and has been shown to be acceptable through operating experience.

REFERENCES

1. 10 CFR 50.36.

# B 3.6 CONTAINMENT SYSTEMS

## B 3.6.1 Containment

BASES

BACKGROUND

The containment consists of the reactor building (RB) structure, its steel liner, and the penetrations of this liner and structure. The containment is designed to contain radioactive material that may be released from the reactor core following an accident. Additionally, the containment provides shielding from the fission products that may be present in the containment atmosphere following accident conditions.

The containment is a reinforced concrete structure with a cylindrical wall, a flat foundation mat, and a shallow dome roof. The containment design includes ungrouted tendons where the cylinder wall is prestressed with a post tensioning system in the vertical and horizontal directions, and the dome roof is prestressed using a three way post tensioning system. The inside surface of the containment is lined with a carbon steel liner to ensure a high degree of leak tightness during operating and accident conditions.

The reinforced concrete structure is required for structural integrity of the containment under accident conditions. The steel liner and its penetrations establish the leakage limiting boundary of the containment. Maintaining the containment OPERABLE limits the leakage of fission product radioactivity from the containment to the environment. SR 3.6.1.1 and SR 3.6.1.2 leakage rate requirements comply with 10 CFR 50, Appendix J, Option A and B as applicable (Ref. 1), as modified by approved exemptions.

The isolation devices for the penetrations in the containment boundary are a part of the containment leak tight barrier. To maintain this leak tight barrier:

a. All penetrations required to be closed during accident conditions are either:

   1. capable of being closed by an OPERABLE automatic containment isolation system, or

   2. closed by manual valves, blind flanges, or de-activated automatic valves in their closed positions, except as provided in LCO 3.6.3, "Containment Isolation Valves";

BASES

BACKGROUND (continued)

b. Each air lock is OPERABLE, except as provided in LCO 3.6.2, "Containment Air Locks"; and

c. The equipment hatch is closed.

APPLICABLE SAFETY ANALYSES

The safety design basis for the containment is that the containment must withstand the pressures and temperatures of the limiting accident without exceeding the design leakage rate.

The accidents that result in a challenge to containment from high pressures and temperatures are a loss of coolant accident (LOCA) and a steam line break (Ref. 2). In addition, release of significant fission product radioactivity within containment can occur from a LOCA. In the accident analyses, it is assumed that the containment is OPERABLE such that, for the accidents involving release of fission product radioactivity, release to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of 0.25% of containment air weight per day (Ref. 3). This leakage rate, used in the evaluation of offsite doses resulting from accidents, is defined in 10 CFR 50, Appendix J, Option A and B (Ref. 1), as $L_a$: the maximum allowable leakage rate at the calculated maximum peak containment pressure ($P_a$) resulting from the limiting accident. The allowable leakage rate represented by $L_a$ forms the basis for the acceptance criteria imposed on all containment leakage rate testing. $L_a$ is assumed to be 0.25% per day in the safety analysis at $P_a$ = 59.0 psig (Ref. 3).

The containment satisfies Criterion 3 of the 10 CFR 50.36 (Ref. 4).

LCO

Containment OPERABILITY is maintained by limiting leakage to $\leq 1.0\ L_a$, except prior to the first startup after performing a required 10 CFR 50, Appendix J, leakage test. At this time, the applicable leakage limits must be met. Compliance with this LCO will ensure a containment configuration, including equipment hatches, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analysis.

APPLICABILITY

In MODES 1, 2, 3, and 4, an accident could cause a release of radioactive material into containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, containment is not required to be OPERABLE in MODE 5 to prevent leakage of radioactive material from containment. The requirements for containment during MODE 6 are addressed in LCO 3.9.3, "Containment Penetrations."

BASES (continued)

ACTIONS

A.1

In the event containment is inoperable, containment must be restored to OPERABLE status within 1 hour. The 1 hour Completion Time provides a period of time to correct the problem commensurate with the importance of maintaining containment during MODES 1, 2, 3, and 4. This time period also ensures the probability of an accident (requiring containment OPERABILITY) occurring during periods when containment is inoperable is minimal.

B.1 and B.2

If the Required Action and associated Completion Time 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 12 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1

Maintaining the containment OPERABLE requires compliance with the visual examinations and Type A leakage rate test requirements of the Containment Leakage Rate Testing Program. As left leakage prior to the first startup after performing a required leakage test is required to be $< 0.75\ L_a$ for overall Type A leakage following an outage or shutdown that included Type A testing. At all other times between required leakage rate tests, the acceptance criteria is based on an overall Type A leakage limit of $\leq 1.0\ L_a$. At $\leq 1.0\ L_a$ the offsite dose consequences are bounded by the assumptions of the safety analysis. SR Frequencies are as required by the Containment Leakage Rate Testing Program. These periodic testing requirements verify that the containment leakage rate does not exceed the leakage rate assumed in the safety analysis.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.2

Maintaining the containment OPERABLE requires compliance with the Type B and C leakage rate test requirements of 10 CFR 50, Appendix J, Option A (Ref. 1), as modified by approved exemptions. As left leakage prior to the first startup after performing a required 10 CFR 50, Appendix J, Option A, leakage test is required to be < 0.6 $L_a$ for combined Type B and C leakage. At all other times between required leakage rate tests, the acceptance criteria is based on an overall Type A leakage limit of ≤ 1.0 $L_a$. At ≤ 1.0 $L_a$ the offsite dose consequences are bounded by the assumptions of the safety analysis. SR Frequencies are as required by Appendix J, Option A, as modified by approved exemptions. Thus, SR 3.0.2 (which allows Frequency extensions) does not apply. These periodic testing requirements verify that the containment leakage rate does not exceed the leakage rate assumed in the safety analysis.

SR 3.6.1.3

This SR ensures that the structural integrity of the containment will be maintained in accordance with the provisions of the Containment Tendon Surveillance Program. Testing and Frequency are as described in Specification 5.5.7, "Pre-stressed Concrete Containment Tendon Surveillance Program."

REFERENCES

1. 10 CFR 50, Appendix J, Option A and B.
2. UFSAR, Sections 15.13 and 15.14.
3. UFSAR, Section 6.2.
4. 10 CFR 50.36.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.2 Containment Air Locks

BASES

BACKGROUND

Containment air locks, also known as the personnel hatch and the emergency hatch, form part of the containment pressure boundary and provide a means for personnel access during all MODES of operation.

Each air lock is nominally a right circular cylinder with a door at each end. The doors are interlocked to prevent simultaneous opening. During periods when containment is not required to be OPERABLE, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary. Each air lock door has been designed and is tested to certify its ability to withstand a pressure in excess of the maximum expected pressure following an accident in containment. As such, closure of a single door supports containment OPERABILITY. Each of the outer doors contains double gasketed seals and local leakage rate testing capability to ensure pressure integrity. To effect a leak tight seal, the air lock design uses pressure seated doors (i.e., an increase in containment internal pressure results in increased sealing force on each door). Each personnel air lock door is provided with limit switches that provide control room indication of door position.

The containment air locks form part of the containment pressure boundary. As such, air lock integrity and leak tightness are essential for maintaining the containment leakage rate within limit in the event of an accident. Not maintaining air lock integrity or leak tightness may result in a leakage rate in excess of that assumed in the unit safety analysis.

APPLICABLE SAFETY ANALYSES

The accident that results in a release of radioactive material within containment is a loss of coolant accident (LOCA) (Ref. 2). In the analysis of this accident, it is assumed that containment is OPERABLE such that release of fission products to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of 0.25% of containment air weight per day (Ref. 3). This leakage rate is defined in 10 CFR 50, Appendix J, Option A and B (Ref. 1), as $L_a$: the maximum allowable containment leakage rate at the calculated

BASES

APPLICABLE SAFETY ANALYSES (continued)

maximum peak containment pressure ($P_a$) following an accident. This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air lock.

The containment air locks satisfy Criterion 3 of 10 CFR 50.36 (Ref. 4).

LCO

Each containment air lock forms part of the containment pressure boundary. As a part of the containment pressure boundary, the air lock safety function is related to control of the containment leakage rate resulting from an accident. Thus, each air lock's structural integrity and leak tightness are essential to the successful mitigation of such an event.

Each air lock is required to be OPERABLE. For the air lock to be considered OPERABLE, the air lock interlock mechanism must be OPERABLE, the air lock must be in compliance with the Type B air lock leakage test, and both air lock doors must be OPERABLE. The interlock allows only one air lock door of an air lock to be opened at one time. This provision ensures that a gross breach of containment does not exist when containment is required to be OPERABLE. Closure of a single door in each air lock is sufficient to provide a leak tight barrier following postulated events. Nevertheless, both doors are normally closed when the air lock is not being used for normal entry into or exit from containment.

APPLICABILITY

In MODES 1, 2, 3, and 4, an accident could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the containment air locks are not required in MODE 5 to prevent leakage of radioactive material from containment. The requirements for the containment air locks during MODE 6 are addressed in LCO 3.9.3, "Containment Penetrations."

ACTIONS

The ACTIONS are modified by a Note that allows entry and exit to perform repairs on the affected air lock component. If the outer door is inoperable, then it may be easily accessed for most repairs. An inoperable inner door can be accessed from inside containment by entering through the other OPERABLE air lock. However, if this is not practicable, or if repairs on either door must be performed from the barrel side of the door then it is permissible to enter the air lock through the OPERABLE door, which means there is a short time during which the containment boundary is not intact (during access through the OPERABLE door). The ability to open the OPERABLE door, even if it means the containment boundary is

BASES

ACTIONS (continued)

temporarily not intact, is acceptable due to the low probability of an event that could pressurize the containment during the short time in which the OPERABLE door is expected to be open. After each entry and exit the OPERABLE door must be immediately closed. If conditions permit, entry and exit should be via an OPERABLE air lock.

A second Note has been added to provide clarification that, for this LCO, separate Condition entry is allowed for each air lock. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable air lock. Complying with the Required Actions may allow for continued operation, and a subsequent inoperable air lock is governed by subsequent Condition entry and application of associated Required Actions.

In the event the air lock leakage results in exceeding the overall (combined) containment leakage rate, Note 3 directs entry into the applicable Conditions and Required Actions of LCO 3.6.1, "Containment."

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

With one air lock door inoperable in one or more containment air locks, the OPERABLE door must be verified closed (Required Action A.1) in each affected containment air lock.

This ensures that a leak tight containment barrier is maintained by the use of an OPERABLE air lock door. This action must be completed within 1 hour. This specified time period is consistent with the ACTIONS of LCO 3.6.1, which requires containment be restored to OPERABLE status within 1 hour.

In addition, the affected air lock penetration must be isolated by locking closed the remaining OPERABLE air lock door within the 24 hour Completion Time. The 24 hour Completion Time is considered reasonable for locking the OPERABLE air lock door, considering the OPERABLE door of the affected air lock is being maintained closed.

Required Action A.3 verifies that an air lock with an inoperable door has been isolated by the use of a locked and closed OPERABLE air lock door. This ensures that an acceptable containment leakage boundary is maintained. The Completion Time of once per 31 days is based on engineering judgment and is considered adequate in view of the low likelihood of a locked door being mispositioned and other administrative

BASES

ACTIONS

A.1, A.2, and A.3 (continued)

controls. Required Action A.3 is modified by a Note that applies to air lock doors located in high radiation areas and allows these doors to be verified locked closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of the door, once it has been verified to be in the proper position, is small.

The Required Actions have been modified by two Notes. Note 1 clarifies that only the Required Actions and associated Completion Times of Condition C are required if both doors in the same air lock are inoperable. With both doors in the same air lock inoperable, an OPERABLE door is not available to be closed. Required Actions C.1 and C.2 are the appropriate remedial actions. The exception of Note 1 does not affect tracking the Completion Time from the initial entry into Condition A; only the requirement to comply with the Required Actions. Note 2 allows use of the air lock for entry and exit for 7 days under administrative controls if both air locks are inoperable. This 7 day restriction begins when the second air lock is discovered inoperable. Containment entry may be required to perform Technical Specifications (TS) Surveillances and Required Actions, as well as other activities on equipment inside containment that are required by TS or activities on equipment that support TS-required equipment. This Note is not intended to preclude performing other activities (i.e., non-TS-required activities) if the containment was entered, using the inoperable air lock, to perform an allowed activity listed above. This allowance is acceptable due to the low probability of an event that could pressurize the containment during the short time that the OPERABLE door is expected to be open.

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

With an air lock interlock mechanism inoperable in one or more air locks, the Required Actions and associated Completion Times are consistent with those specified in Condition A.

The Required Actions have been modified by two Notes. Note 1 clarifies that only the Required Actions and associated Completion Times of Condition C are required if both doors in the same air lock are inoperable. With both doors in the same air lock inoperable, an OPERABLE door is not available to be closed. Required Actions C.1 and C.2 are the appropriate remedial actions. Note 2 allows entry into and exit from the containment

BASES

ACTIONS

B.1, B.2, and B.3 (continued)

under the control of a dedicated individual stationed at the air lock to ensure that only one door is opened at a time (i.e., the individual performs the function of the interlock).

Required Action B.3 is modified by a Note that applies to air lock doors located in high radiation areas and allows these doors to be verified locked closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of the door, once it has been verified to be in the proper position, is small.

C.1, C.2, and C.3

With one or more air locks inoperable for reasons other than those described in Condition A or B, Required Action C.1 requires action to be immediately initiated to evaluate previous combined leakage rates using current air lock test results. An evaluation is acceptable since it is overly conservative to immediately declare the containment inoperable if the overall air lock leakage is not within limits. In many instances, containment remains OPERABLE, yet only 1 hour (per LCO 3.6.1) would be provided to restore the air lock door to OPERABLE status prior to requiring a unit shutdown.

Required Action C.2 requires that one door in each affected containment air lock must be verified to be closed. This action must be completed within the 1 hour Completion Time. This specified time period is consistent with the ACTIONS of LCO 3.6.1, which requires that containment be restored to OPERABLE status within 1 hour.

Additionally, the affected air lock(s) must be restored to OPERABLE status within the 24 hour Completion Time. The specified time period is considered reasonable for restoring an inoperable air lock to OPERABLE status assuming that at least one door is maintained closed in each affected air lock.

D.1 and D.2

If the Required Actions and associated Completion Times are 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

BASES

ACTIONS

D.1 and D.2 (continued)

12 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.2.1

Maintaining containment air locks OPERABLE requires compliance with the leakage rate test requirements of 10 CFR 50, Appendix J, Option A (Ref. 1), as modified by approved exemptions. This SR reflects the leakage rate testing requirements with regard to air lock leakage (Type B leakage tests). The periodic testing requirements verify that the air lock leakage does not exceed the allowed fraction of the overall containment leakage rate. The Frequency is required by 10 CFR 50, Appendix J, Option A, as modified by approved exemptions. Thus, SR 3.0.2 (which allows Frequency extensions) does not apply. Either a full air lock leak test or a leak test of the outer air lock door seal performed within 3 days of initial opening, and during periods of frequent use, at least once every 3 days, is an acceptable method of complying with 10 CFR 50, Appendix J requirements (References 5 and 6).

The SR has been modified by two Notes. Note 1 states that an inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. This is considered reasonable, since either air lock door is capable of providing a fission product barrier in the event of an accident. Note 2 has been added to this SR requiring the results to be evaluated against the acceptance criteria of SR 3.6.1.2. This ensures that air lock leakage is properly accounted for in determining the combined Type B and C containment leakage rate.

SR 3.6.2.2

The air lock interlock is designed to prevent simultaneous opening of both doors in a single air lock. Since both the inner and outer doors of an air lock are designed to withstand the maximum expected post accident containment pressure, closure of either door will support containment OPERABILITY. Thus, the door interlock feature supports containment OPERABILITY while the air lock is being used for personnel transit in and out of the containment. Periodic testing of this interlock demonstrates that

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.2.2 (continued)

the interlock will function as designed and that simultaneous opening of the inner and outer doors will not inadvertently occur. Due to the purely mechanical nature of this interlock, and given that the interlock mechanism is not normally challenged when the containment air lock door is used for entry or exit (procedures require strict adherence to single door opening), this test is only required to be performed every 18 months. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage, and the potential loss of containment OPERABILITY if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 18 month Frequency. The 18 month Frequency is based on engineering judgment and is considered adequate given that the interlock is not challenged during use of the air lock.

REFERENCES

1. 10 CFR 50, Appendix J, Option A and B.

2. UFSAR, Section 15.14.

3. UFSAR, Section 6.2.

4. 10 CFR 50.36.

5. Duke Power Company letter from William O. Parker, Jr. to Harold R. Denton (NRC) dated July 24, 1981.

6. NRC Letter from Philip C. Wagner to William O. Parker, Jr., dated November 6, 1981, Issuance of Amendment 104, 104 and 101 to Licenses DPR-38, DPR-47 and DPR-55 for the Oconee Nuclear Station Units Nos 1, 2 and 3.

## B 3.6 CONTAINMENT SYSTEMS

### B 3.6.3 Containment Isolation Valves

BASES

BACKGROUND

The containment isolation valves form part of the containment pressure boundary and provide a means for fluid penetrations not serving accident consequence limiting systems to be provided with two isolation barriers that are closed on an automatic isolation signal. These isolation devices consist of either passive devices or active (automatic) devices. Manual valves, non-automatic power operated valves in their closed position, de-activated automatic valves secured in their closed position (including check valves with flow through the valve secured), blind flanges, and closed systems are considered passive devices. Check valves, or other automatic valves designed to close following an accident without operator action, are considered active devices. Two barriers in series are provided for each penetration so that no single credible failure or malfunction of an active component can result in a loss of isolation or leakage that exceeds limits assumed in the safety analyses. One of these barriers may be a closed system. These barriers (typically containment isolation valves) make up the Containment Isolation System.

Containment isolation occurs upon receipt of a high containment pressure or diverse containment isolation signal. The containment isolation signal closes automatic containment isolation valves in fluid penetrations not required for operation of engineered safeguard systems to prevent leakage of radioactive material. Upon actuation, automatic containment valves also isolate systems not required for containment or Reactor Coolant System (RCS) heat removal. Other penetrations are isolated by the use of valves in the closed position or blind flanges. As a result, the containment isolation valves (and blind flanges) help ensure that the containment atmosphere will be isolated in the event of a release of radioactive material to containment atmosphere from the RCS following an accident.

OPERABILITY of the containment isolation valves (and blind flanges) supports containment OPERABILITY during accident conditions.

The OPERABILITY requirements for containment isolation valves help ensure that containment is isolated within the time limits assumed in the safety analysis. Therefore, the OPERABILITY requirements provide assurance that the containment function assumed in the safety analysis will be maintained.

BASES

BACKGROUND (continued)

The Reactor Building Purge System is part of the Reactor Building Ventilation System. The Purge System was designed for intermittent operation, providing a means of removing airborne radioactivity caused by minor leakage from the RCS prior to personnel entry into containment. The Reactor Building Purge System consists of one 48 inch line for exhaust and one 48 inch line for supply, with exhaust fans capable of purging the containment atmosphere at a rate of approximately 35,000 $ft^3$/min. The reactor building purge supply and exhaust lines each contain two isolation valves that receive a reactor building isolation signal.

Failure of the purge valves to close following a design basis event would cause a significant increase in the radioactive release because of the large containment leakage path introduced by these 48 inch purge lines. Failure of the purge valves to close would result in leakage considerably in excess of the containment design leakage rate of 0.25% of containment air weight per day ($L_a$) (Ref. 1). Because of their large size, the 48 inch purge valves are not qualified for automatic closure from their open position under accident conditions. Therefore, the 48 inch purge valves are maintained sealed closed (SR 3.6.3.1) in MODES 1, 2, 3, and 4 to ensure the containment boundary is maintained.

APPLICABLE SAFETY ANALYSES

The containment isolation valve LCO was derived from the assumptions related to minimizing the loss of reactor coolant inventory and establishing containment boundary during major accidents. As part of the containment boundary, containment isolation valve OPERABILITY supports leak tightness of the containment. Therefore, the safety analysis of any event requiring isolation of containment is applicable to this LCO.

The accident that results in a significant release of radioactive material within containment is a loss of coolant accident (LOCA)(Ref. 2). In the analysis for this accident, it is assumed that containment isolation valves are either closed or function to close within the required isolation time following event initiation. This ensures that potential paths to the environment through containment isolation valves (including reactor building purge valves) are minimized. The safety analysis assumes that the 48 inch purge valves are closed at event initiation.

The LOCA analysis assumes a fixed amount of core inventory escapes. No mechanistic scenario is evaluated to determine what portion of the inventory is released prior to closure of the containment isolation valves. Industry standards for sizing valve operators govern the closure times of the containment isolation valves.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The purge valves may be unable to close in the environment following a LOCA. Therefore, each of the purge valves is required to remain sealed closed during MODES 1, 2, 3, and 4.

The containment isolation valves satisfy Criterion 3 of 10 CFR 50.36 (Ref. 3).

LCO

Containment isolation valves form a part of the containment boundary. The containment isolation valve safety function is related to minimizing the loss of reactor coolant inventory and establishing the containment boundary during an accident.

The automatic power operated isolation valves are required to have isolation times within limits and to actuate on an automatic isolation signal. The 48 inch purge valves must be maintained sealed closed. The valves covered by this LCO are listed in the UFSAR (Ref. 4).

The normally closed isolation valves are considered OPERABLE when non-automatic power operated valves are closed, manual valves are closed, check valves have flow through the valve secured, blind flanges are in place, and closed systems are intact.

The containment isolation valve leakage rates are addressed by LCO 3.6.1, "Containment," as Type C testing.

This LCO provides assurance that the containment isolation valves and purge valves will perform their designated safety functions to minimize the loss of reactor coolant inventory and establish the containment boundary during accidents.

APPLICABILITY

In MODES 1, 2, 3, and 4, an accident could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the containment isolation valves are not required to be OPERABLE in MODE 5. The requirements for containment isolation valves during MODE 6 are addressed in LCO 3.9.3, "Containment Penetrations."

BASES (continued)

ACTIONS

The ACTIONS are modified by a Note allowing penetration flow paths, except for 48 inch purge valve penetration flow paths, to be unisolated intermittently under administrative controls. These administrative controls consist of stationing a dedicated individual, who is in continuous communication with the control room. In this way, the penetration can be rapidly isolated when a need for containment isolation is indicated. Due to the size of the reactor building purge line penetration and the fact that those penetrations exhaust directly from the containment atmosphere to the environment, the penetration flow paths containing these valves may not be opened under administrative controls.

A second Note has been added to provide clarification that, for this LCO, separate Condition entry is allowed for each penetration flow path. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable containment isolation valve. Complying with the Required Actions may allow for continued operation, and subsequent inoperable containment isolation valves are governed by subsequent Condition entry and application of associated Required Actions.

The ACTIONS are further modified by a third Note, which ensures appropriate remedial actions are taken, if necessary, if the affected systems are rendered inoperable by an inoperable containment isolation valve.

A.1 and A.2

In the event one containment isolation valve in one or more penetration flow paths is inoperable, the affected penetration flow path must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic containment isolation valve, a closed and de-activated non-automatic power operated valve, a closed manual valve, a blind flange, and a check valve with flow through the valve secured. For a penetration isolated in accordance with Required Action A.1, the device used to isolate the penetration should be the closest available one to containment. Required Action A.1 must be completed within the 4 hour Completion Time. The specified time period is reasonable, considering the time required to isolate the penetration and the relative importance of supporting containment OPERABILITY during MODES 1, 2, 3, and 4.

BASES

ACTIONS

A.1 and A.2 (continued)

For affected penetration flow paths that cannot be restored to OPERABLE status within the 4 hour Completion Time and that have been isolated in accordance with Required Action A.1, the affected penetration flow paths must be verified to be isolated on a periodic basis. This periodic verification is necessary to ensure that containment penetrations required to be isolated following an accident and no longer capable of being automatically isolated will be in the isolation position should an event occur. This Required Action does not require any testing or device manipulation. Rather, it involves verification, through a system walkdown, that those isolation devices outside containment and capable of being mispositioned are in the correct position. The Completion Time of "once per 31 days for isolation devices outside containment" is appropriate considering the fact that the devices are operated under administrative controls and the probability of their misalignment is low. For the isolation devices inside containment, the time period specified as "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is based on engineering judgment and is considered reasonable in view of the inaccessibility of the isolation devices and other administrative controls that will ensure that isolation device misalignment is an unlikely possibility.

Condition A has been modified by a Note indicating this Condition is only applicable to those penetration flow paths with two containment isolation valves. For penetration flow paths with only one containment isolation valve in closed systems, Condition C provides appropriate actions.

Required Action A.2 is modified by a Note that applies to isolation devices located in high radiation areas and allows the devices to be verified by use of administrative means. Allowing verification by administrative means is considered acceptable since access to these areas is typically restricted during MODES 1, 2, 3, and 4 for ALARA reasons. Therefore, the probability of misalignment of these devices, once they have been verified to be in the proper position, is small.

B 1

With two containment isolation valves in one or more penetration flow paths inoperable, the affected penetration flow path must be isolated within 1 hour. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated

BASES

ACTIONS

B.1 (continued)

automatic valve, a closed and de-activated non-automatic power operated valve, a closed manual valve, and a blind flange. A check valve may not be used to isolate the penetration. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1. In the event the affected penetration is isolated in accordance with Required Action B.1, the affected penetration must be verified to be isolated on a periodic basis per Required Action A.2, which remains in effect. This periodic verification is necessary to assure leak tightness of containment and that penetrations requiring isolation following an accident are isolated. The Completion Time of once per 31 days for verifying each affected penetration flow path is isolated is appropriate considering the fact that the valves are operated under administrative controls and the probability of their misalignment is low.

Condition B is modified by a Note indicating this Condition is only applicable to penetration flow paths with two containment isolation valves. Condition A of this LCO addresses the condition of one containment isolation valve inoperable in this type of penetration flow path.

C.1 and C.2

With one or more penetration flow paths with one containment isolation valve inoperable, the inoperable valve must be restored to OPERABLE status or the affected penetration flow path must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, a closed and de-activated non-automatic power operated valve, a closed manual valve, and a blind flange. A check valve may not be used to isolate the affected penetration. Required Action C.1 must be completed within the 4 hour Completion Time. The specified time period is reasonable, considering the relative stability of the closed system (hence, reliability) to act as a penetration isolation boundary and the relative importance of supporting containment OPERABILITY during MODES 1, 2, 3, and 4. In the event the affected penetration is isolated in accordance with Required Action C.1, the affected penetration flow path must be verified to be isolated on a periodic basis. This periodic verification is necessary to assure leak tightness of containment and that containment penetrations requiring isolation following an accident are isolated. The Completion Time of once per 31 days for verifying that each affected penetration flow path is isolated is appropriate considering the fact that the valves are operated under administrative controls and the probability of their misalignment is low.

BASES

ACTIONS

C.1 and C.2 (continued)

Condition C is modified by a Note indicating that this Condition is only applicable to those penetration flow paths with only one containment isolation valve in closed systems. This Note is necessary since this Condition is written to specifically address those penetration flow paths in a closed system.

Required Action C.2 is modified by a Note that applies to valves and blind flanges located in high radiation areas and allows these devices to be verified by use of administrative means. Allowing verification by administrative means is considered acceptable since access to these areas is typically restricted during MODES 1, 2, 3, and 4 for ALARA reasons. Therefore, the probability of misalignment of these devices, once verified to be in the proper position, is small.

D.1 and D.2

If the Required Actions and associated Completion Times are 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 12 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.3.1

Each 48 inch reactor building purge valve is required to be verified sealed closed at 31 day intervals. This Surveillance is designed to ensure that a gross breach of containment is not caused by an inadvertent or spurious opening of a reactor building purge valve. Detailed analysis of the purge valves failed to conclusively demonstrate their ability to close during a LOCA in time to limit offsite doses. Therefore, these valves are required to be in the sealed closed position during MODES 1, 2, 3, and 4. A reactor building purge valve that is sealed closed must have motive power to the valve operator removed. This can be accomplished by de-energizing the source of electric power or by removing the air supply to the valve operator.

In this application, the term "sealed" has no connotation of leak tightness. The 31 day Frequency is consistent with other containment isolation valve requirements discussed in SR 3.6.3.2.

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.3.2

This SR requires verification that each containment isolation manual and non-automatic power operated valve and blind flange located outside containment and not locked, sealed, or otherwise secured, and required to be closed during accident conditions is closed. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside the containment boundary is within design limits. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those containment isolation valves outside containment and capable of being mispositioned are in the correct position. Since verification of valve position for containment isolation valves outside containment is relatively easy, the 31 day Frequency is based on engineering judgment and was chosen to provide added assurance of the correct positions. The SR specifies that containment isolation valves open under administrative controls are not required to meet the SR during the time the valves are open. These administrative controls consist of stationing a dedicated individual at the valve, who is in continuous communication with the control room. The dedicated individual can be responsible for closing more than one valve provided that the valves are all in close vicinity and can be closed in a timely manner. This SR does not apply to valves that are locked, sealed, or otherwise secured, since these were verified to be in the correct position upon locking, sealing, or securing.

The Note applies to valves and blind flanges located in high radiation areas and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted during MODES 1, 2, 3, and 4 for ALARA reasons. Therefore, the probability of misalignment of these containment isolation valves, once they have been verified to be in the proper position, is low.

SR 3.6.3.3

This SR requires verification that each containment isolation manual and non-automatic power operated valve and blind flange that is located inside containment and not locked, sealed, or otherwise secured, and required to be closed during accident conditions is closed. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside the containment boundary is within design limits. For containment isolation valves inside containment, the Frequency of "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is appropriate, since these containment isolation valves are operated under administrative controls and the probability of their misalignment is low. The SR specifies that containment isolation valves open under administrative controls are not



(Bases Change 00 2-29-2000)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.3.3 (continued)

required to meet the SR during the time they are open. These administrative controls consist of stationing a dedicated individual at the valve, who is in continuous communication with the control room. The dedicated individual can be responsible for closing more than one valve provided that the valves are all in close vicinity and can be closed in a timely manner. This SR does not apply to valves that are locked, sealed, or otherwise secured, since these were verified to be in the correct position upon locking, sealing, or securing.

The Note allows valves and blind flanges located in high radiation areas to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since the access to these areas is typically restricted during MODES 1, 2, 3, and 4 for ALARA reasons. Therefore, the probability of misalignment of these containment isolation valves, once they have been verified to be in their proper position, is small.

SR 3.6.3.4

Verifying that the isolation time of each automatic power operated containment isolation valve is within limits is required to demonstrate OPERABILITY. The isolation time and Frequency of this SR are in accordance with the Inservice Testing Program.

SR 3.6.3.5

Automatic containment isolation valves close on a containment isolation signal to prevent leakage of radioactive material from containment following an accident. This SR ensures that each automatic containment isolation valve will actuate to its isolation position on a containment isolation signal. This SR is not required for valves that are locked, sealed, or otherwise secured in position under administrative controls. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass this Surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

BASES (continued)

REFERENCES

1. UFSAR, Section 6.2.
2. UFSAR, Section 15.14.
3. 10 CFR 50.36.
4. UFSAR, Table 6-7.

## B 3.6 CONTAINMENT SYSTEMS

### B 3.6.4 Containment Pressure

BASES

BACKGROUND

The containment pressure is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a loss of coolant accident (LOCA) or steam line break (SLB). These limits also prevent the containment pressure from exceeding the containment design negative pressure differential with respect to the outside atmosphere in the event of inadvertent actuation of the Containment Spray System.

Containment pressure is a process variable that is monitored and controlled. The containment pressure limits are derived from the input conditions used in the containment functional analyses and the containment structure external pressure analysis. Should operation occur above the containment pressure limit coincident with an accident, post accident containment pressures could exceed calculated values.

APPLICABLE SAFETY ANALYSES

Containment internal pressure is an initial condition used in the accident analyses to establish the maximum peak containment internal pressure. The limiting accidents considered, relative to containment pressure, are the LOCA and SLB, which are analyzed using computer transient simulations. The worst-case LOCA generates larger mass and energy release than the worst-case SLB. Thus, the LOCA event bounds the SLB event from the containment peak pressure standpoint (Ref. 1).

The initial pressure condition used in the containment analysis was 16.2 psia (1.5 psig). This resulted in a maximum peak pressure from a LOCA of 58.9 psig. The LCO limit of 1.2 psig ensures that, in the event of an accident, the design pressure of 59 psig for containment is not exceeded. In addition, the building was designed for an internal pressure equal to 3.0 psig above external pressure which corresponds to a margin of 0.5 psi above the differential pressure that could be developed if the building is sealed with an internal temperature of 120°F with a barometric pressure of 29.0 inches of Hg and the building is subsequently cooled to an internal temperature of 80°F with a concurrent rise in barometric pressure to 31.0 inches of Hg. The LCO limit of -2.45 psig ensures that operation within the design limit of -3.0 psig is maintained (Ref. 2).

BASES

APPLICABLE SAFETY ANALYSES (continued)

For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the cooling effectiveness of the Emergency Core Cooling Systems during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. Therefore, for the reflood phase, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the containment pressure response in accordance with 10 CFR 50, Appendix K (Ref. 3).

Containment pressure satisfies Criterion 2 of 10 CFR 50.36 (Ref. 4).

LCO

Maintaining containment pressure less than or equal to the LCO upper pressure limit ensures that, in the event of an accident, the resultant peak containment accident pressure will remain below the containment design pressure. Maintaining containment pressure greater than or equal to the LCO lower pressure limit ensures that the containment will not exceed the design negative differential pressure.

APPLICABILITY

In MODES 1, 2, 3, and 4, an accident could cause a release of radioactive material to containment. Since maintaining containment pressure within design basis limits is essential to ensure initial conditions assumed in the accident analysis are maintained, the LCO is applicable in MODES 1, 2, 3, and 4.

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, maintaining containment pressure within the limits of the LCO is not required in MODES 5 and 6.

ACTIONS

A.1

When containment pressure is not within the limits of the LCO, containment pressure must be restored to within these limits within 1 hour. The Required Action is necessary to return operation to within the bounds of the containment analysis. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1, "Containment," which requires that containment be restored to OPERABLE status within 1 hour.

BASES

ACTIONS (continued)

B.1 and B.2

If the Required Action and associated Completion Time 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 12 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.4.1

Verifying that containment pressure is within limits ensures that operation remains within the limits assumed in the containment analysis. The 12 hour Frequency of this SR was developed after taking into consideration operating experience related to trending of containment pressure variations during the applicable MODES. Furthermore, the 12 hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to an abnormal containment pressure condition.

REFERENCES

1. UFSAR, Chapter 15.
2. UFSAR, Section 6.2.
3. 10 CFR 50, Appendix K.
4. 10 CFR 50.36.

# B 3.6 CONTAINMENT SYSTEMS

## B 3.6.5 Reactor Building Spray and Cooling Systems

BASES

BACKGROUND

The Reactor Building Spray and Reactor Building Cooling systems provide containment atmosphere cooling to limit post accident pressure and temperature in containment to less than the design values. Reduction of containment pressure and the iodine removal capability of the spray reduces the release of fission product radioactivity from containment to the environment, in the event of an accident, to within limits. The Reactor Building Spray and Reactor Building Cooling systems are designed to meet ONS Design Criteria (Ref. 1).

The Reactor Building Cooling System and Reactor Building Spray System are Engineered Safeguards (ES) systems. They are designed to ensure that the heat removal capability required during the post accident period can be attained. The Reactor Building Spray System and Reactor Building Cooling System provide containment heat removal operation. The Reactor Building Spray System and Reactor Building Cooling System provide methods to limit and maintain post accident conditions to less than the containment design values.

Reactor Building Spray System

The Reactor Building Spray System consists of two separate trains of equal capacity, each capable of meeting the design basis. Each train includes a reactor building spray pump, spray headers, nozzles, valves, piping and a flow indicator. The safety grade flow indicator of an RBS train and the safety grade flow indicator of the associated LPI train are both required to be OPERABLE to support RBS train OPERABILITY. Both LPI and RBS train flow must be monitored to ensure that the RBS pump does not runout or exceed NPSH requirements. Each train is powered from a separate ES bus. The borated water storage tank (BWST) supplies borated water to the Reactor Building Spray System during the injection phase of operation. In the recirculation mode of operation, Reactor Building Spray System pump suction is manually transferred to the reactor building sump.

BASES

BACKGROUND

Reactor Building Spray System (continued)

The Reactor Building Spray System provides a spray of relatively cold borated water into the upper regions of containment to reduce the containment pressure and temperature and to reduce the concentration of fission products in the containment atmosphere during an accident. In the recirculation mode of operation, heat is removed from the reactor building sump water by the decay heat removal coolers. Each train of the Reactor Building Spray System provides adequate spray coverage to meet the system design requirements for containment heat removal.

The Reactor Building Spray System is actuated automatically by a containment High-High pressure signal. An automatic actuation opens the Reactor Building Spray System pump discharge valves and starts the two Reactor Building Spray System pumps.

Reactor Building Cooling System

The Reactor Building Cooling System consists of three reactor building cooling trains. Each cooling train is equipped with cooling coils, and an axial vane flow fan driven by a two speed electric motor.

During normal operation, two reactor building cooling trains with two fans operating at high speed, serve to cool the containment atmosphere. The third unit is on standby. Upon receipt of an emergency signal, the two operating cooling fans running at high speed will automatically change to low speed, and the idle unit is energized at low speed. The fans are operated at the lower speed during accident conditions to prevent motor overload from the higher density atmosphere.

APPLICABLE SAFETY ANALYSES

The Reactor Building Spray System and Reactor Building Cooling System reduce the temperature and pressure following an accident. The limiting accidents considered are the loss of coolant accident (LOCA) and the steam line break. The postulated accidents are analyzed, with regard to containment ES systems, assuming the loss of one ES bus. This is the worst-case single active failure, resulting in one train of the Reactor Building Spray System and one train of the Reactor Building Cooling System being inoperable.

The analysis and evaluation show that, under the worst-case scenario (LOCA with worst-case single active failure), the highest peak containment pressure is 58.9 psig. The analysis shows that the peak containment

BASES

APPLICABLE SAFETY ANALYSES (continued)

temperature is 285°F. Both results are less than the design values. The analyses and evaluations assume a power level of 2619 MWt, one reactor building spray train and two reactor building cooling trains operating, and initial (pre-accident) conditions of 110°F and 16.2 psia. The analyses also assume a delayed initiation to provide conservative peak calculated containment pressure and temperature responses.

The Reactor Building Spray System total delay time of 92 seconds includes Keowee Hydro Unit startup (for loss of offsite power), reactor building spray pump startup, and spray line filling (Ref. 2).

Reactor building cooling train performance for post accident conditions is given in Reference 2. The result of the analysis is that any combination of two trains can provide 100% of the required cooling capacity during the post accident condition. The train post accident cooling capacity under varying containment ambient conditions is also shown in Reference 2.

The Reactor Building Cooling System total delay time of 78 seconds includes signal delay, Keowee Hydro Unit startup (for loss of offsite power), low pressure service water pump startup and low pressure service water valve stroke times (Ref. 2).

The Reactor Building Spray System and the Reactor Building Cooling System satisfy Criterion 3 of 10 CFR 50.36 (Ref. 3).

LCO

During an accident, a minimum of two reactor building cooling trains and one reactor building spray train are required to maintain the containment pressure and temperature following a LOCA. Additionally, one reactor building spray train is required to remove iodine from the containment atmosphere and maintain concentrations below those assumed in the safety analysis. To ensure that these requirements are met, two reactor building spray trains and three reactor building cooling trains must be OPERABLE in MODES 1 and 2. In MODES 3 or 4, one reactor building spray train and two reactor building cooling trains are required to be OPERABLE. The LCO is provided with a note that clarifies this requirement. Therefore, in the event of an accident, the minimum requirements are met, assuming the worst-case single active failure occurs.

Each reactor building spray train shall include a spray pump, spray headers, nozzles, valves, piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the BWST (via the

BASES

LCO (continued)

LPI System) upon an Engineered Safeguards Protective System signal and manually transferring suction to the reactor building sump. The safety grade flow indicator of an RBS train and the safety grade flow indicator of the associated LPI train are both required to be OPERABLE to support RBS train OPERABILITY.

Each reactor building cooling train shall include cooling coils, fusible dropout plates, an axial vane flow fan, instruments, valves, and controls to ensure an OPERABLE flow path. Valve LPSW-108 shall be locked open to support system OPERABILITY.

APPLICABILITY

In MODES 1, 2, 3, and 4, an accident could cause a release of radioactive material to containment and an increase in containment pressure and temperature, requiring the operation of the reactor building spray trains and reactor building cooling trains.

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Thus, the Reactor Building Spray System and the Reactor Building Cooling System are not required to be OPERABLE in MODES 5 and 6.

ACTIONS

A.1

With one reactor building spray train inoperable in MODE 1 or 2, the inoperable reactor building spray train must be restored to OPERABLE status within 7 days. In this Condition, the remaining OPERABLE spray and cooling trains are adequate to perform the iodine removal and containment cooling functions. The 7 day Completion Time takes into account the redundant heat removal capability afforded by the OPERABLE reactor building spray train, reasonable time for repairs, and the low probability of an accident occurring during this period.

The 14 day portion of the Completion Time for Required Action A.1 is based upon engineering judgment. It takes into account the low probability of coincident entry into two Conditions in this LCO coupled with the low probability of an accident occurring during this time. Refer to Section 1.3, Completion Times, for a more detailed discussion of the purpose of the "from discovery of failure to meet the LCO" portion of the Completion Time.

BASES

ACTIONS (continued)

B.1

With one of the reactor building cooling trains inoperable in MODE 1 or 2, the inoperable reactor building cooling train must be restored to OPERABLE status within 7 days. The components in this degraded condition provide iodine removal capabilities and are capable of providing at least 100% of the heat removal needs after an accident. The 7 day Completion Time was developed taking into account the redundant heat removal capabilities afforded by combinations of the Reactor Building Spray System and Reactor Building Cooling System and the low probability of an accident occurring during this period.

The 14 day portion of the Completion Time for Required Action B.1 is based upon engineering judgment. It takes into account the low probability of coincident entry into two Conditions in this LCO coupled with the low probability of an accident occurring during this time. Refer to Section 1.3 for a more detailed discussion of the purpose of the "from discovery of failure to meet the LCO" portion of the Completion Time.

C.1

With one reactor building spray train and one reactor building cooling train inoperable in MODE 1 or 2, at least one of the inoperable trains must be restored to OPERABLE status within 24 hours. In this Condition, the remaining OPERABLE spray and cooling trains are adequate to provide iodine removal capabilities and are capable of providing at least 100% of the heat removal needs after an accident. The 24 hour Completion Time takes into account the heat removal capability afforded by the remaining OPERABLE spray train and cooling trains, reasonable time for repairs, and the low probability of an accident occurring during this period.

D.1

If the Required Action and associated Completion Time of Condition A, B or C are not met, the unit must be brought to a MODE in which the LCO, as modified by the Note, does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 12 hours. The allowed Completion Time is reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

BASES

ACTIONS (continued)

E.1

With one of the required reactor building cooling trains inoperable in MODE 3 or 4, the required reactor building cooling train must be restored to OPERABLE status within 24 hours.

The 24 hour Completion Time is reasonable based on engineering judgement taking into account the iodine and heat removal capabilities of the remaining required train of reactor building spray and cooling.

F.1

With one required reactor building spray train inoperable in MODE 3 or 4, the required reactor building spray train must be restored to OPERABLE status within 24 hours. The 24 hour Completion Time is reasonable based on engineering judgement taking into account the heat removal capabilities of the remaining required trains of reactor building cooling.

G.1

If the Required Actions and associated Completion Times of Condition E or F of this LCO are 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 MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

H.1

With two reactor building spray trains, two reactor building cooling trains or any combination of three or more reactor building spray and reactor building cooling trains inoperable in MODE 1 or 2, the unit is in a condition outside the accident analysis. Therefore, LCO 3.0.3 must be entered immediately.

With any combination of two or more required reactor building spray and reactor building cooling trains inoperable in MODE 3 or 4, the unit is in a condition outside the accident analysis. Therefore, LCO 3.0.3 must be entered immediately.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.5.1

Verifying the correct alignment for manual and non-automatic power operated valves in the reactor building spray flow path provides assurance that the proper flow paths will exist for Reactor Building Spray System operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these were verified to be in the correct position prior to locking, sealing, or securing. Similarly, this SR does not apply to automatic valves since automatic valves actuate to their required position upon an accident signal. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those valves outside containment and capable of potentially being mispositioned are in the correct position.

SR 3.6.5.2

Operating each required reactor building cooling train fan unit for ≥ 15 minutes ensures that all trains are OPERABLE and that all associated controls are functioning properly. It also ensures that blockage, fan or motor failure, or excessive vibration can be detected for corrective action. The 31 day Frequency was developed considering the known reliability of the fan units and controls, the three train redundancy available, and the low probability of a significant degradation of the reactor building cooling trains occurring between surveillances and has been shown to be acceptable through operating experience.

SR 3.6.5.3

Verifying that each required Reactor Building Spray pump's developed head at the flow test point is greater than or equal to the required developed head ensures that spray pump performance has not degraded during the cycle. Flow and differential pressure are normal tests of centrifugal pump performance required by Section XI of the ASME Code (Ref. 4). Since the Reactor Building Spray System pumps cannot be tested with flow through the spray headers, they are tested on recirculation flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and may detect incipient failures by indicating abnormal performance. The Frequency of this SR is in accordance with the Inservice Testing Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.5.4

Verifying the containment heat removal capability provides assurance that the containment heat removal systems are capable of maintaining containment temperature below design limits following an accident. This test verifies the heat removal capability of the Low Pressure Injection (LPI) Coolers and Reactor Building Cooling Units. The 18 month Frequency was developed considering the known reliability of the low pressure service water, reactor building spray and reactor building cooling systems and other testing performed at shorter intervals that is intended to identify the possible loss of heat removal capability.

SR 3.6.5.5 and SR 3.6.5.6

These SRs require verification that each automatic reactor building spray valve actuates to its correct position and that each reactor building spray pump starts upon receipt of an actual or simulated actuation signal. The test will be considered satisfactory if visual observation and control board indication verifies that all components have responded to the actuation signal properly; the appropriate pump breakers have closed, and all valves have completed their travel. This SR is not required for valves that are locked, sealed, or otherwise secured in position under administrative controls. The 18 month Frequency is based on the need to perform these Surveillances under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillances were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillances when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.5.7

This SR requires verification that each required reactor building cooling train actuates upon receipt of an actual or simulated actuation signal. The test will be considered satisfactory if control board indication verifies that all components have responded to the actuation signal properly, the appropriate valves have completed their travel, and fans are running at half speed. The 18 month Frequency is based on engineering judgment and has been shown to be acceptable through operating experience. See SR 3.6.5.5 and SR 3.6.5.6, above, for further discussion of the basis for the 18 month Frequency.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.5.8

With the reactor building spray header isolated and drained of any solution, station compressed air is introduced into the spray headers to verify the availability of the headers and spray nozzles. Performance of this Surveillance demonstrates that each spray nozzle is unobstructed and provides assurance that spray coverage of the containment during an accident is not degraded. Due to the passive nature of the design of the nozzles, a test at 10 year intervals is considered adequate to detect obstruction of the spray nozzles.

REFERENCES

1. UFSAR, Section 3.1.
2. UFSAR, Section 6.2.
3. 10 CFR 50.36.
4. ASME, Boiler and Pressure Vessel Code, Section XI.

# B 3.7 PLANT SYSTEMS

## B 3.7.1 Main Steam Relief Valves (MSRVs)

BASES

BACKGROUND

The primary purpose of the MSRVs is to provide overpressure protection for the secondary system. The MSRVs also provide protection against overpressurizing the reactor coolant pressure boundary (RCPB) by providing a heat sink for removal of energy from the Reactor Coolant System (RCS) if the preferred heat sink, provided by the Condenser and Circulating Water System, is not available.

Eight MSRVs are located on each main steam header, outside containment as described in the UFSAR, Section 10.3 (Ref. 1). The MSRV rated capacity passes the full steam flow at 114% RTP with the valves full open. This meets the requirements of the ASME Code, Section III (Ref. 2). The MSRV design includes staggered setpoints, (Ref. 1) so that only the needed number of valves will actuate. Staggered setpoints reduce the potential for valve chattering because of insufficient steam pressure to fully open the valves.

APPLICABLE SAFETY ANALYSES

The design basis of the MSRVs (Ref. 2) is to limit secondary system pressure to ≤ 110% of design pressure when passing 105% of design steam flow. This design basis is sufficient to cope with any anticipated transient or accident considered in the accident and transient analysis.

The events that challenge the relieving capacity of the MSRVs, and thus RCS pressure, are those characterized as decreased heat removal or increased heat addition events. MSRV relief capacity is utilized in the UFSAR (Ref. 3 and Ref. 4) for mitigation of the following events:

a. Loss of main feedwater;

b. Steam line break;

c. Steam generator tube rupture;

d. Rod withdrawal at rated power; and

e. Loss of Electric Load.

BASES

APPLICABLE SAFETY ANALYSIS (continued)

The MSRVs satisfy Criterion 3 of 10 CFR 50.36, (Ref. 5).

LCO

The MSRVs are provided to prevent overpressurization as discussed in the Applicable Safety Analysis section of these Bases. The LCO requires sixteen MSRVs, eight on each main steam line, to be OPERABLE to ensure compliance with the ASME Code following accidents and transients initiated at full power. Operation with less than a full complement of MSRVs is not permitted. To be OPERABLE, lift setpoints must remain within limits, specified in the UFSAR.

The safety function of the MSRVs is to open, relieve steam generator overpressure, and reseat when pressure has been reduced.

OPERABILITY of the MSRVs requires periodic surveillance testing in accordance with the Inservice Testing Program.

The lift settings correspond to ambient conditions of the valve at nominal operating temperature and pressure.

This LCO provides assurance that the MSRVs will perform the design safety function.

APPLICABILITY

In MODES 1, 2, and 3, the MSRVs must be OPERABLE to prevent overpressurization of the main steam system.

In MODES 4 and 5, there is no credible transient requiring the MSRVs.

The steam generators are not normally used for heat removal in MODES 5 and 6, and thus cannot be overpressurized. There is no requirement for the MSRVs to be OPERABLE in these MODES.

ACTIONS

A.1 and A.2

With one or more MSRVs inoperable, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 12 hours, and in MODE 4 within 18 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 (continued)

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1

This SR verifies the OPERABILITY of the MSRVs by the verification of MSRV lift setpoints in accordance with the Inservice Testing Program. The safety and relief valve tests are performed in accordance with ANSI/ASME OM-1-1987 (Ref. 5) and include the following for MSRVs:

a. Visual examination;

b. Seat tightness determination;

c. Setpoint pressure determination (lift setting);

d. Compliance with owner's seat tightness criteria; and

e. Verification of the balancing device integrity on balanced valves.

The ANSI/ASME Standard requires the testing of all valves every 5 years, with a minimum of 20% of the valves tested every 24 months. Reference 4 provides the activities and frequencies necessary to satisfy the requirements.

This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. The MSRVs may be either bench tested or tested in situ at hot conditions using an assist device to simulate lift pressure. If the MSRVs are not tested at hot conditions, the lift setting pressure must be corrected to ambient conditions of the valve at operating temperature and pressure.

REFERENCES

1. UFSAR, Section 10.3.

2. ASME, Boiler and Pressure Vessel Code, Section III, Article NC-7000, Class 2 Components.

3. UFSAR, Chapter 15.

4. UFSAR, Section 10.3.3.

5. 10 CFR 50.36.

6. ANSI/ASME OM-1-1987.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The TSVs remain open during power operation. These valves close upon a reactor trip.

a. For an HELB or an MSLB inside containment, steam is discharged into containment from both steam generators until closure of the TSVs. After TSV closure, steam is discharged into containment only from the affected steam generator.

b. An MSLB outside of containment and upstream from the TSVs is not a containment pressurization concern. The uncontrolled blowdown of both steam generators must be prevented to limit the potential for uncontrolled RCS cooldown and positive reactivity addition. Closure of the TSVs isolates the break and limits the blowdown to a single steam generator.

c. Steam flow to the turbine if not controlled by the turbine control valves will terminate on closing the TSVs.

d. Following a steam generator tube rupture, closure of the TSVs isolates the ruptured steam generator from the intact steam generator.

The TSVs satisfy Criterion 3 of 10 CFR 50.36, (Ref. 3).

LCO

This LCO requires that the two TSVs in each steam line be OPERABLE. The TSVs are considered OPERABLE when the isolation times are within limits and they close on an isolation actuation signal.

This LCO provides assurance that the TSVs will perform their design safety function to mitigate the consequences of accidents that could result in offsite exposures comparable to the 10 CFR 100 limits (Ref. 4).

APPLICABILITY

The TSVs must be OPERABLE in MODES 1, 2 and 3 with any TSVs open. In these conditions when there is significant mass and energy in the RCS and steam generators, the TSVs must be OPERABLE or closed. When the TSVs are closed, they are already performing the safety function.

In MODE 4, the steam generator energy is low. Therefore, the TSVs are not required to be OPERABLE.

BASES

APPLICABILITY (continued)

In MODES 5 and 6, the steam generators do not contain a significant amount of energy because their temperature is below the boiling point of water; therefore, the TSVs are not required for isolation of potential high energy secondary system pipe breaks in these MODES.

ACTIONS

A.1

With one or both TSVs for one main steam line inoperable in MODE 1, action must be taken to restore the components to OPERABLE status within 8 hours. Some repairs can be made to the TSV with the unit hot. The 8 hour Completion Time is reasonable, considering the probability of an accident that would require actuation of the TSVs occurring during this time interval. The turbine control valves may be available to provide the isolation for the postulated accidents although control valve response is not as rapid.

The Completion Time is reasonable because the TSVs isolate a closed system which provides an additional barrier against releases.

B.1

If the TSVs cannot be restored to OPERABLE status within 12 hours, the unit must be placed in MODE 2 and the inoperable TSVs closed within the next 6 hours. The Completion Time is reasonable, based on operating experience, to reach MODE 2.

C.1 and C.2

Condition C is modified by a Note indicating that separate Condition entry is allowed for each TSV.

Since the TSVs are required to be OPERABLE in MODES 2 and 3, the inoperable TSVs may either be restored to OPERABLE status or closed. When closed, the TSVs are already in the position required by the assumptions in the safety analysis.

The 8 hour Completion Time is reasonable, considering the low probability of an accident occurring during this time period that would require closure of the TSVs.

BASES

ACTIONS

C.1 and C.2 (continued)

Inoperable TSVs that cannot be restored to OPERABLE status within the specified Completion Time, but are closed, must be verified on a periodic basis to be closed. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of TSV status indications available in the control room, and other administrative controls, to ensure these valves are in the closed position.

D.1 and D.2

If the TSV cannot be restored to OPERABLE status or closed in the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 12 hours and in MODE 4 within 18 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from MODE 2 conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.2.1

This SR verifies that TSV closure time of each TSV is ≤ 1.0 second on an actual or simulated actuation signal from Channel A. The 1.0 second TSV closure time is assumed in the accident and containment analyses. This Surveillance is normally performed upon returning the unit to operation following a refueling outage.

The Frequency for this SR is 18 months. The 18 month Frequency to demonstrate valve closure time is based on the refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

This test is conducted in MODE 3, with the unit at operating temperature and pressure, as discussed in the Reference 5 exercising requirements. This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. This allows delaying testing until MODE 3 in order to establish conditions consistent with those under which the acceptance criterion was generated.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.2.2

This SR verifies that TSV closure time of each TSV is ≤ 15.0 seconds on an actual or simulated actuation signal from Channel B. This Surveillance is normally performed upon returning the unit to operation following a refueling outage.

The Frequency for this SR is 18 months. The 18 month Frequency to demonstrate valve closure time is based on the refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

This test is conducted in MODE 3, with the unit at operating temperature and pressure, as discussed in the Reference 5 exercising requirements. This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. This allows delaying testing until MODE 3 in order to establish conditions consistent with those under which the acceptance criterion was generated.

REFERENCES

1. UFSAR, Section 10.3.
2. UFSAR, Section 15.13.
3. 10 CFR 50.36.
4. 10 CFR 100.11.

## B 3.7 PLANT SYSTEMS

### B 3.7.3 Main Feedwater Control Valves (MFCVs), and Startup Feedwater Control Valves (SFCVs)

BASES

**BACKGROUND**

The main feedwater isolation valves (MFIVs) for each steam generator consist of the MFCVs and the SFCVs. The MFIVs isolate main feedwater (MFW) flow to the secondary side of the steam generators following a high energy line break (HELB). The consequences of events occurring in the main steam lines will be mitigated by their closure. Closing the MFCVs and associated SFCVs valves effectively terminates the addition of feedwater to an affected steam generator, limiting the mass and energy release for steam line breaks (SLBs) inside containment and reducing the cooldown effects for SLBs.

The MFIVs close on receipt of a MSLB detection signal generated by low steam header pressure. The MFIVs can also be closed manually.

**APPLICABLE SAFETY ANALYSES**

The design basis of the MFIVs is established by the containment analysis for the main steam line break (MSLB).

Failure of an MFIV to close following an MSLB, can result in additional mass and energy being delivered to the steam generators, contributing to cooldown. This failure also results in additional mass and energy releases following an MSLB.

The MFIVs satisfy Criterion 3 of 10 CFR 50.36 (Ref. 1).

**LCO**

This LCO ensures that the MFIVs will isolate MFW flow to the steam generators following a main steam line break.

Two MFCVs and two SFCVs are required to be OPERABLE. The MFIVs are considered OPERABLE when the isolation times are within limits and they close on a feedwater isolation actuation signal.

Automatic initiation instrumentation is not required to be OPERABLE in MODE 3 when main steam header pressure is < 700 psig in accordance with LCO 3.3.11, "Main Steam Line Break (MSLB) Detection and Main

BASES

LCO (continued)

Feedwater (MFW) Isolation Instrumentation." When automatic initiation circuitry is not required to be OPERABLE, the MFCVs and SFCVs are OPERABLE provided manual closure capability is OPERABLE. Automatic initiation is not required in this condition since additional time is available for the operator to manually close the valves if required.

Failure to meet the LCO requirements can result in excessive cooldown and additional mass and energy being released to containment following an MSLB inside containment.

APPLICABILITY

The MFCVs and SFCVs must be OPERABLE whenever there is significant mass and energy in the RCS and steam generators.

In MODES 1, 2, and 3, the MFCVs and SFCVs are required to be OPERABLE in order to limit the cooldown and the amount of available fluid that could be added to containment in the case of an MSLB inside containment. When the valves are closed, they are already performing their safety function.

In MODES 4, 5, and 6, feedwater and steam generator energy are low. Therefore, the MFCVs and SFCVs are not required for isolation of potential main steam pipe breaks in these MODES.

ACTIONS

The ACTIONS table is modified by a Note indicating that separate Condition entry is allowed for each valve.

A.1 and A.2

With one MFCV in one or more flow paths inoperable, action must be taken to restore the affected valves to OPERABLE status, or to close or isolate inoperable affected valves within 8 hours. When these valves are closed or isolated, they are performing their required safety function.

The 8 hour Completion Time provides a reasonable time to restore an inoperable MFIV to OPERABLE status and is acceptable due to the low probability of an event occurring during this time period that would require isolation of the MFW flow paths.

BASES

ACTIONS

A.1 and A.2 (continued)

Inoperable MFCVs that are closed or isolated must be verified on a periodic basis that they are closed or isolated. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of valve status indications available in the control room, and other administrative controls, to ensure that these valves are closed or isolated.

B.1 and B.2

With one SFCV in one or more flow paths inoperable, action must be taken to restore the affected valves to OPERABLE status, or to close or isolate inoperable affected valves within 8 hours. When these valves are closed or isolated, they are performing their required safety function.

The 8 hour Completion Time provides a reasonable time to restore an inoperable MFIV to OPERABLE status and is acceptable due to the low probability of an event occurring during this time period that would require isolation of the MFW flow paths.

Inoperable SFCVs that are closed or isolated must be verified on a periodic basis that they are closed or isolated. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of valve status indications available in the control room, and other administrative controls, to ensure that these valves are closed or isolated.

C.1 and C.2

If the Required Actions and associated Completion Time are not met, the unit must be in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 12 hours and in MODE 4 within 18 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 (continued)

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1

This SR verifies that the closure time of each MFCV and SFCV is ≤ 25 seconds on an actual or simulated actuation signal. The 25 seconds includes a 10 second signal delay and 15 seconds for valve movement.

The MFCV and SFCV closure time is assumed in the containment analyses. This Surveillance is normally performed upon returning the unit to operation following a refueling outage. The MFCV and SFCV should not be tested at power since even a part stroke exercise increases the risk of a valve closure with the unit generating power. This is consistent with the ASME Code, Section XI (Ref. 2) requirements during operation in MODES 1 and 2.

This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR.

The Frequency for this SR is in accordance with the Inservice Testing Program.

REFERENCES

1. 10 CFR 50.36.

2. ASME, Boiler and Pressure Vessel Code, Section XI.

## B 3.7 PLANT SYSTEMS

### B 3.7.4 Atmospheric Dump Valve (ADV) Flow Paths

BASES

BACKGROUND

The ADV flow paths provide a method for cooling the unit to decay heat removal (DHR) entry conditions, should the preferred heat sink via the Turbine Bypass System to the condenser not be available, as discussed in the UFSAR (Ref. 2). This is done in conjunction with the secondary cooling water from the Emergency Feedwater (EFW) System.

The steam generator tube rupture (SGTR) analysis (Ref. 3) credits operator action to depressurize the steam generators by opening each of the ADV flow paths.

For each steam generator, the ADV flow path is comprised of the atmospheric dump block valve bypass (1" bypass), the atmospheric vent valve (a 12" block valve), the atmospheric dump control valve (i.e., throttle valve), and the atmospheric vent block valve (i.e., isolation valve). The throttle valve and the isolation valve are in parallel and are located downstream of the atmospheric vent valve.

The atmospheric vent valve should be opened prior to opening the throttle valve or isolation valve. This is accomplished by first opening the atmospheric dump block valve bypass.

This equalizes the differential pressure across the atmospheric vent valve. Once the atmospheric vent valve is opened, the cool down rate is controlled using the throttle valve. If additional relief capacity is needed, the isolation valve can be opened. The capacity of the throttle or isolation valve exceeds decay heat loads and is sufficient to cool down the plant.

BASES

APPLICABLE SAFETY ANALYSIS

The SGTR analysis credits operator action to depressurize the steam generators by opening both ADV flow paths (i.e., the ADV flow path for each steam generator) within 40 minutes of identifying the ruptured steam generator. Within this 40-minute time period, the operators are only required to open the bypass valve, the block valve, and the throttle valve. However, later in the event, the analysis also assumes that the operators will open the isolation valves in each ADV flow path.

The ADV flow paths satisfy Criterion 3 of 10 CFR 50.36 (Ref.1).

LCO

The ADV flow path for each steam generator is required to be OPERABLE. The failure to meet the LCO can result in the inability to depressurize the steam generators following a SGTR.

An ADV flow path is considered OPERABLE when it is capable of providing a controlled relief of the main steam flow, and each valve which comprises the ADV flow path is capable of opening and closing.

APPLICABILITY

The ADV flow path for each steam generator is required to be OPERABLE in MODES 1, 2, and 3, and in MODE 4, when a steam generator is being relied upon for heat removal. In MODE 4, steam generators are relied upon for heat removal whenever an RCS loop is required to be OPERABLE or operating to satisfy LCO 3.4.5, "RCS Loops - MODE 4" or available to transfer decay heat to satisfy LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled." The steam generators do not contain a significant amount of energy in MODE 4 when the unit is not relying upon a steam generator for heat transfer, and MODES 5 and 6; therefore, the ADV flow paths are not required to be OPERABLE in these MODES and condition.

BASES

ACTIONS

A.1 and A.2

With one or both of the ADV flow path(s) inoperable, the Unit must be placed in a condition in which the LCO does not apply. To achieve this status, the Unit must be paced in at least MODE 3 within 12 hours, and at least MODE 4 without reliance on a steam generator for heat removal within 24 hours. The Completion Times are reasonable, based on operating experience, to reach the required Unit conditions from full power conditions in an orderly manner and without challenging Unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.4.1

To perform a controlled cool down of the RCS, the valves that comprise the ADV flow path for each steam generator must be able to perform the following functions:

a) the atmospheric dump block valve bypass and the atmospheric vent valve must be capable of being opened and closed; and

b) the atmospheric dump control valve and atmospheric vent block valve must be capable of being opened and throttled through their full range.

This SR ensures that the valves that comprise the ADV flow path for each steam generator are cycled through the full control range at least once per 18 months. Performance of inservice testing or use of an ADV flow path during a unit cool down satisfies this requirement. This surveillance does not require the valves to be tested at pressure. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

REFERENCES

1. 10 CFR 50.36.

2. UFSAR, Section 10.3.

3. UFSAR, Section 15.9.

## B 3.7 PLANT SYSTEMS

### B 3.7.5 Emergency Feedwater (EFW) System

BASES

BACKGROUND

The EFW System automatically supplies feedwater to the steam generators to remove decay heat from the Reactor Coolant System (RCS) upon the loss of normal feedwater supply. The EFW pumps take suction through suction lines from the upper surge tank (UST) and condenser Hotwell and pump to the steam generator secondary side through the EFW nozzles. The steam generators function as a heat sink for core decay heat. The heat load is dissipated by releasing steam to the atmosphere from the steam generators via the main steam relief valves (MSRVs) (LCO 3.7.1, "Main Steam Relief Valves (MSRVs)"), or atmospheric dump valves (ADVs). If the main condenser is available, steam may be released via the Turbine Bypass System and recirculated to the condenser Hotwell.

The EFW System consists of two motor driven EFW pumps and one turbine driven EFW pump, any one of which can provide the required heat removal capability. Thus, the requirements for diversity in motive power sources for the EFW System are met. The steam turbine driven EFW pump receives steam from either of the two main steam headers, upstream of the main turbine stop valves (TSVs), or from the Auxiliary Steam System which can be supplied from the other two unit's Main Steam System. The EFW System supplies a common header capable of feeding either or both steam generators. The EFW System normally receives a supply of water from the UST. The EFW System can also be aligned to the condenser Hotwell. An additional source of water is the condensate storage tank which can be pumped to the USTs.

The EFW System is capable of supplying feedwater to the steam generators during normal unit startup, shutdown, and hot standby conditions.

The three emergency feedwater pumps are started automatically upon a loss of both main feedwater pumps or a signal from the ATWS Mitigation System Actuation Circuitry (AMSAC). The two motor driven emergency feedwater pumps are also started automatically upon a low steam generator level which exists for at least 30 seconds.

The EFW System is discussed in the UFSAR, Section 10.4.7, (Ref. 1).

BASES (continued)

APPLICABLE SAFETY ANALYSES

The EFW System mitigates the consequences of any event with a loss of normal feedwater.

The design basis of the EFW System is to supply water to the steam generator to remove decay heat and other residual heat by delivering at least the minimum required flow rate to the steam generators at 1064 psia for the MDEFW pump and 1100 psig for the TDEFW pump.

The limiting event for the EFW System is the loss of main feedwater with offsite power available.

The EFW System design is such that it can perform its function following a loss of the turbine driven main feedwater pumps combined with a loss of normal or emergency electric power.

The EFW System satisfies Criterion 3 of 10 CFR 50.36 (Ref. 2).

LCO

This LCO provides assurance that the EFW System will perform its design safety function to mitigate the consequences of accidents that could result in overpressurization of the reactor coolant pressure boundary. Three independent EFW pumps and two flow paths are required to be OPERABLE to ensure the availability of residual heat removal capability for all events accompanied by a loss of offsite power and a single failure. This is accomplished by powering one pump by a steam driven turbine supplied with steam from a source not isolated by the closure of the TSVs, and two pumps from a power source that, in the event of loss of offsite power, is supplied by the emergency power source.

The EFW System is considered to be OPERABLE when the components and flow paths required to provide EFW flow to the steam generators are OPERABLE. This requires that the turbine driven EFW pump be OPERABLE with a steam supply from either one of the main steam lines upstream of the TSVs or from the Auxiliary Steam System. The two motor driven EFW pump(s) are also required to be OPERABLE. The two required flow paths shall also be OPERABLE. A flowpath is defined as the flowpath to either steam generator including associated valves and piping capable of being supplied by either the turbine driven pump or the associated motor driven pump. The sources of water to the EFW System are required to be OPERABLE. The associated flow paths from the EFW System sources of water to all EFW pumps also are required to be OPERABLE. In MODES 1 and 2 automatic EFW initiation is required to be

BASES

LCO (continued)

OPERABLE in accordance with Specification 3.3.14, "Emergency Feedwater (EFW) Pump Initiation Circuitry." Automatic EFW steam generator level control is required to be OPERABLE when automatic EFW initiation is required to be OPERABLE. EFW automatic initiation instrumentation is not required to be OPERABLE in MODES 3 and 4 in accordance with LCO 3.3.14. In MODES 3 and 4 the EFW System is OPERABLE provided manual initiation capability is OPERABLE. Automatic initiation is not required in MODES 3 and 4 since additional time is available in these MODES for the operator to manually initiate the system if required. When in MODE 3 and 4 automatic EFW flow control is not required to be OPERABLE provided manual steam generator level control is OPERABLE.

The LCO is modified by a Note indicating that one motor driven EFW pump and EFW flow path, is required in MODE 4 when an SG is relied upon for heat removal. This is because of reduced heat removal requirements, the short duration of MODE 4 in which feedwater is required, and the insufficient steam supply available in MODE 4 to power the turbine driven EFW pump.

APPLICABILITY

In MODES 1, 2, and 3, the EFW System is required to be OPERABLE and to function in the event that the main feedwater is lost. In MODE 4, with RCS temperature above 212°F, the EFW System may be used for heat removal via the steam generators. In MODE 4, the steam generators are used for heat removal unless the DHR System is in operation. In MODE 4 steam generators are relied upon for heat removal whenever an RCS loop is required to be OPERABLE or operating to satisfy LCO 3.4.6, "RCS Loops – Mode 4."

In MODES 5 and 6, the steam generators are not used for DHR and the EFW System is not required.

ACTIONS

A.1

With one of the motor driven EFW pumps inoperable, action must be taken to restore the MDEFW pump to OPERABLE status within 7 days. The 7 day Completion Time is reasonable, based on the following reasons:

a. The redundant OPERABLE turbine driven EFW pump(s);

b. The availability of the redundant OPERABLE motor driven EFW pump; and

BASES

ACTIONS

A.1 (continued)

c. The low probability of an event occurring that would require the EFW System during the 7 day period.

The second Completion Time for Required Action A.1 establishes a limit on the maximum time allowed for any combination of Conditions to be inoperable during any continuous failure to meet this LCO.

The 10 day Completion Time provides a limitation time allowed in this specified Condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which Conditions A and B exist concurrently. The AND connector between 7 days and 10 days dictates that both Completion Times apply simultaneously, and the more restrictive must be met.

B.1

When the turbine driven EFW pump or one EFW flow path is inoperable, action must be taken to restore the pump and flow path to OPERABLE status within 72 hours. The 72 hour Completion Time is reasonable, based on the redundant capabilities afforded by the EFW System, time needed for repairs, and the low probability of an accident occurring during this time period. The second Completion Time for Required Action B.1 establishes a limit on the maximum time allowed for any combination of Conditions to be inoperable during any continuous failure to meet this LCO.

The 10 day Completion Time provides a limitation time allowed in this specified Condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which Conditions A and B exist concurrently. The AND connector between 72 hours and 10 days dictates that both Completion Times apply simultaneously, and the more restrictive must be met.

C.1

With the two motor driven EFW pumps inoperable, action must be taken to restore at least one pump to OPERABLE status within 12 hours. The 12 hour Completion Time is reasonable, based on the redundant capabilities afforded by the turbine driven EFW pump, time needed for repairs, and the low probability of an accident occurring during this time period.

BASES

ACTIONS (continued)

D.1 and D.2

When Required Action or Completion Time for Condition A, B or C is not met or when the turbine driven EFW pump and one EFW flow path are inoperable in MODE 1, 2, or 3, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 12 hours and in MODE 4 within 24 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.

In MODE 4, with two EFW pumps and one flow path inoperable, operation is allowed to continue because only one motor driven EFW train is required in accordance with the Note that modifies the LCO. Although not required, the unit may continue to cool down and initiate DHR.

E.1

Required Action E.1 is modified by a Note indicating that all required MODE changes or power reductions are suspended until at least one EFW pump and one flow path are restored to OPERABLE status.

With all EFW pumps or flow paths inoperable in MODE 1, 2, or 3, the unit is in a seriously degraded condition. In such a condition, the unit should not be perturbed by any action, including a power change, that might result in a trip. The seriousness of this condition requires that action be started immediately to restore at least one EFW pump and flow path to OPERABLE status. LCO 3.0.3 is not applicable, as it could force the units into a less safe condition.

F.1

In MODE 4, either the steam generator loops or the DHR loops can be used to provide heat removal, which is addressed in LCO 3.4.6, "RCS Loops – MODE 4." With one required EFW pump or flow path inoperable, action must be taken to immediately restore the inoperable pump or flow path to OPERABLE status.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.7.5.1

Verifying the correct alignment for manual, and non-automatic power operated valves in the EFW water and steam supply flow paths provides assurance that the proper flow paths exist for EFW operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since those valves are verified to be in the correct position prior to locking, sealing, or securing.

This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This Surveillance does not require any testing or valve manipulation; rather, it involves verification that those valves capable of potentially being mispositioned are in the correct position.

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

SR 3.7.5.2

Verifying that each EFW pump's developed head at the flow test point is greater than or equal to the required developed head ensures that EFW pump performance has not degraded below the acceptance criteria during the cycle. Flow and differential head are normal indications of pump performance required by Section XI of the ASME Code (Ref. 3). Because it is undesirable to introduce cold EFW into the steam generators while they are operating, this test may be performed on a test flow path.

This test confirms OPERABILITY, trends performance, and detects incipient failures by indicating abnormal performance. Performance of inservice testing in the ASME Code, Section XI (Ref. 3), at 3 month intervals, satisfies this requirement.

SR 3.7.5.3

This SR verifies that EFW can be delivered to the appropriate steam generator in the event of any accident or transient that generates an Emergency Feedwater System initiation signal by demonstrating that each automatic valve in the flow path actuates to its correct position on an actual or simulated actuation signal. This SR is not required for valves that are locked, sealed, or otherwise secured in position under administrative

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.5.3 (continued)

controls. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The 18 month Frequency is also acceptable based on operating experience and design reliability of the equipment. This SR is modified by a Note which states that the SR is not required in MODES 3 and 4. In MODES 3 and 4, the heat removal requirements would be less, thereby providing more time for operator action to manually start the required EFW pump.

SR 3.7.5.4

This SR verifies that each EFW pump starts in the event of any accident or transient that generates an initiation signal. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. This SR is modified by a Note which states that the SR is not required in MODES 3 and 4. In MODE 3 and 4, the heat removal requirements would be less, thereby providing more time for operator action to manually start the required EFW pump.

SR 3.7.5.5

This SR ensures that the EFW System is properly aligned by verifying the flow paths to each steam generator prior to entering MODE 2 after more than 30 days in MODE 5 or 6. OPERABILITY of EFW flow paths must be demonstrated before sufficient core heat is generated that would require the operation of the EFW System during a subsequent shutdown. The Frequency is reasonable, based on engineering judgment, in view of other administrative controls to ensure that the flow paths are OPERABLE. To further ensure EFW System alignment, flow path OPERABILITY is verified, following extended outages to determine no misalignment of valves has occurred. This SR ensures that the flow path from the UST to the steam generator is properly aligned.

BASES (continued)

REFERENCES

1. UFSAR, Section 10.4.7.

2. 10 CFR 50.36.

3. ASME, Boiler and Pressure Vessel Code, Section XI.

## B 3.7 PLANT SYSTEMS

### B 3.7.6 Condensate Storage Tank (CST), Upper Surge Tank (UST), and Hotwell (HW)

BASES

BACKGROUND

The CST, UST, and HW provide a source of water to the steam generators for removing decay and sensible heat from the Reactor Coolant System (RCS). The UST and HW provide a passive flow of water to the Emergency Feedwater (EFW) System (LCO 3.7.5, "Emergency Feedwater (EFW) System"). The steam produced is released to the atmosphere by the main steam safety valves (MSSVs) and the atmospheric dump valves.

The preferred means of heat removal is to discharge to the condenser by the nonsafety grade path of the turbine bypass valves.

The emergency feedwater pumps are normally aligned to the upper surge tanks (UST). The UST provides the initial source of water for the EFW System. When that supply is exhausted, the pumps may be aligned to draw water from the hotwell. The UST can be replenished by pumping from the condensate storage tank (CST) or from the Makeup Demineralized Water System. A minimum level of 6 feet (at least 30,000 gallons) is maintained in the UST to assure an adequate source of water to the EFW until other sources can be aligned. This minimum level of 6 feet includes an allowance for instrument uncertainty and depletion of inventory while transferring the EFW suction to an alternative source of water.

The UST and the piping connecting them to the EFW pumps has been analyzed and qualified to withstand a design basis seismic event. This includes piping up to the first normally closed valve. The hotwell and connected piping used for the EFW pump suction supply has been evaluated using a "seismic experience" approach and found capable of withstanding a seismic event. Although the evaluation methodology is not recognized for licensing basis, this secondary water supply is considered to be a "seismic assured source of water." Feedwater is also available from alternate source(s).

A description of the condensate/feedwater reserves available to the EFW System is found in the UFSAR, Section 10.4, (Ref. 1).

BASES (continued)

APPLICABLE SAFETY ANALYSES

The CST, UST, and HW provides cooling water to remove decay heat following events in the accident analysis, as discussed in the UFSAR, Chapters 10 and 15 (Refs. 2 and 3, respectively).

The water inventory from the CST will not be available during a station blackout due to unavailability of power to the CST transfer pumps.

The required inventory in the UST, CST, and HW has not been specifically analyzed regarding the capability to permit cooling the unit down and transferring to the decay heat removal loops. The required inventory permits maintaining EFW capability until either cooling capability using the main condenser can be restored or until the Standby Shutdown Facility (SSF) (LCO 3.10.1, "Standby Shutdown Facility (SSF)") is placed in service.

The CST, UST, and HW satisfy Criteria 2 and 3 of 10 CFR 50.36 (Ref. 4).

LCO

To satisfy LCO requirements, the UST, CST, and HW must contain the specified volume of water available to the EFW System.

The OPERABILITY of CST, UST, and HW is determined by maintaining the tank volume at or above the minimum required volume.

APPLICABILITY

In MODES 1, 2, 3, and in MODE 4, when steam generator is being relied upon for heat removal, the CST, UST, and HW are required to be OPERABLE.

In MODES 5 and 6, the CST, UST, and HW are not required because the EFW System is not required.

ACTIONS

A.1 and A.2

If the requirements of the LCO are not met, the unit must be placed in a MODE in which the LCO does not apply, with the DHR System in operation. To achieve this status, the unit must be placed in at least MODE 3 within 12 hours, and in MODE 4, without reliance on steam generators for heat removal, within 24 hours. This allows an additional 6 hours for the DHR System to be placed in service after entering MODE 4.

BASES

ACTIONS

A.1 and A.2 (continued)

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.6.1

This SR verifies that the CST, UST, and HW contain the required volume of cooling water. The 12 hour Frequency is based on operating experience and the need for operator awareness of unit evolutions that may affect the CST, UST, and HW inventory between checks. The 12 hour Frequency is considered adequate in view of other indications in the control room, including alarms to alert the operator to abnormal deviations in CST, UST, and HW levels.

REFERENCES

1. UFSAR, Section 10.4.
2. UFSAR, Chapter 10.
3. UFSAR, Chapter 15.
4. 10 CFR 50.36.

## B 3.7 PLANT SYSTEMS

## B 3.7.7 Low Pressure Service Water (LPSW) System

BASES

**BACKGROUND**

The LPSW System provides a heat sink for the removal of process and operating heat from safety related components during a transient or accident. During normal operation and normal shutdown, the LPSW System also provides this function for various safety related and nonsafety related components.

The LPSW system for Unit 1 and Unit 2 is shared and consists of three LPSW pumps which can supply multiple combinations of path ways to supply required components. The LPSW system for Unit 3 consists of two LPSW pumps which can supply multiple combinations of path ways to supply required components. Although multiple combinations of path ways exist, only one flow path is necessary, since no single failure of an active component can prevent the LPSW system from supplying necessary components. The pumps and valves are remote manually aligned, except in the unlikely event of a loss of coolant accident (LOCA) or other accidents. The pumps are automatically started upon receipt of an Engineered Safeguards actuation signal, and automatic valves are aligned to their post accident positions. The LPSW System also provides cooling directly to the Reactor Building Cooling Units (RBCU) and Low Pressure Injection coolers, turbine driven EFW pump, HPI pump motor coolers, and the motor driven EFW pumps.

Additional information about the design and operation of the LPSW System, along with a list of the components served, is presented in the UFSAR, Section 9.2.2 (Ref. 1).

**APPLICABLE SAFETY ANALYSES**

The primary safety function of the LPSW is, in conjunction with a 100% capacity reactor building cooling system, (a combination of the reactor building spray and reactor building air coolers) to remove core decay heat following a design basis LOCA, as discussed in the UFSAR, Section 6.3 (Ref. 2). This provides for a gradual reduction in the temperature of the fluid, as it is supplied to the Reactor Coolant System (RCS) by the High Pressure and Low Pressure Injection pumps.

The LPSW System is designed to perform its function with a single active failure of any component, assuming loss of offsite power.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The LPSW System also cools the unit from Decay Heat Removal (DHR) System entry conditions, to MODE 5 during normal and post accident operation. The time required for this evolution is a function of the number of DHR System trains that are operating. One LPSW pump per unit and a flowpath is sufficient to remove decay heat during subsequent operations in MODES 5 and 6. This assumes a maximum LPSW System temperature of 90°F occurring simultaneously with maximum heat loads on the system.

The LPSW satisfies Criterion 3 of 10 CFR 50.36 (Ref. 2).

LCO

For the LPSW system shared by Units 1 and 2, three LPSW pumps are required to be OPERABLE to provide the required redundancy to ensure that the system functions to remove post accident heat loads, assuming the worst case single active failure occurs coincident with the loss of offsite power. The LCO is modified by a Note which requires only two LPSW pumps to be OPERABLE for Units 1 or 2 if either Unit is defueled and one LPSW pump is capable of mitigating the DBA on the fueled Unit. The Units 1 and 2 LPSW System requires only two pumps to meet the single failure criterion provided that one of the units has been defueled and the following LPSW System loads on the defueled unit are isolated: Reactor Building Cooling Units (RBCU), Reactor Building Auxiliary Coolers, Component Cooling, main turbine oil tank, reactor coolant (RC) pumps, and Low Pressure Injection (LPI) coolers.

For the LPSW system for Unit 3, two LPSW pumps are required to be OPERABLE to provide the required redundancy to ensure that the system functions to remove post accident heat loads, assuming the worst case single active failure occurs coincident with the loss of offsite power.

An LPSW flow path is considered OPERABLE when the associated piping, valves, heat exchangers, and instrumentation and controls required to perform the safety related function are OPERABLE. Any combination of path ways to supply the required components is acceptable, provided there is no single active failure which can prevent supplying necessary loads and applicable design criteria (e.g., seismic qualification) are satisfied.

APPLICABILITY

In MODES 1, 2, 3, and 4, the LPSW System is a normally operating system that is required to support the OPERABILITY of the equipment serviced by the LPSW System. Therefore, the LPSW System is required to be OPERABLE in these MODES.

BASES

APPLICABILITY (continued)

In MODES 5 and 6, the OPERABILITY requirements of the LPSW System are determined by the systems it supports.

ACTIONS

A.1

If one required LPSW pump is inoperable, action must be taken to restore the required LPSW pump to OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE LPSW pump(s) are adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure in the OPERABLE LPSW pump(s) could result in loss of LPSW system function. The 72 hour Completion Time is based on the redundant capabilities afforded by the OPERABLE pump, and the low probability of a DBA occurring during this period.

B.1 and B.2

If the LPSW pump cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 12 hours, and in MODE 5 within 60 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. The extended interval to reach MODE 5 provides additional time to restore the required LPSW pump and is reasonable considering that the potential for an accident or transient is reduced in MODE 3.

SURVEILLANCE REQUIREMENTS

SR 3.7.7.1

Verifying the correct alignment for manual, and power operated valves in the LPSW System flow path provides assurance that the proper flow paths exist for LPSW System operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since they are verified to be in the correct position prior to locking, sealing, or securing. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of potentially being mispositioned are in the correct position. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.7.1 (continued)

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

This SR is modified by a Note indicating that the isolation of components or systems supported by the LPSW System does not affect the OPERABILITY of the LPSW System.

SR 3.7.7.2

The SR verifies proper automatic operation of the LPSW System valves. The LPSW System is a normally operating system that cannot be fully actuated as part of the normal testing. This SR is not required for valves that are locked, sealed, or otherwise secured in position under administrative controls. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

SR 3.7.7.3

The SR verifies proper automatic operation of the LPSW System pumps on an actual or simulated actuation signal. The LPSW System is a normally operating system that cannot be fully actuated as part of normal testing during normal operation. The 18 month Frequency is consistent with the Inservice Testing Program. Operating experience has shown that these components usually pass the Surveillance when performed at an 18 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

REFERENCES

1. UFSAR, Section 9.2.2.

2. UFSAR, Section 6.3.

3. 10 CFR 50.36.

## B 3.7 PLANT SYSTEMS

## B 3.7.8 Emergency Condenser Circulating Water (ECCW)

BASES

BACKGROUND

Lake Keowee provides a heat sink for process and operating heat from safety related components during a transient or accident as well as during normal operation. This is done utilizing the ECCW System in conjunction with the Low Pressure Service Water System (LPSW).

The ECCW System consists of six siphon headers shared among the three Units. Each Unit's Condenser Circulating Water (CCW) inlet piping provides two ECCW siphon headers. Each siphon header takes suction from the CCW intake canal and supplies water to the CCW crossover header which connects to the LPSW suction piping. Although sharing some portions of the flow path, each CCW inlet header on a given unit is independent of the other header for the purposes of siphoning water from the intake canal to the CCW crossover header. A loss of siphon header flow from one ECCW siphon header does not prevent the other siphon header from supplying flow to the LPSW suction.

The Essential Siphon Vacuum (ESV) System is provided to remove air accumulation in the ECCW siphon headers. The ESV system consists of three ESV pumps per unit. One ESV pump is associated with each ECCW siphon header. The third ESV pump is a spare pump which can be aligned to replace either one of the other two pumps. Vacuum piping is connected to the top of the ECCW siphon header and contains an automatic float valve which prevents unacceptable amounts of water from entering the system during normal operation. Vacuum piping from each siphon header connects to a small receiver tank which functions to collect entrained liquids and increases system capacitance. The tank also provides a suitable location for installation of instrumentation.

Two Siphon Seal Water (SSW) headers are routed from the LPSW System in the turbine building to the CCW intake structure. One header is supplied from the shared Unit 1 and 2 LPSW System. The other header is supplied from the Unit 3 LPSW System. Two ESV seal supply headers, one from each SSW header, are provided and cross-connected at each ESV pump.

A solenoid valve, located downstream from ESV seal supply cross-connect, is interlocked with ESV pump controls to isolate and restore SSW to an ESV pump that has lost and regained power. The Siphon

BASES

BACKGROUND (continued)

Seal Water (SSW) System's safety function is to provide a seal water supply for the ESV pumps. The system also supplies the normal source of seal/cooling water for the CCW pumps. Supplying seal/cooling water to the CCW pumps is not a safety requirement since it is not required for accident mitigation.

Additional information on the design and operation of the system, along with a list of components served, can be found in Reference 1.

APPLICABLE SAFETY ANALYSES

The ECCW siphon headers supply water from Lake Keowee to the LPSW system suction piping and supports the safety function of the LPSW system. Additional information regarding the LPSW System can be found in the Bases of 3.7.7, "LPSW System."

Maintaining the ECCW siphon headers OPERABLE during accident and transient events is an assumption in the accident and transient analysis. The ESV System and SSW System are required to ensure ECCW siphon header piping remains sufficiently primed to supply siphon flow to the LPSW suction piping.

The ECCW siphon headers satisfy Criterion 3 of 10 CFR 50.36 (Ref. 2).

LCO

Two ECCW siphon headers are required to be OPERABLE during normal unit operation. An ECCW siphon header consists of a flow path from the intake canal through an open CCW pump discharge valve to the LPSW suction piping connection on the CCW crossover header. For an ECCW siphon header to be OPERABLE, an ESV pump must be OPERABLE, in operation, and aligned to the ECCW siphon header. Additionally, the ESV float valve must be OPERABLE. Heat tracing on the ESV float valve must be OPERABLE when the potential for freezing exists.

The ESV pump must be capable of restarting, after an appropriate time delay following restoration of emergency power after a loss of off-site power. This ensures air introduced by inleakage or degassing does not prevent siphon header function. Operation of an ESV pump requires a continuous seal water supply from the SSW System. The cross connection between ESV System headers on a Unit must be closed. Instrumentation necessary to provide indication of SSW flow to ESV is required to be OPERABLE to ensure continued availability following a design basis event.

BASES

LCO (continued)

Sharing of siphon headers between units is acceptable with certain restrictions. Net positive suction head requirements for Unit 1 and 2 LPSW pumps cannot be met with suction supplied from a Unit 3 siphon header. Therefore, the two ECCW siphon headers for Units 1 and 2 must be two of the four ECCW siphon headers associated with the Units 1 and 2 CCW piping. Units 1 and 2 may simultaneously share two of the four suction headers. Similarly, NPSH requirements for the Unit 3 LPSW pumps cannot be met with suction supplied from a Unit 1 ECCW siphon header. Therefore, the two ECCW siphon headers for Unit 3 must be two of the four siphon headers associated with Units 2 and 3 CCW piping. The Unit 2 ECCW siphon headers may be credited with supplying either the Unit 1 and 2 LPSW System or the Unit 3 LPSW System but not both LPSW Systems simultaneously. Both Unit 2 ECCW siphon headers must be credited to the same LPSW System (i.e., the two Unit 2 headers may not be split with one siphon header credited to the Unit 1 and 2 LPSW System and the other siphon header credited to the Unit 3 LPSW System). The two Unit 3 siphon headers may only be credited to Unit 3 (i.e., they cannot be credited to another unit).

The LCO is modified by a Note which indicates the requirements are not applicable to a Unit until after completion of the Service Water upgrade modifications on the respective Unit. This is necessary since the specification is based on the Unit's design after implementation of the modifications.

APPLICABILITY

In MODES 1, 2, 3, and 4, the ECCW siphon headers are normally operating to support the OPERABILITY of the equipment serviced by the ECCW siphon headers and are required to be OPERABLE in these MODES.

In MODES 5 and 6, the OPERABILITY requirements of the ECCW siphon headers are determined by the systems they support.

ACTIONS

A.1

If one required ECCW siphon header is inoperable, action must be taken to restore the inoperable ECCW siphon header to OPERABLE status within 72 hours.

BASES

ACTIONS

A.1 (continued)

In this Condition, the remaining ECCW siphon header is adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure in the OPERABLE ECCW siphon header could result in loss of ECCW system function. The 72 hour Completion Time is based on the redundant capabilities afforded by the OPERABLE ECCW siphon header, and the low probability of a accident occurring during this period.

B.1 and B.2

If the Required Action and associated Completion Time are not met, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 12 hours and in MODE 5 within 60 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.8.1

This SR requires verification every 12 hours that the required ESV pumps are in operation. Verification includes confirming appropriate vacuum tank pressure or pump status monitoring, which help ensure that ECCW siphon headers are maintained sufficiently primed. The 12 hour Frequency has been shown by operating practice to be sufficient to regularly assess degradation. In addition, control room indication normally indicate pump status and an alarm is provided for low vacuum tank vacuum.

SR 3.7.8.2

Verifying Keowee Lake level is within limit ensures ECCW siphons can provide sufficient flow to ensure adequate NPSH is available for operating the LPSW pumps. The 24 hour Frequency is based on operating experience related to the trending of the parameter variations during the applicable MODES. This SR verifies that the Keowee water level is ≥ limit specified in UFSAR Chapter 16. Lake level requirements are maintained in UFSAR Chapter 16 (Ref. 3) since the values are subject to change resulting from modifications and changes in operating practices, which may impact LPSW System flow requirements.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.8.3

This SR verifies that the average water temperature at the CCW inlet is ≤ 90°F. This SR verifies that CCW inlet temperature is consistent with assumptions in the safety analysis regarding inlet temperature for the LPSW system. The 24 hour Frequency is based on operating experience related to the trending of the parameter variations during the applicable MODES.

SR 3.7.8.4

Verifying the correct alignment for manual, and non-automatic power operated valves in the ECCW siphon header flow paths, required ESV flow paths and required SSW flow paths provides assurance that the proper flow paths exist for ECCW siphon header operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since those valves are verified to be in the correct position prior to locking, sealing, or securing. Additionally, this SR does not apply to automatic valves since these valves actuate to the correct position upon initiation. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves.

This Surveillance does not require any testing or valve manipulation; rather, it involves verification that those valves capable of potentially being mispositioned are in the correct position.

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

SR 3.7.8.5

Verification that ESV float valves open upon an actual or simulated actuation ensures a flow path is provided to the ESV pumps to assure the ECCW siphon headers are maintained sufficiently primed. The basis for the Frequency of 92 days is ASME Code, Section XI (Ref. 4).

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.8.6

Verification that required ESV valves actuate to the correct position ensures the ESV tank minimum flow valves will automatically close during a loss of offsite power event so that the full capacity of the ESV pumps will be aligned to the ECCW siphon headers. Verification that required SSW valves actuate to the correct position ensures sufficient seal water is provided to ESV pumps. The basis for the Frequency of 92 days is ASME Code, Section XI (Ref. 4).

SR 3.7.8.7

Verifying that each ESV pump's capacity at the test point is greater than or equal to the required capacity ensures that pump performance has not degraded below the acceptance criteria during the cycle. ESV pump capacity is determined by measuring the "apparent" flow rate and calculating the "corrected" flow rate by adjusting for air density changes between the measurement point and the pump inlet. The vacuum level must be within a prescribed range during this measurement to ensure that the flowmeter is on-scale and the pump operating liquid is not cavitating. Note that the pump is a constant volume machine. Thus, there is not a single test point but a range of acceptable vacuum levels. Although ASME code for inservice testing does not specifically address vacuum pumps, manufacturers test methods coupled with the ASME standard (OM-6) (Ref. 5) requirements for testing methodology are used as a guide for testing. Accordingly, the basis for the Frequency of 92 days is ASME Code, Section XI (Ref. 4).

SR 3.7.8.8

Verification that each required ESV pump automatically starts within 1200 seconds after an actual or simulated restoration of emergency power assures required ESV pumps will function after a loss of offsite power to maintain ECCW siphon headers sufficiently primed to maintain necessary flow to the suction of LPSW pumps. The Frequency of 18 months is based on engineering judgement.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.8.9

This SR verifies the ECCW system functions to supply siphon header flow to the suction of the LPSW pumps during design basis conditions by ensuring air accumulation in the ECCW siphon headers is within the removal capabilities of the ESV System. This SR establishes siphon flow with the ESV pumps off. Air accumulation in the pipe results in a corresponding reduction in water level in the CCW piping over a time period. The rate of water level reduction is recorded and compared to limits established in design basis documents. The limits on the rate of water level reduction over a time period are established to ensure ECCW siphon header air accumulation rate is within the removal capabilities of the ESV System under design basis conditions. The Frequency of 18 months is based on the need to perform this SR when the Unit is shutdown. This SR is not required to be performed with the Unit 3 LPSW System taking suction from the siphon. This is acceptable since aligning the LPSW pumps to the Unit 3 ECCW siphon headers is not necessary to demonstrate that the ECCW air accumulation is within the ESV capacity, which is the basic purpose of the test. The flow path from the Unit 3 CCW piping to the suction of the Unit 3 LPSW pumps is demonstrated by normal operation of the LPSW pumps.

A Note states that for Units 1 and 2, the SR is not required to be performed with the shared LPSW System for Units 1 and 2 taking suction from the siphon. This is necessary to avoid potential effects on an operating unit and is acceptable since the capability of the LPSW pumps to take suction from the CCW crossover header is demonstrated by normal, day-to-day operation of the LPSW pumps. Although a loss of suction to the LPSW pumps is unlikely during this SR, it is prudent to minimize the potential for jeopardizing the LPSW suction supply to the LPSW pumps when they are supporting an operating Unit.

REFERENCES

1. UFSAR, Chapter 9.
2. 10 CFR 50.36.
3. UFSAR, Chapter 16.
4. ASME, Boiler and Pressure Vessel Code, Section XI.
5. ASME Standard OM-6.

## B 3.7 PLANT SYSTEMS

### B 3.7.9 Control Room Ventilation System (CRVS) Booster Fans

BASES

**BACKGROUND**

The CRVS Booster Fan trains provide a protected environment from which operators can control the unit following an uncontrolled release of radioactivity, chemicals, or toxic gas.

Each CRVS Booster Fan train consist of a fan filter assembly, Booster Fans, Ducting, and Dampers. Each filter train consists of a pre-filter, a high efficiency particulate air (HEPA) filter, and a charcoal filter.

The CRVS Booster Fan trains are an emergency system. Upon receipt of a radiation alarm from the control room air radiation monitor, the CRVS Booster Fan trains can be started manually to minimize unfiltered air from entering the control room. Upon starting the fans, dampers are automatically positioned to isolate the control room. The pre-filters remove any large particles in the air, and any entrained water droplets present, to prevent excessive loading of the HEPA and carbon filters.

The two CRVS Booster Fan trains, when operated simultaneously, can pressurize the control room to minimize infiltration of unfiltered air. The CRVS operation is discussed in the UFSAR, Section 9.4 (Ref. 1).

**APPLICABLE SAFETY ANALYSES**

The CRVS Booster Fan train components are arranged in two ventilation trains. The location of components and ducting ensures an adequate supply of filtered air to all areas requiring access. The CRVS provides airborne radiological protection for the control room operators for the most limiting design basis loss of coolant accident fission product release presented in the UFSAR, Chapter 15 (Ref. 2).

The CRVS Booster Fan trains satisfy Criterion 3 of 10 CFR 50.36 (Ref. 3).

**LCO**

Two CRVS trains are required to be OPERABLE. Total system failure could result in excessive doses to the control room operators in the event of a large radioactive release.

BASES

LCO (continued)

The CRVS Booster Fan trains are considered OPERABLE when the individual components necessary to control operator exposure are OPERABLE in both trains. A CRVS Booster Fan train is considered OPERABLE when the associated:

a. Booster Fan is OPERABLE;

b. HEPA filter and carbon absorber are not excessively restricting flow, and are capable of performing their filtration functions; and

c. Ductwork, valves, and dampers are OPERABLE, and control room pressurization can be maintained with both trains operating.

In addition, the control room boundary, including the integrity of the walls, floors, ceilings, ductwork, and access doors, must be maintained within the assumptions of the design analysis.

APPLICABILITY

In MODES 1, 2, 3, and 4, the CRVS Booster Fan trains must be OPERABLE to reduce radiation dose to personnel in the control room during and following an accident.

ACTIONS

A.1

With the two CRVS Booster Fan trains incapable of pressurizing the control room, the capability to pressurize the control room must be restored within 30 days. In this Condition, the capability to minimize the radiation dose to personnel located in the control room during and after an accident is not assured. One or both CRVS Booster Fan trains may be OPERABLE in this Condition. If one or both CRVS Booster Fans are simultaneously inoperable, the Completion Time for these separate Conditions is more limiting than the 30 day Completion Time for Action A.1. If OPERABLE the CRVS Booster Fan train(s) can provide some dose reduction. The 30 day Completion Time is based on the low probability of an accident occurring during the time period and the potential for OPERABLE CRVS Booster Fan trains to provide some dose reduction.

BASES

ACTIONS (continued)

B.1

With one CRVS Booster Fan train inoperable for reasons other than Condition A, action must be taken to restore the train to OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE CRVS Booster Fan train provides some dose reduction for personnel in the control room. The 72 hour Completion Time is based on the low probability of an accident occurring during this time period, and ability of the remaining train to provide some dose reduction.

C.1

With the two CRVS Booster Fan trains inoperable for reasons other than Condition A, one train must be restored to OPERABLE status within 24 hours. In this Condition, the capability to minimize the radiation dose to personnel located in the control room during and after an accident is unavailable. The 24 hour Completion Time is based on the low probability of an accident occurring during this time period.

D.1

If the inoperable CRVS Booster Fan trains cannot be restored to OPERABLE status within the required Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 12 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.9.1

Standby systems should be checked periodically to ensure that they function properly. As the environment and normal operating conditions on this system are not severe, testing each train once every 92 days adequately checks this system. The trains need only be operated for ≥ one hour and all louvers verified to be OPERABLE to demonstrate the function of the system. This test includes an external visual inspection of the CRVS Booster Fan trains. The 92 day Frequency is based on the known reliability of the equipment.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.9.2

This SR verifies that the required CRVS Booster Fan train testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The CRVS Booster Fan train filter test frequencies are in accordance with Regulatory Guide 1.52 (Ref. 4). The VFTP includes testing HEPA filter performance and carbon adsorber efficiency. Specific test frequencies and additional information are discussed in detail in the VFTP.

SR 3.7.9.3

This SR verifies the integrity of the control room enclosure. The control room positive pressure, with respect to potentially contaminated adjacent areas, is periodically tested to verify that the CRVS Booster Fan trains are functioning properly. During the emergency mode of operation, the CRVS Booster Fan trains are designed to pressurize the control room to minimize unfiltered inleakage. The CRVS Booster Fan trains are designed to maintain this positive pressure with both trains in operation. The Frequency of 18 months is consistent with industry practice.

REFERENCES

1. UFSAR, Section 9.4.
2. UFSAR, Chapter 15.
3. 10 CFR 50.36.
4. Regulatory Guide 1.52.

## B 3.7 PLANT SYSTEMS

## B 3.7.10 Penetration Room Ventilation System (PRVS)

BASES

BACKGROUND

The PRVS filters air from the area of the active penetration rooms during the recirculation phase of a loss of coolant accident (LOCA).

The PRVS consists of two independent, redundant trains. Each train consists of a prefilter, a high efficiency particulate air (HEPA) filter, an activated carbon adsorber section for removal of gaseous activity (principally iodines), and a fan. Ductwork, valves or dampers, and instrumentation also form part of the system. The system initiates filtered ventilation of the Reactor Building penetration rooms area following receipt of an Engineered Safeguards actuation signal (ESAS).

The PRVS is a standby system. During emergency operations, the PRVS valves are realigned, and fans are started to begin filtration. Upon receipt of the ESAS signal(s), the stream of ventilation air discharges through the system filter trains. The prefilters remove any large particles in the air, and any entrained water droplets present, to prevent excessive loading of the HEPA filters and carbon adsorbers.

The PRVS is discussed in the UFSAR, Sections 6.5.1, 9.4.7, and 15.4.7 (Refs. 1, 2, and 3, respectively).

APPLICABLE SAFETY ANALYSES

The design basis of the PRVS is established by the Maximum Hypothetical Accident (MHA). In such a case, the system limits radioactive releases to within 10 CFR 100 (Ref. 7) requirements and personnel doses in the Control Room are maintained within the limits of 10 CFR 20 (Ref. 4). The analysis of the effects and consequences of an MHA is presented in Reference 3. No credit is taken in the analysis for any reduction in Control Room Dose provided by the PRVS filters.

The PRVS also actuates following a large and small break LOCA, in those cases where the unit goes into the recirculation mode of long term cooling, and to cleanup releases of smaller leaks, such as from valve stem packing.

Following a LOCA, an ESAS starts the PRVS fans and opens the dampers located in the penetration room outlet ductwork.

The PRVS satisfies Criterion 3 of 10 CFR 50.36 (Ref. 5).

BASES (continued)

LCO

Two independent and redundant trains of the PRVS are required to be OPERABLE to ensure that at least one is available, assuming that a single failure disables the other train coincident with loss of offsite power.

The PRVS is considered OPERABLE when the individual components necessary to maintain the penetration room filtration are OPERABLE in both trains.

A PRVS train is considered OPERABLE when its associated:

a. Fan is OPERABLE;

b. HEPA filter and carbon adsorber are not excessively restricting flow, and are capable of performing their filtration functions; and

c. Ductwork, valves, and dampers are OPERABLE, and air flow can be maintained.

In addition, the penetration room boundaries, including the integrity of the walls, floors, ceilings, ductwork, and access doors, must be maintained within the assumptions of the design analysis.

APPLICABILITY

In MODES 1, 2, 3, and 4, the PRVS is required to be OPERABLE consistent with the OPERABILITY requirements of the containment.

In MODES 5 and 6, the PRVS is not required to be OPERABLE since the containment is not required to be OPERABLE.

ACTIONS

A.1

With one PRVS train inoperable, action must be taken to restore the PRVS train to OPERABLE status within 7 days. During this time, the remaining OPERABLE train is adequate to perform the PRVS safety function. However, the overall reliability is reduced because a single failure in the OPERABLE PRVS train could result in loss of PRVS function.

The 7 day Completion Time is appropriate because the risk contribution is less than that of the ECCS (72 hour Completion Time), and this system is not a direct support system for the ECCS. The 7 day Completion Time is based on the low probability of an accident occurring during this time period, and ability of the remaining train to provide the required capability.

BASES

ACTIONS (continued)

B.1 and B.2

If the required Action and associated Completion Time are not met, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 12 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.10.1

Standby systems should be checked periodically to ensure that they function properly. Since the environment and normal operating conditions on this system are not severe, testing each train once a month provides an adequate check on this system. The 31 day Frequency is based on known reliability of equipment and the two train redundancy available.

SR 3.7.10.2

This SR verifies that the required PRVS testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The VFTP includes testing HEPA filter performance and carbon adsorber efficiency. Specific test frequencies and additional information are discussed in detail in the VFTP.

SR 3.7.10.3

This SR verifies that each PRVS train starts and operates on an actual or simulated actuation signal. The 18 month Frequency is consistent with the guidance in Reference 6.

SR 3.7.10.4

This SR verifies the integrity of the penetration rooms area. The ability of the PRVS to maintain a negative pressure, with respect to outside atmosphere, is periodically tested to verify proper functioning of the PRVS. During the post accident mode of operation, the PRVS is

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.10.4 (continued)

designed to maintain a slight negative pressure in the penetration rooms with respect to outside atmosphere to prevent unfiltered LEAKAGE. The PRVS is designed to maintain this negative pressure at a flow rate of 1000 ± 10% cfm from the area. The Frequency of 18 months on a STAGGERED TEST BASIS is consistent with industry practice and other filtration SRs.

SR 3.7.10.5

Operating the PRVS filter bypass valve is necessary to ensure that the system functions properly. The OPERABILITY of the PRVS filter bypass valve is verified if it can be opened. An 18 month Frequency is consistent with the guidance in Reference 6.

REFERENCES

1. UFSAR, Section 6.5.1.
2. UFSAR, Section 9.4.7.
3. UFSAR, Section 15.15.
4. 10 CFR 20.
5. 10 CFR 50.36.
6. Regulatory Guide 1.52.
7. 10 CFR 100.

## B 3.7 PLANT SYSTEMS

### B 3.7.11 Spent Fuel Pool Water Level

BASES

BACKGROUND

The minimum water level in the Spent Fuel Pool is consistent with the assumption of iodine decontamination factors following a fuel handling or cask drop accident. The specified water level shields and minimizes the general area dose when the storage racks are filled to their maximum capacity. The water also provides shielding during the movement of spent fuel.

A general description of the Spent Fuel Pool design is given in the UFSAR, Section 9.1.2, Reference 1. The Spent Fuel Pool Cooling and Cleanup System is given in the UFSAR, Section 9.1.3 (Ref. 2). The assumptions of the fuel handling accident or cask drop are given in the UFSAR, Section 15.11.2 (Ref. 3).

APPLICABLE SAFETY ANALYSES

During movement of irradiated fuel assemblies or crane operations with loads in the Spent Fuel Pool, the water level in the pool is an initial condition design parameter in the analysis of the fuel handling accident and cask drop accidents in the fuel pool. A minimum water level of 23 ft (Regulatory Position C.1.c of Ref. 4) allows a decontamination factor (DF) of 100 (Regulatory Position C.1.g of Ref. 4) to be used in the accident analysis for iodine. This relates to the assumption that 99% of the total iodine released from the pellet to cladding gap of all the damaged fuel assembly(ies) rods is retained by the Spent Fuel Pool water. The fuel pellet to cladding gap is assumed to contain 10% of the total fuel rod iodine inventory (Ref. 4).

The fuel handling accident and cask drop accident analysis in the Spent Fuel Pool is described in Reference 3. Since the minimum water level of 21.34 feet is less than 23 feet, the assumed iodine DF must be less than 100, according to Ref. 4, and calculated with comparable conservatism. An experimental test program described in WCAP-7828 (Ref. 6) evaluated the extent of removal of iodine released from a damaged irradiated fuel assembly. Using the analytical results from the test program described in WCAP-7828, with a water depth of 21.34 feet, a comparable DF of 89 was determined. With a minimum water level of

BASES

APPLICABLE SAFETY ANALYSES (continued)

21.34 ft, and a minimum decay time of 72 hours prior to fuel handling, the analysis and test programs demonstrate that the iodine release due to a postulated fuel handling or cask drop accident is adequately captured by the water, and offsite doses are maintained within allowable limits (Ref. 7).

The Spent Fuel Pool water level satisfies Criterion 2 and 3 of 10 CFR 50.36 (Ref. 7).

LCO

The specified water level preserves the assumptions of the fuel handling and cask drop accident analyses (Ref. 3). As such, it is the minimum required for fuel storage and movement within the Spent Fuel Pool or movement of the cask over the Spent Fuel Pool.

APPLICABILITY

This LCO applies during movement of irradiated fuel assemblies in the Spent Fuel Pool or movement of the cask over the Spent Fuel Pool since the potential for a release of fission products exists.

ACTIONS

Required Actions A.1 and A.2 are modified by a Note indicating that LCO 3.0.3 does not apply.

If moving irradiated fuel assemblies or a cask while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies or a cask while in MODES 1, 2, 3, and 4, the fuel or cask movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of irradiated fuel assemblies or a cask is not sufficient reason to require a reactor shutdown.

A.1

When the initial conditions for an accident cannot be met, immediate action must be taken to preclude the occurrence of an accident. With the Spent Fuel Pool at less than the required level, the movement of fuel assemblies in the Spent Fuel Pool is immediately suspended. This effectively precludes the occurrence of a fuel handling accident. In such a case, unit procedures control the movement of other (non cask) loads over the spent fuel. This does not preclude movement of a fuel assembly to a safe position.

BASES (continued)

APPLICABILITY

This LCO applies whenever fuel assemblies are stored in the spent fuel pool.

ACTIONS

A.1 and A.2

The Required Actions are modified by a Note indicating that LCO 3.0.3 does not apply.

If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, inability to suspend movement of fuel assemblies is not a sufficient reason to require a reactor shutdown.

When the concentration of boron in the fuel storage pool is less than required, immediate action must be taken to preclude the occurrence of an accident or to mitigate the consequences of an accident in progress. This is achieved by immediately suspending the movement of the fuel assemblies. This does not preclude movement of a fuel assembly to a safe position. Immediate action is also required to initiate action to restore the SFP boron concentration to within limits.

SURVEILLANCE REQUIREMENTS

SR 3.7.12.1

This SR verifies that the concentration of boron in the fuel storage pool is within the required limit. As long as this SR is met, the analyzed incidents are fully addressed. The 31 day Frequency is appropriate because no major replenishment of pool water is expected to take place over a short period of time and verification is required after each makeup to the SFP. The verification after each makeup should be completed within 12 hours after a 24 hour recirculation period to allow for mixing. This Completion Time is appropriate since no major replenishment of pool water is expected to take place over this period.

REFERENCES

1. ANSI N-16.1-1975.

2. Letter from B.K. Grimes (USNRC) to Power Reactor Licensees dated April 14, 1978.

3. 10 CFR 50.36.

## B 3.7 PLANT SYSTEMS

### B 3.7.13 Fuel Assembly Storage

BASES

BACKGROUND

The spent fuel pool is designed to store either new (nonirradiated) nuclear fuel assemblies, or burned (irradiated) fuel assemblies in a vertical configuration underwater. The shared spent fuel pool between Unit 1 and Unit 2 is sized to store 1312 fuel assemblies. The Unit 1 and Unit 2 spent fuel storage cells are installed in parallel rows with center to center spacing of 10.65 inches. The Unit 3 storage pool is sized to store 822 fuel assemblies. The spent fuel storage cells are installed in parallel rows with center to center spacing of 10.60 inches. This spacing and construction, whereby the fuel assemblies are inserted into stainless steel cans with neutron absorbing Boraflex attached, is sufficient to maintain a $k_{eff}$ of $\leq 0.95$ for spent fuel of a maximum nominal initial enrichment of up to 5.0 wt % which have accumulated burnups $\geq$ the minimum qualifying burnups of Figure 3.7.13-1 for the spent fuel pool shared by Units 1 and 2 or Figure 3.7.13-2 for the Unit 3 spent fuel pool. Fuel which has not accumulated the minimum qualifying burnups is required to be stored in the specified pattern for restricted fuel.

APPLICABLE SAFETY ANALYSES

The spent fuel pool is designed for noncriticality by use of adequate spacing, and "flux trap" construction whereby the fuel assemblies are inserted into stainless steel cans with neutron absorbing Boraflex attached.

The fuel assembly storage satisfies Criterion 2 of 10 CFR 50.36 (Ref. 1).

LCO

The restrictions on the placement of fuel assemblies within the fuel pool, according to the Figures in the accompanying LCO, ensure that the $k_{eff}$ of the spent fuel pool will always remain $\leq 0.95$ assuming the pool to be flooded with unborated water. The restrictions are consistent with the criticality safety analysis performed for the spent fuel pool.

APPLICABILITY

This LCO applies whenever any fuel assembly is stored in the spent fuel pool.

BASES (continued)

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply.

If moving fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, in either case, inability to move fuel assemblies is not sufficient reason to require a reactor shutdown.

When the configuration of fuel assemblies stored in the spent fuel pool is not in accordance with the LCO, immediate action must be taken to make the necessary fuel assembly movement(s) to bring the configuration into compliance with the LCO.

SURVEILLANCE REQUIREMENTS

SR 3.7.13.1

This SR verifies by administrative means that the initial enrichment and burnup of the fuel assembly is in accordance with the appropriate Figure in the accompanying LCO.

REFERENCES

1. 10 CFR 50.36.

## B 3.7 PLANT SYSTEMS

### B 3.7.14 Secondary Specific Activity

BASES

BACKGROUND

Activity in the secondary coolant results from steam generator tube out-LEAKAGE from the Reactor Coolant System (RCS). Under steady state conditions, the activity is primarily iodines with relatively short half lives and, thus, indicative of current conditions. During transients, I-131 spikes have been observed, as well as increased releases of some noble gases. Other fission product isotopes, as well as activated corrosion products, in lesser amounts, may also be found in the secondary coolant.

A limit on secondary coolant specific activity during power operation minimizes releases to the environment because of normal operation, anticipated transients and accidents.

This limit bounds the activity value that might be expected from 300 gallon per day tube leakage (LCO 3.4.13, "RCS Operational Leakage") of primary coolant at the limit of 1.0 μCi/gm (LCO 3.4.11, "RCS Specific Activity"). The steam line failure, steam generator tube rupture and other design basis accidents or transients can result in the release of the noble gas and iodine activity contained in the steam generator inventory, the feedwater, and the reactor coolant leakage.

Operating a unit at the allowable limits could result in a 2 hour EAB exposure that is within the 10 CFR 100 (Ref. 1) limits.

APPLICABLE SAFETY ANALYSES

The accident analyses of the steam generator tube rupture, rod ejection and main steam line break, as discussed in the UFSAR, Chapter 15 (Ref. 2) assume the initial secondary coolant specific activity to have a radioactive isotope concentration of 0.10 μCi/gm DOSE EQUIVALENT I-131. This assumption is used in the analysis for determining the radiological consequences of the postulated accident. The accident analysis, based on this and other assumptions, shows that the radiological consequences of the plant's design basis accidents and transients do not exceed established limits, (Ref. 1) for whole body and thyroid dose rates.

Secondary specific activity limits satisfy Criterion 2 of 10 CFR 50.36 (Ref. 3).

BASES (continued)

LCO

As indicated in the Applicable Safety Analyses, the specific activity limit in the secondary coolant system of ≤ 0.10 μCi/gm DOSE EQUIVALENT I-131 maintains the radiological consequences of an accident to within Reference 1 limits.

Monitoring the specific activity of the secondary coolant ensures that, when secondary specific activity limits are exceeded, appropriate actions are taken, in a timely manner, to place the unit in an operational MODE that would minimize the radiological consequences of an accident.

APPLICABILITY

In MODES 1, 2, 3, and 4, the limits on secondary specific activity apply due to the potential for secondary steam releases to the atmosphere.

In MODES 5 and 6, the steam generators are not being used for heat removal. Both the RCS and steam generators are at low pressure and primary to secondary LEAKAGE is minimal. Therefore, secondary specific activity is not a concern.

ACTIONS

A.1 and A.2

DOSE EQUIVALENT I-131 exceeding the allowable value in the secondary coolant contributes to increased post accident doses. If secondary specific activity cannot be restored to within limits within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 12 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.14.1

This SR verifies that the secondary specific activity is within the limits of the accident analysis assumptions. A gamma isotopic analysis of the secondary coolant, which determines DOSE EQUIVALENT I-131, confirms the assumptions of Reference 1 are met. It also serves to identify and trend any unusual isotopic concentrations that might indicate changes in reactor coolant activity or LEAKAGE. The 31 day Frequency is based on the detection of increasing trends of the level of DOSE EQUIVALENT I-131, and allows for appropriate action to be taken to maintain levels below the LCO limit.

BASES (continued)

REFERENCES

1. 10 CFR 100.11.

2. UFSAR, Chapter 15.

3. 10 CFR 50.36.

## B 3.7 PLANT SYSTEMS

## B 3.7.15 Decay Time for Fuel Assemblies in Spent Fuel Pool (SFP)

BASES

BACKGROUND

Spent fuel shipping casks are used to transport irradiated fuel assemblies (FA) from the site and also between the Oconee 1 and 2 spent fuel pool and the Oconee 3 spent fuel pool. Dry storage transfer operations from the spent fuel pool (SFP) buildings to the Independent Spent Fuel Storage Facility (ISFSI) are routinely performed using dry storage transfer casks. Use of these casks requires placing them in the spent fuel pools. Movement of these casks in the spent fuel area creates a potential for a cask to fall into the spent fuel pool, damaging stored fuel assemblies.

APPLICABLE SAFETY ANALYSES

Two hypothetical accident scenarios, the drop of a spent fuel shipping cask and the drop of a dry storage transfer cask onto the irradiated assemblies in the storage racks in the spent fuel pools are considered. The analysis of cask drop accidents in the SFP are presented in UFSAR Section 15.11 (Ref. 1).

The fuel handling accident sequence in which the spent fuel shipping cask impacts on the irradiated fuel assemblies in a spent fuel pool has been evaluated. The analysis has been performed separately for the shared Unit 1 and 2 spent fuel pool and the Unit 3 spent fuel pool. Appropriate conservative assumptions have been employed for determining the number of fuel assemblies damaged. The gap fractions and decontamination factors used are consistent with the guidance in Regulatory Guide 1.25 (Ref. 2).

Based upon the number of fuel assemblies which could be damaged, dose analyses were performed which are consistent with Regulatory Guide 1.25, and NUREG-0612 (Ref. 3). The radiological consequences resulting from a spent fuel shipping cask accident in either the Unit 1 and 2 SFP or the Unit 3 SFP were analyzed including assumptions regarding irradiated fuel decay time and associated storage location. Irradiated fuel stored in the first 36 rows of the Unit 1 and 2 spent fuel pool closest to the spent fuel cask handling area are assumed to have been decayed at least 55 days. Fuel assemblies assumed damaged in excess of two full cores (354 assemblies) in the unit 1 and 2 SFP are assumed to have decayed at least

BASES

APPLICABLE SAFETY ANALYSES (continued)

one year. Irradiated fuel assemblies stored in the first 33 rows of the Unit 3 spent fuel pool closest to the spent fuel cask handling area are assumed to have decayed at least 70 days. Fuel assemblies assumed damaged in excess of one full core (177 assemblies) in the Unit 3 SFP are assumed to have decayed at least one year. The radiological consequences resulting from a dry storage transfer cask drop accident into either the Unit 1 and 2 or the Unit 3 SFP were analyzed. The analysis has been performed separately for the shared Unit 1 and 2 spent fuel pool and the Unit 3 spent fuel pool. The analysis for the Unit 1 and 2 SFP assumes that fuel stored in the first 64 rows closest to the cask handling area have been decayed a minimum of 65 days. Likewise, all irradiated FAs stored in the Unit 3 pool are assumed to have been decayed a minimum of 57 days. These decay time assumptions, in addition to other assumptions consistent with Regulatory Guide 1.25 were used to determine the curies of each nuclide released from the postulated dry storage transfer cask drop accident. The total activity releases for each pool were used to determine the corresponding offsite dose consequences.

The offsite radiological consequences of the postulated cask drop accidents are within the 10 CFR 100 guidelines.

Decay Time for Fuel Assemblies in Spent Fuel Pool (SFP) satisfies Criterion 2 of 10 CFR 50.36 (Ref. 4).

LCO

The LCO requires that irradiated fuel assemblies in the specified storage locations be decayed at least as long as the minimum time specified. The specified decay time limits are dependent upon the combination of the specific cask being used and the SFP area in which the cask movement is taking place.

APPLICABILITY

The LCO applies during movement of either the spent fuel shipping cask or the dry storage transfer cask in the SFP area. When a cask is not being moved in the SFP area, the potential for a cask drop accident does not exist. The SFP area includes the area immediately surrounding the SFP itself but does not include the truck bay area since its elevation is well below the SFP area.

BASES (continued)

ACTIONS

A.1

When the requirements of the LCO are not met, immediate action must be taken to preclude the occurrence of an accident.

This is most efficiently achieved by immediately suspending the movement of the cask associated with the LCO which is not met. This does not preclude movement of the cask to a safe position.

The Required Action is modified by a Note indicating that LCO 3.0.3 does not apply. If moving the cask while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving the cask while in MODE 1, 2, 3, or 4, the cask movement is independent of reactor operation. Therefore, inability to suspend movement of cask is not a sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.7.15.1

This SR verifies by administrative means that the decay time of the fuel assemblies are in accordance with the accompanying LCO.

REFERENCES

1. UFSAR, Section 15.11
2. Regulatory Guide 1.25
3. NUREG-0612
4. 10 CFR 50.36

## B 3.7 PLANT SYSTEMS

## B 3.7.16 Control Room Area Cooling Systems (CRACS)

BASES

BACKGROUND

The CRACS provides temperature control for the control areas.

The control area is defined as the control room, cable room, and equipment room for each unit. Units 1 and 2 have a shared control room, and Unit 3 has an independent control room. The cable and equipment rooms are independent for each unit. The control rooms, cable rooms, and equipment rooms for each unit contain vital electrical equipment, such as 125 VDC Vital I&C Power and 120 VAC Vital I&C Power, which is essential for achieving safe shutdown on the units. A control area portion is defined as a cable room, equipment room, or control room, for which a set of redundant CRVS cooling trains is required. The control area portions are listed in the table below. Through the use of alternative air flow paths, air handling units AHU-34 and AHU-35 provide redundant cooling to both Units 1 and 2 cable rooms.

The AHUs which cool the control areas are part of the CRVS for each unit. The Chilled Water System (WC) serves as the heat sink for the CRVS on all three units. The WC System consists of two redundant cooling trains which serve all three units.

UFSAR Section 3.11.4 (Ref. 1) requires that redundant air conditioning and ventilation equipment be available to assure that no single failure of an active component within the CRVS and WC System will prevent proper control area environmental control. During a LOOP event, power will be temporarily lost to the equipment within these systems. Upon restoration of power the equipment will be required to restart. This restart makes the equipment susceptible to a single active failure. Without redundant cooling capability, acceptable temperatures within the control area could be exceeded. This could result in the potential failure of vital electrical equipment which is needed for safe shutdown of the units.

BASES

BACKGROUND (continued)

The following table identifies each portion of the CRVS where redundancy is required:

Table B 3.7.16-1
CRVS Redundant Equipment

| Control Area Portion | Associated CRVS Cooling Trains |
|---|---|
| Unit 1&2 Control Room | AHU-11 and AHU-12 |
| Unit 1 Cable Room | AHU-34 and AHU-35 |
| Unit 1 Equipment Room | AHU-22 and AHU-34 |
| Unit 2 Cable Room | AHU-34 and AHU-35 |
| Unit 2 Equipment Room | AHU-23 and AHU-35 |
| Unit 3 Control Room | AHUs 3-13 and 3-14 |
| Unit 3 Cable Room | AHUs 3-11 and 3-12 |
| Unit 3 Equipment Room | AHUs 3-15 and 3-16 |

A single train will provide the required temperature control. The CRACS operation to maintain control room temperature is discussed in the UFSAR, Section 9.4.1 (Ref. 2).

APPLICABLE SAFETY ANALYSES

The design basis of the CRACS is to maintain control area temperature to ensure cooling of vital equipment.

The CRACS components are arranged in redundant trains. A single active failure of a CRACS component does not impair the ability of the system to perform as designed. The CRACS is designed to remove sensible and latent heat loads from the control area, including consideration of equipment heat loads to ensure equipment OPERABILITY.

The CRACS satisfies Criterion 3 of the NRC Policy Statement.

BASES (continued)

LCO

Two redundant trains of the CRACS and WC Systems train are required to be OPERABLE to ensure that at least one train in each system is available, assuming a single active failure disables the other train in one or both systems. Total system failure could result in the equipment operating temperature exceeding limits. A Train of CRVS consists of one of the redundant AHUs specified in Table B 3.7.16-1 for each of the three portions of the control area for an Oconee unit and associated ducts, dampers, instrumentation and controls. A single AHU can function as a component in more than one train on an Oconee unit and can function as a component on trains in multiple Oconee units. For example AHU-34, and its associated ducts, damper, instrumentation and controls, can simultaneously function as the AHU for a train of CRVS serving the Unit 1 cable room, the Unit 1 equipment room as well as the Unit 2 cable room. The combination of AHU-34 and either AHU-11 or AHU-12 along with their associated equipment constitutes a combination of equipment which can satisfy the requirement for one train of CRVS for Unit 1. Additionally, AHU-34 can simultaneously serve as the AHU for the portion of a Unit 2 CRVS train serving the Unit 2 cable room. AHU-35 in combination with either AHU-11 or AHU-12 along with their associated equipment constitutes a combination of equipment which can satisfy the requirement for one train of CRVS for Unit 2.

The CRACS is considered OPERABLE when the individual components that are necessary to maintain control area temperature are OPERABLE in both trains of CRVS and WC System. These components include the cooling coils, water cooled condensing units, and associated temperature control instrumentation. The two redundant trains can include a temporarily installed full-capacity control area cooling train. Any temporary cooling train shall have a power source with availability equivalent to the source of the permanently installed train. In addition, the CRACS must be OPERABLE to the extent that air circulation can be maintained.

APPLICABILITY

In MODES 1, 2, 3, and 4 the CRACS must be OPERABLE to ensure that the control area temperature will not exceed equipment OPERABILITY requirements.

BASES (continued)

ACTIONS

A.1

With one CRVS train inoperable for the control area, action must be taken to restore the CRVS train to OPERABLE status within 30 days. In this Condition, the remaining OPERABLE CRVS train is adequate to maintain the control area temperature within limits. However, the overall reliability is reduced because a failure in the OPERABLE CRVS train could result in a loss of CRVS cooling function. The 30 day Completion Time is based on the low probability of a loss of CRVS cooling component and the time necessary to perform repairs to CRVS cooling equipment.

B.1

With one WC train inoperable for a control area portion, action must be taken to restore the WC train to OPERABLE status within 30 days. In this Condition, the remaining OPERABLE WC train is adequate to maintain the control area portion temperature within limits. However, the overall reliability is reduced because a failure in the OPERABLE WC train could result in a loss of CRACS cooling function. The 30 day Completion Time is based on the low probability of a loss of WC cooling component, and on the time necessary to perform repairs to WC cooling equipment.

C.1

With the control room area air temperature outside its limit, action must be taken to restore the air temperature to within the limit within 7 days. If the control room area air temperature exceeds its limit, the ability of a single train of CRACS to maintain control room area temperature may be affected. The Completion Time of 7 days is reasonable considering the remaining CRACS train available to perform the required temperature control function and the low probability of an event occurring that would require the CRACS operation during that time.

The Required Actions are modified by a Note that states LCO 3.0.4 is not applicable. In consideration of the redundant CRACS train available, the small variation in temperature expected between 12 hour surveillances, and the marginal impact small temperature variations may have on the ability of a CRACS train to maintain the control room temperature within limits, an exception to LCO 3.0.4 is applicable for this condition.

BASES

ACTIONS (continued)

D.1 and D.2

If the Required Actions and associated Completion Times of Conditions A, B, or C are not met, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 12 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner without challenging unit systems.

E.1

If both CRVS trains or both WC trains are inoperable, the CRACS may not be capable of performing the intended function and the unit is in a condition outside the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.7.16.1

This SR verifies that the heat removal capability of the system is sufficient to maintain the temperature in the control room and cable room at or below 80°F and maintain the temperature in the electrical equipment room at or below 85°F. The temperature is determined by reading gauges in each area which are considered representative of the average area temperature. These temperature limits are based on operating history and are intended to provide an indication of degradation of the cooling systems. The limits are conservative with respect to equipment operability temperature limits. The values for the SR are values at which the system is removing sufficient heat to meet design requirements (i.e., OPERABLE) and sufficiently above the values associated with normal operation during hot weather. The temperature in the equipment room is typically slightly higher than the temperature in the control room or cable room. Because of that, a higher value is specified for this area. The 12 hour Frequency is appropriate since significant degradation of the CRACS is slow and is not expected over this time period.

REFERENCES

1. UFSAR, Section 3.11.4.

2. UFSAR, Section 9.4.1.

B 3.7 PLANT SYSTEMS

B 3.7.17 Spent Fuel Pool Ventilation System (SFPVS)

BASES

BACKGROUND

Ventilation air for the Spent Fuel Pool Area is supplied by an air handling unit which consists of roughing filters, steam heating coil, cooling coil supplied by low pressure service water, and a centrifugal fan. In the normal mode of operation, the air from the Spent Fuel Pool Area is exhausted directly to the unit vents by the general Auxiliary Building exhaust fans. The filtered exhaust system consists of a single filter train and two 100 percent capacity vane axial fans. The filter train utilized is the Reactor Building Purge Filter Train. The Unit 2 Reactor Building purge filter train is used for the combined Unit 1 and 2 Spent Fuel Pool Ventilation System, The Unit 3 Reactor Building purge filter train is used for the Unit 3 SFP Ventilation System. The filter train is comprised of prefilters, HEPA filters, and charcoal filters. To control the direction of air flow, i.e., to direct the air from the Fuel Pool Area to the Reactor Building Purge Filter Train, a series of pneumatic motor operated dampers are provided along with a crossover duct from the Fuel Pool to the filter train.

The SFPVS is discussed in the UFSAR, Section 9.4.2, (Ref. 1).

APPLICABLE SAFETY ANALYSES

The analysis of the limiting fuel handling accident, the cask drop accident, given in Reference 2, assumes that a certain number of fuel assemblies are damaged. The DBA analysis for the cask drop accident, does not assume operation of the SFPVS. These assumptions and the analysis are consistent with the guidance provided in Regulatory Guide 1.25 (Ref. 3).

The SFPVS does not satisfy the criteria in 10 CFR 50.36

LCO

Two redundant trains of the SFPVS are required to be OPERABLE to ensure that at least one is available, assuming a single failure that disables the other train.

BASES

LCO (continued)

An SFPVS train is considered OPERABLE when its associated:

1. Fan is OPERABLE;

2. HEPA filter and charcoal adsorber are not excessively restricting flow, and are capable of performing their filtration functions; and

3. Ductwork and dampers are OPERABLE, and air flow can be maintained.

The LCO is modified by two Notes. Note 1 states LCO 3.0.3 does not apply. If moving fuel or conducting crane operations with load over the storage pool while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving fuel or conducting crane operations with load over the storage pool while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, inability to suspend movement of fuel assemblies is not a sufficient reason to require a reactor shutdown. Note 2 states the requirements of this LCO is not applicable during reracking operations with no fuel in the spent fuel pool. With no fuel in the spent fuel pool, the potential release of radioactive material to the environs resulting from crane operations with load over the storage pool is substantially reduced.

APPLICABILITY

During movement of fuel in the fuel handling area or during crane operations with loads over the spent fuel pool, the SFPVS is always required to be OPERABLE.

ACTIONS

A.1 and A.2

With one SFPVS train inoperable, the OPERABLE SFPVS train must be started immediately with its discharge through the associated reactor building purge filter or fuel movement in the spent fuel pool and crane operations with loads over the spent fuel pool suspended. This action ensures that the remaining train is OPERABLE, and that any active failures will be readily detected.

If the system is not placed in operation, this action requires suspension of fuel movement and suspension of crane operation with loads over the spent fuel pool, which precludes a fuel handling accident. This action does not preclude the movement of fuel assemblies or crane loads to a safe position.

BASES

ACTIONS (continued)

B.1

When two trains of the SFPVS are inoperable during movement of fuel in the spent fuel pool, the unit must be placed in a condition in which the LCO does not apply. This Action involves immediately suspending movement of fuel assemblies in the spent fuel pool and suspension of crane operations with loads over the spent fuel pool. This does not preclude the movement of fuel or crane loads to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.7.17.1

Standby systems should be checked periodically to ensure that they function properly. As the environment and normal operating conditions on this system are not severe, testing each train once every month provides an adequate check on this system. Systems without heaters need only be operated through the associated reactor building purge filters at a design flow ± 10% for ≥ 15 minutes to demonstrate the function of the system. The 31 day Frequency is based on the known reliability of the equipment and the two train redundancy.

SR 3.7.17.2

This SR verifies that the required SFPVS testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The VFTP includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the VFTP.

REFERENCES

1. UFSAR, Section 9.4.2.

2. UFSAR, Section 15.11.

3. Regulatory Guide 1.25.

# B 3.8 ELECTRICAL POWER SYSTEMS

## B 3.8.1 AC Sources – Operating

BASES

BACKGROUND

The AC Power System consists of the offsite power sources (preferred power) and the onsite standby power sources, Keowee Hydro Units (KHU). This system is designed to supply the required Engineered Safeguards (ES) loads of one unit and safe shutdown loads of the other two units and is so arranged that no single failure can disable enough loads to jeopardize plant safety. The design of the AC Power System provides independence and redundancy to ensure an available source of power to the ES systems (Ref. 1). The KHU turbine generators are powered through a common penstock by water taken from Lake Keowee. The use of a common penstock is justified on the basis of past hydro plant experience of the licensee (since 1919) which indicates that the cumulative need to dewater the penstock can be expected to be limited to about one day a year, principally for inspection, plus perhaps four days every tenth year.

The preferred power source is provided from offsite power to the red or yellow bus in the 230 kV switchyard to the units startup transformer and the E breakers. The 230 kV switchyard is electrically connected to the 525 kV switchyard via the autobank transformer. Emergency power is provided using two emergency power paths, an overhead path and an underground path. The underground emergency power path is from one KHU through the underground feeder circuit, transformer CT-4, the CT-4 incoming breakers (SK breakers), standby bus and the standby breakers (S breakers). The standby buses may also receive offsite power from the 100 kV transmission system through transformer CT-5 and the CT-5 incoming breakers (SL breakers). The overhead emergency power path is from the other KHU through the startup transformer and the startup incoming breakers (E breakers). In addition to supplying emergency power for Oconee, the KHUs provide peaking power to the generation system. During periods of commercial power generation, the KHUs are operated within the acceptable region of the KHU operating restrictions. This ensures that the KHUs are able to perform their emergency power functions from an initial condition of commercial power generation. The KHU operating restrictions for commercial power generation are contained in UFSAR Chapter 16, (Ref. 2). The standby buses can also

BASES

BACKGROUND (continued)

receive power from a combustion turbine generator at the Lee Steam Station through a dedicated 100 kV transmission line, transformer CT-5, and both SL breakers. The 100 kV transmission line can be supplied from a Lee combustion turbine (LCT) and electrically separated from the system grid and offsite loads. The minimum capacity available from any of the multiple sources of AC power is 22.4MVA (limited by CT-4 and CT-5 transformer capacities).

APPLICABLE SAFETY ANALYSIS

The initial conditions of design basis transient and accident analyses in the UFSAR Chapter 6 (Ref. 4) and Chapter 15 (Ref. 5) assume ES systems are OPERABLE. The AC power system is designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ES systems so that the fuel, reactor coolant system, and containment design limits are not exceeded.

Consistent with the accident analysis assumptions of a loss of offsite power (LOOP) and a single failure of one onsite emergency power path, two onsite emergency power sources are required to be OPERABLE.

AC Sources – Operating are part of the primary success path and function to mitigate an accident or transient that presents a challenge to the integrity of a fission product barrier. As such, AC Sources – Operating satisfies the requirements of Criterion 3 of 10 CFR 50.36 (Ref. 3).

LCO

Two sources on separate towers connected to the 230 kV switchyard to a unit startup transformer and one main feeder bus are required to be OPERABLE. Two KHUs with one capable of automatically providing power through the underground emergency power path to both main feeder buses and the other capable of automatically providing power through the overhead emergency power path to both main feeder buses are required to be OPERABLE. The Keowee Reservoir level is required to be ≥ 775 feet above sea level to support OPERABILITY of the KHUs. The zone overlap protection circuitry is required to be OPERABLE when the overhead electrical disconnects for the KHU associated with the underground power path are closed to provide single failure protection for the KHUs.

BASES

LCO (continued)

Operable offsite sources are required to be "physically independent" (separate towers) prior to entering the 230 kV switchyard. Once the 230 kV lines enter the switchyard, an electrical pathway must exist through OPERABLE power circuit breakers (PCBs) and disconnects such that both sources are available to energize the Unit's startup transformer either automatically or with operator action. Once within the boundary of the switchyard, the electrical pathway may be the same for both independent offsite sources.' In addition, at least one E breaker must be available to automatically supply power to a main feeder bus from the energized startup transformer. The voltage provided to the startup transformer by the two independent offsite sources must be sufficient to ensure ES equipment will operate. Two of the following offsite sources are required:

1) Jocassee (from Jocassee) Black or White,
2) Dacus (from North Greenville) Black or White,
3) Oconee (from Central) Black or White,
4) Calhoun (from Central) Black or White,
5) Autobank transformer fed from either the Asbury (from Newport), Norcross (from Georgia Power), or Katoma (from Jocassee) 525 kV line.

An OPERABLE KHU and its required emergency power path are required to be able to provide sufficient power within specified limits of voltage and frequency within 23 seconds after an emergency start initiate signal and includes its required emergency power path, required instrumentation, controls, auxiliary and DC power, cooling and seal water, lubrication and other auxiliary equipment necessary to perform its safety function. Two emergency power paths are available. One emergency power path consists of an underground circuit while the other emergency power pathway uses an overhead circuit through the 230 kV switchyard.

BASES

LCO (continued)

An OPERABLE KHU and its required overhead emergency power path must be capable of automatically supplying power from the KHU through the KHU main step-up transformer, the 230 kV yellow bus, the Unit startup transformer and both E breakers to both main feeder buses. At least one channel of switchyard isolation (by actuation from degraded grid voltage protection) is required to be OPERABLE to isolate the 230 kV switchyard yellow bus. If closed, each N breaker must be capable of opening using either of its associated breaker trip circuits. Either of the following combinations provides an acceptable KHU and required overhead emergency power path:

| Keowee Hydro Unit | | Keowee Hydro Unit | |
|---|---|---|---|
| 1A) | Keowee Unit 1 generator, | 1B) | Keowee Unit 2 generator, |
| 2A) | Keowee ACB 1 (enabled by one channel of Switchyard Isolate Complete), | 2B) | Keowee ACB 2 (enabled by one channel of Switchyard Isolate Complete), |
| 3A) | Keowee auxiliary transformer 1X, Keowee ACB 5, Keowee Load Center 1X, | 3B) | Keowee auxiliary transformer 2X, Keowee ACB 6, Keowee Load Center 2X, |
| 4A) | Keowee MCC 1XA, | 4B) | Keowee MCC 2XA, |
| 5A) | Keowee Battery #1, Charger #1 or Standby Charger, and Distribution Center 1DA, | 5B) | Keowee Battery #2, Charger #2 or Standby Charger, and Distribution Center 2DA, |
| 6A) | ACB-1 to ACB-3 interlock, | 6B) | ACB-2 to ACB-4 interlock, |

7) Keowee reservoir level ≥ 775 feet above sea level,

Overhead Emergency Power Path

8) Keowee main step-up transformer,
9) PCB 9 (enabled by one channel of Switchyard Isolate Complete),
10) The 230kV switchyard yellow bus capable of being isolated by one channel of Switchyard Isolate,
11) A unit startup transformer and associated yellow bus PCB (CT-1 / PCB 18, CT-2 / PCB 27, CT-3 / PCB 30),
12) Both E breakers.

BASES

LCO (continued)

An OPERABLE KHU and its required underground emergency power path must be capable of automatically supplying power from the KHU through the underground feeder, transformer CT-4, both standby buses, and both Unit S breakers to both main feeder buses. If closed, each N breaker and each SL breaker must be capable of opening using either of its associated breaker trip circuits. Either of the following combinations provides an acceptable KHU and required underground emergency power path:

| Keowee Hydro Unit | | Keowee Hydro Unit | |
|---|---|---|---|
| 1A) | Keowee Unit 1 generator, | 1B) | Keowee Unit 2 generator, |
| 2A) | Keowee ACB 3, | 2B) | Keowee ACB 4, |
| 3A.1) | Keowee auxiliary transformer CX, Keowee ACB 7, Keowee Load Center 1X, | 3B.1) | Keowee auxiliary transformer CX, Keowee ACB 8, Keowee Load Center 2X, |
| 3A.2) | One Oconee Unit 1 S breaker capable of feeding switchgear 1TC, | 3B.2) | One Oconee Unit 1 S breaker capable of feeding switchgear 1TC, |
| 3A.3) | Switchgear 1TC capable of feeding Keowee auxiliary transformer CX, | 3B.3) | Switchgear 1TC capable of feeding Keowee auxiliary transformer CX, |
| 4A) | Keowee MCC 1XA, | 4B) | Keowee MCC 2XA, |
| 5A) | Keowee Battery #1, Charger #1 or Standby Charger, and Distribution Center 1DA, | 5B) | Keowee Battery #2, Charger #2 or Standby Charger, and Distribution Center 2DA, |
| 6A) | ACB-1 to ACB-3 interlock, | 6A) | ACB-2 to ACB-4 interlock, |

7) Keowee reservoir level ≥ 775 feet above sea level,

Underground Emergency Power Path

8) The underground feeder,
9) Transformer CT-4,
10) Both SK breakers,
11) Both standby buses,
12) Both S breakers, and
13) ACB-3 to ACB-4 interlock.

BASES

LCO (continued)

This LCO is modified by three Notes. Note 1 indicates that a unit startup transformer may be shared with a unit in MODES 5 and 6. Note 2 indicates that the requirements of Specification 5.5.18, "KHU Commercial Power Generation Testing Program," shall be met for commercial KHU power generation. Note 3 indicates that the requirements of Specification 5.5.19, "Lee Combustion Turbine Testing Program," shall be met when a Lee Combustion Turbine (LCT) is used to comply with Required Actions.

APPLICABILITY

The AC power sources are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure that:

a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of accidents and transients, and

b. Adequate core cooling is provided, and containment OPERABILITY and other vital functions are maintained in the event of a postulated accident.

AC source requirements during MODE 5 and 6 are covered in LCO 3.8.2, AC Sources-Shutdown.

ACTIONS

The ACTIONS are modified by a Note. The Note excludes the MODE change restriction of LCO 3.0.4 when both standby buses are energized from an LCT via an isolated power path to comply with Required Actions. This exception allow 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 additional capabilities afforded when both standby buses are energized from an LCT via an isolated power path.

A.1, A.2, A.3.1, and A.3.2

In the event a startup transformer becomes inoperable, it effectively causes the emergency overhead power path and both of the offsite sources to be inoperable. A KHU and its required underground power path remain available to ensure safe shutdown of the unit in the event of a transient or accident without a single failure.

BASES

ACTIONS

A.1, A.2, A.3.1, and A.3.2 (continued)

Operation may continue provided the KHU and its required underground emergency power path are tested using SR 3.8.1.3 within one hour if not performed in the previous 12 hours. This Required Action provides assurance that no undetected failures have occurred in the KHU and its required underground emergency power path. Since Required Action A.1 only specifies "perform," a failure of SR 3.8.1.3 acceptance criteria does not result in a Required Action not met. However, if the KHU and its required underground emergency path fails SR 3.8.1.3, both emergency power paths and both required offsite circuits are inoperable, and Condition I for both KHUs and their required emergency power paths inoperable for reasons other than Condition G and H is entered concurrent with Condition A.

If available, another Unit's startup transformer should be aligned to supply power to the affected Unit's auxiliaries so that offsite power sources and the KHU and its required overhead emergency power path will also be available if needed. Although this alignment restores the availability of the offsite sources and the KHU and its required overhead emergency power path, the shared startup transformer's capacity and voltage adequacy could be challenged under certain DBA conditions. The shared alignment is acceptable because the preferred mode of Unit shutdown is with reactor coolant pumps providing forced circulation and due to the low likelihood of an event challenging the capacity of the shared transformer during a 72 hour period to bring a Unit to MODE 5. Required Action A.3.1 requires that the unit startup transformer be restored to OPERABLE status and normal startup bus alignment in 36 hours or Required Action 3.2 requires designating one unit sharing the startup transformer, to be shutdown. For example, if Unit 1 and 2 are operating and CT-2 becomes inoperable, Unit 2 may align CT-1 to be available to the Unit 2 main feeder buses and continue operating for up to 36 hours. At that time, if CT-2 has not been restored to OPERABLE status, one Unit must be "designated" to be shutdown. The designated Unit must be shut down per ACTION B. Note that with one Unit in MODES 1, 2, 3 or 4 and another Unit in a condition other than MODES 1, 2, 3, or 4, the units may share a startup transformer indefinitely provided that the loads on the unit not in MODES 1, 2, 3 or 4 are maintained within acceptable limits. For example, if Unit 1 is in MODE 5 and CT-2 becomes inoperable, Unit 2 may align CT-1 to the Unit 2 main feeder buses and continue operation indefinitely.

BASES

ACTIONS (continued)

B.1 and B.2

When a unit is designated to be shutdown due to sharing a unit startup transformer per Required Action A.3.2, the unit must be brought to a MODE in which the LCO does not apply, since the shared unit startup transformer's capacity could be challenged under certain DBA conditions. To achieve this status, the unit must be brought to at least MODE 3 within 12 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

C.1, C.2.1, C.2.2.1, C.2.2.2, C.2.2.3, C.2.2.4, and C.2.2.5

With the KHU or its required overhead emergency power path inoperable due to reasons other than an inoperable startup transformer (Condition A), sufficient AC power sources remain available to ensure safe shutdown of the unit in the event of a transient or accident. Operation may continue if the OPERABILITY of the remaining KHU and its required underground emergency power path is determined by performing SR 3.8.1.3 within 1 hour if not performed in the previous 12 hours and once every 7 days thereafter. This demonstration assures the remaining emergency power path is not inoperable due to a common cause or other failure. Testing on a 7 day Frequency is acceptable since both standby buses must be energized from an LCT via an isolated power path when in Condition C for > 72 hours. When the standby buses are energized by an LCT via an isolated power path, the likelihood that the OPERABLE KHU and its required underground emergency power path will be required is decreased. Since Required Action C.1 only specifies "perform," a failure of SR 3.8.1.3 acceptance criteria does not result in a Required Action not met. SR 3.8.1.3 is only required to be performed when the KHU associated with the underground emergency power path is OPERABLE.

If the KHU and its required underground emergency path fails SR 3.8.1.3, both KHUs and their required emergency power paths are inoperable, and Condition I (Both KHUs or their required emergency power paths inoperable for reasons other than Condition G or H) is entered concurrent with Condition C.

BASES

ACTIONS

C.1, C.2.1, C.2.2.1, C.2.2.2, C.2.2.3, C.2.2.4, and C.2.2.5 (continued)

If the inoperable KHU or its required overhead emergency power path are not restored to OPERABLE status within 72 hours as required by Required Action C.2.1, a controlled shutdown must be initiated as required by the Required Actions for Condition M unless the extended Completion Times of Required Action C.2.2.5 are applicable. The second Completion Time for Required Action C.2.1 establishes a limit on the maximum time allowed for a KHU to be inoperable during any single contiguous occurrence of having a KHU inoperable. If Condition C is entered as a result of switching an inoperable KHU from the underground to the overhead emergency power path, it may have been inoperable for up to 72 hours. This could lead to a total of 144 hours since the initial failure of the KHU. The second Completion Time allows for an exception to the normal "time zero" for beginning the allowed time "clock." This will result in establishing the "time zero" at the time the KHU become inoperable, instead of at the time Condition C was entered.

The extended Completion Times of Required Action C.2.2.5 apply when the KHU or its required overhead emergency power path is inoperable due to an inoperable Keowee main step-up transformer or an inoperable KHU (if not used for that KHU in the previous 3 years). In order to use the extended Completion Times, within 72 hours of entering Condition C both standby buses must be energized from an LCT (Required Action C.2.2.1), KHU generation to the grid except for testing must be suspended (Required Action C.2.2.2), the remaining KHU and its required underground emergency power path and both required offsite sources must be verified OPERABLE, the LCOs indicated in Required Action C.2.2.3 must be verified to be met, and alternate power source capability must be verified by performing SR 3.8.1.16.

Required Action C.2.2.5 permits maintenance and repair of a Keowee main step-up transformer which requires longer than 72 hours. Transformer replacement is rare but is time extensive. A 28 day Completion Time is permitted by Required Action C.2.2.5 to restore the KHU and its overhead power path to OPERABLE status when inoperable due to an inoperable Keowee main step-up transformer. This allows a reasonable period of time for transformer replacement.

Required Action C.2.2.5 also permits maintenance and repair of a KHU which requires longer than 72 hours. The primary long term maintenance items are expected to be hydro turbine runner and discharge ring welding

BASES

ACTIONS

C.1, C.2.1, C.2.2.1, C.2.2.2, C.2.2.3, C.2.2.4, and C.2.2.5 (continued)

repairs which are estimated to be necessary every six to eight years. Also, generator thrust and guide bearing replacements are necessary. Other items which manifest as failures are expected to be rare and may be performed during the permitted maintenance periods. As such, the 45 day restoration time of Required Action C.2.2.5 is allowed only once in a three year period for each KHU. This Completion Time is 45 days from discovery of initial inoperability of the KHU. This effectively limits the time the KHU can be inoperable to 45 days from discovery of initial inoperability rather than 45 days from entry into Condition C and precludes any additional time that may be gained as a result of switching an inoperable KHU from the underground to the overhead emergency power path.

Required Actions C.2.2.1, C.2.2.2, C.2.2.3, and C.2.2.4 must be met in order to allow the longer restoration times of Required Action C.2.2.5. Required Action C.2.2.1 requires that both standby buses be energized using an LCT through the 100 kV transmission circuit. With this arrangement (100 kV transmission circuit electrically separated from the system grid and all offsite loads), a high degree of reliability for the emergency power system is provided. In this configuration, the LCT is serving as a second emergency power source, however, since the 100 kV transmission circuit is vulnerable to severe weather a time limit is imposed. The second Completion Time of Required Action C.2.2.1 permits the standby buses to be re-energized by an LCT within 1 hour in the event this source is subsequently lost. Required Action C.2.2.2 requires suspension of KHU generation to the grid except for testing. The restriction reduces the number of possible failures which could cause loss of the underground emergency power path. Required Action C.2.2.3 requires verifying by administrative means that the remaining KHU and its required underground emergency power path and both required offsite sources are OPERABLE. This provides additional assurance that offsite power will be available. In addition, this assures that the KHU and its required underground emergency power path are available. Required Action C.2.2.3 also requires verifying by administrative means that the requirements of the following LCOs are met:

LCO 3.8.3, "DC Sources – Operating;"

LCO 3.8.6, "Vital Inverters – Operating;"

LCO 3.8.8, "Distribution Systems – Operating;"

BASES

ACTIONS

C.1, C.2.1, C.2.2.1, C.2.2.2, C.2.2.3, C.2.2.4, and C.2.2.5 (continued)

LCO 3.3.17, "EPSL Automatic Transfer Function;"

LCO 3.3.18, "EPSL Voltage Sensing Circuits;"

LCO 3.3.19, "EPSL 230 kV Switchyard DGVP;" and

LCO 3.3.21, "EPSL Keowee Emergency Start Function."

This increases the probability, even in the unlikely event of an additional failure, that the DC power system and the 120 VAC Vital Instrumentation power panelboards will function as required to support EPSL, power will not be lost to ES equipment, and EPSL will function as required.

Verifying by administrative means allows a check of logs or other information to determine the OPERABILITY status of required equipment in place of requiring unique performance of Surveillance Requirements. If the AC Source is subsequently determined inoperable, or an LCO stated in Required Action C.2.2.3 is subsequently determined not met, continued operation up to a maximum of four hours is allowed by ACTION L.

Required Action C.2.2.4 requires verifying alternate power source capability by performing SR 3.8.1.16. This confirms that entry into Condition C is due only to an inoperable main step-up transformer or an inoperable KHU, as applicable. If SR 3.8.1.16 is subsequently determined not met, continued operation up to a maximum of four hours is allowed by ACTION L.

D.1, D.2 and D.3

With the KHU or its required underground emergency power path inoperable, sufficient AC power sources remain available to ensure safe shutdown of the unit in the event of a transient or accident. Operation may continue for 72 hours if the remaining KHU and its required overhead emergency power path are tested using SR 3.8.1.4 within one hour if not performed in the previous 12 hours. SR 3.8.1.4 is only required to be performed when the KHU associated with the overhead emergency power path is OPERABLE. This Required Action provides assurance that no undetected failures have occurred in the overhead emergency power path. Since Required Action D.1 only specifies

BASES

ACTIONS

D.1, D.2 and D.3 (continued)

"perform," a failure of SR 3.8.1.4 acceptance criteria does not result in a Required Action not met. However, if the KHU and its required overhead emergency path fails SR 3.8.1.4, both KHUs and their required emergency power paths are inoperable, and Condition I for both KHUs and their emergency power paths inoperable for reasons other than Condition G or H is entered concurrent with Condition D. This demonstration is to assure that the remaining emergency power path is not inoperable due to a common cause or due to an undetected failure. For outages of the KHU and its required underground emergency power path in excess of 24 hours, an LCT (using the 100 kV transmission circuit electrically separated from the grid and offsite loads) must energize a standby bus prior to the outage exceeding 24 hours. This ensures the availability of a power source on the standby buses when the KHU and its required underground emergency power path are out of service in excess of 24 hours. The second Completion Time of Required Action D.2 permits the standby buses to be re-energized by an LCT within 1 hour in the event this source is subsequently lost.

The second Completion Time for Required Action D.3 establishes a limit on the maximum time allowed for a KHU to be inoperable during any single contiguous occurrence of having a KHU inoperable. If Condition D is entered as a result of switching an inoperable KHU from the overhead to the underground emergency power path, it may have been inoperable for up to 72 hours. This could lead to a total of 144 hours since the initial failure of the KHU. The second Completion Time allows for an exception to the normal "time zero" for beginning the allowed time "clock." This will result in establishing the "time zero" at the time the KHU become inoperable, instead of at the time Condition D was entered.

E 1 and E.2

If the Required Action and associated Completion Time for Required Action D.2 are 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 12 hours for one Oconee unit and 24 hours for other Oconee unit(s) and to MODE 5 within 84 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 plant systems.

BASES

ACTIONS (continued)

F.1 and F.2

With the zone overlap protection circuitry inoperable when the overhead electrical disconnects for the KHU associated with the underground power path are closed, the zone overlap protection circuitry must be restored to OPERABLE status or the overhead electrical disconnects must be opened within 72 hours. In this Condition, both KHUs and their required emergency power paths are OPERABLE, however a single failure could result in the loss of both KHUs.

G.1

With both emergency power paths inoperable due to an E breaker and S breaker inoperable on the same main feeder bus, one breaker must be restored to OPERABLE status. In this Condition, both emergency power paths can still provide power to the remaining main feeder bus.

H.1 and H.2

With both KHUs or their required emergency power paths inoperable for planned maintenance or test with both standby buses energized from an LCT via an isolated power path, the KHU must be restored to OPERABLE status within 60 hours. Operation with both KHUs and their required power paths inoperable is permitted for 60 hours provided that both standby buses are energized using an LCT through the 100 kV transmission circuit and the requirements of the Note to the Condition are met. The Note to the Condition indicates that it may only be entered when both offsite sources are verified by administrative means to be OPERABLE and the requirements of the following LCOs are verified by administrative means to be met:

LCO 3.8.3, "DC Sources – Operating;"

LCO 3.8.6, "Vital Inverters – Operating;"

LCO 3.8.8, "Distribution Systems – Operating;"

LCO 3.3.17, "EPSL Automatic Transfer Function;"

LCO 3.3.18, "EPSL Voltage Sensing Circuits;" and

LCO 3.3.19, "EPSL 230 kV Switchyard DGVP."

BASES

ACTIONS

H.1 and H.2 (continued)

This increases the probability, even in the unlikely event of an additional failure, that the DC power system and the 120 VAC Vital Instrumentation power panelboards will function as required to support EPSL, power will not be lost to ES equipment, and EPSL will function as required.

Verifying by administrative means allows a check of logs or other information to determine the OPERABILITY status of required equipment in place of requiring unique performance of Surveillance Requirements. If the AC Source is subsequently determined inoperable, or an LCO stated in the Note to Condition H is subsequently determined not met, continued operation up to a maximum of four hours is allowed by ACTION L.

With both standby buses energized from an LCT via an isolated power path (100 kV transmission circuit electrically separated from the system grid and all offsite loads), a high degree of reliability for the emergency power system is provided. In this configuration, the LCT is serving as a second emergency power source, however, since the Oconee Units are vulnerable to a single failure of the 100 kV transmission circuit a time limit of 60 hours is imposed. Required Action H.1 permits the standby buses to be re-energized by an LCT within 1 hour in the event this source is subsequently lost.

If both emergency power paths are restored, unrestricted operation may continue. If only one power path is restored, operation may continue per ACTIONS C or D.

I.1, I.2, and I.3

With both KHUs or their required emergency power paths inoperable for reasons other than Conditions G and H, insufficient standby AC power sources are available to supply the minimum required ES functions. In this Condition, the offsite power system is the only source of AC power available for this level of degradation. The risk associated with continued operation for one hour without an emergency power source is considered acceptable due to the low likelihood of a LOOP during this time period, and because of the potential for grid instability caused by the simultaneous shutdown of all three units. This instability would increase the probability of a total loss of AC power. Operation with both KHUs or their required power paths inoperable is permitted for 12 hours provided

BASES

ACTIONS

I.1, I.2, and I.3 (continued)

that Required Actions I.1 and I.2 are met. Required Action I.1 requires that both standby buses be energized using an LCT via an isolated power path. With this arrangement (100 kV transmission circuit electrically separated from the system grid and all offsite loads), a high degree of reliability for the emergency power system is provided. In this configuration, the LCT is serving as a second emergency power source, however, since the Oconee Units are vulnerable to a single failure of the 100 kV transmission circuit a time limit of 12 hours is imposed. The second Completion Time of Required Action I.1 permits the standby buses to be re-energized by an LCT within 1 hour in the event this source is subsequently lost. Required Action I.2 requires that the OPERABILITY status of both offsite sources be determined by administrative means and that the OPERABILITY status of equipment required by the following LCOs be determined by administrative means:

LCO 3.8.3, "DC Sources – Operating;"

LCO 3.8.6, "Vital Inverters – Operating;"

LCO 3.8.8, "Distribution Systems – Operating;"

LCO 3.3.17, "EPSL Automatic Transfer Function;"

LCO 3.3.18, "EPSL Voltage Sensing Circuits;" and

LCO 3.3.19, "EPSL 230 kV Switchyard DGVP."

This increases the probability, even in the unlikely event of an additional failure, that the DC power system and the 120 VAC Vital Instrumentation power panelboards will function as required to support EPSL, power will not be lost to ES equipment, and EPSL will function as required.

Determining by administrative means allows a check of logs or other information to determine the OPERABILITY status of required equipment in place of requiring unique performance of Surveillance Requirements. If the AC Source is initially or subsequently determined inoperable, or an LCO stated in Required Action I.2 is initially or subsequently determined not met, continued operation up to a maximum of four hours is allowed by ACTION L.

If both emergency power paths are restored, unrestricted operation may continue. If only one power path is restored, operation may continue per ACTIONS C or D.

BASES

ACTIONS (continued)

J.1, J.2, and J.3

With one or both required offsite sources inoperable for reasons other than Condition A, sufficient AC power sources are available to supply necessary loads in the event of a DBA. However, since the AC power system is degraded below the Technical Specification requirements, a time limit on continued operation is imposed. With only one of the required offsite sources OPERABLE, the likelihood of a LOOP is increased such that the Required Actions for all required offsite circuits inoperable are conservatively followed. The risk associated with continued operation for one hour without a required offsite AC source is considered acceptable due to the low likelihood of a LOOP during this time period, and because of the potential for grid instability caused by the simultaneous shutdown of all three units.

Operation with one or both required offsite sources inoperable is permitted for 24 hours provided that Required Actions J.1 and J.2 are met. Required Action J.1 requires that both standby buses be energized using an LCT via an isolated power path. With this arrangement (100 kV transmission circuit electrically separated from the system grid and all offsite loads), a high degree of reliability for the emergency power system is provided. In this configuration, the LCT is serving as an emergency power source, however, since the Oconee units are vulnerable to a single failure of the 100 kV transmission circuit a time limit is imposed. The second Completion Time of Required Action J.1 permits the standby buses to be re-energized by an LCT within 1 hour in the event this source is subsequently lost. Required Action J.2 requires that the OPERABILITY status of both KHUs and their required emergency power paths be determined by administrative means and that the OPERABILITY status of equipment required by the following LCOs be determined by administrative means:

LCO 3.8.3, "DC Sources – Operating;"

LCO 3.8.6, "Vital Inverters – Operating;"

LCO 3.8.8, "Distribution Systems – Operating;"

LCO 3.3.17, "EPSL Automatic Transfer Function;"

LCO 3.3.18, "EPSL Voltage Sensing Circuits;"

LCO 3.3.19, "EPSL 230 kV Switchyard DGVP," and

LCO 3.3.21, "EPSL Keowee Emergency Start Function."

BASES

ACTIONS

J.1, J.2, and J.3 (continued)

This increases the probability, even in the unlikely event of an additional failure, that the DC power system and the 120 VAC Vital Instrumentation power panelboards will function as required to support EPSL, power will not be lost to ES equipment, and EPSL will function as required.

Determining by administrative means allows a check of logs or other information to determine the OPERABILITY status of required equipment in place of requiring unique performance of Surveillance Requirements. If the AC Source is initially or subsequently determined inoperable, or an LCO stated in Required Action J.2 is initially or subsequently determined not met, continued operation up to a maximum of four hours is allowed by ACTION L.

K.1

The two trip circuits for each closed N and SL breakers are required to ensure both breakers will open. An N breaker trip circuit encompasses those portions of the breaker control circuits necessary to trip the associated N breaker from the output of the 2 out of 3 logic matrix formed by the auxiliary transformer's undervoltage sensing circuits up to and including an individual trip coil for the associated N breaker. The undervoltage sensing channels for the auxiliary transformer are addressed in LCO 3.3.18, "Emergency Power Switching Logic (EPSL) Voltage Sensing Circuits." An SL breaker trip circuit encompasses those portions of the breaker control circuits necessary to trip the SL breaker from the output of both 2 out of 3 logic matrices formed by each standby bus's undervoltage sensing circuits up to and including an individual trip coil for the associated SL breaker. The undervoltage sensing channels for the CT- 5 transformer are addressed in LCO 3.3.18, "Emergency Power Switching Logic (EPSL) Voltage Sensing Circuits." With one trip circuit inoperable a single failure could cause an N or SL breaker to not open. This could prevent the transfer to other available sources. Therefore, 24 hours is allowed to repair the trip circuit or open the breaker (opening the breaker results in exiting the Condition). The Completion Time is based on engineering judgement taking into consideration the time required to complete the required action and the availability of the remaining trip circuit.

A Note modifies the Condition, indicating that separate Condition Entry is permitted for each breaker. Thus, Completion Times are tracked separately for the N1, N2, SL1, and SL2 breaker.

BASES

ACTIONS (continued)

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

With an AC Source inoperable or LCO not met, as stated in Note for Condition H entry; or with an AC Source inoperable or LCO not met, as stated in Required Action C.2.2.3 when in Condition C for > 72 hours; or with an AC Source inoperable or LCO not met, as stated in Required Action I.2 or J.2 when in Conditions I or J for > 1 hour; or with SR 3.8.1.16 not met, Required Action L.1, L.2 and L.3 requires restoration within four hours. Condition L is modified by a Note indicating that separate Condition entry is permitted for each inoperable AC Source, and LCO or SR not met. The Required Action is modified by a Note that allows the remaining OPERABLE KHU and its required emergency power path to be made inoperable for up to 12 hours if required to restore both KHUs and their required emergency power paths to OPERABLE status. This note is necessary since certain actions such as dewatering the penstock may be necessary to restore the inoperable KHU although these actions would also cause both KHUs to be inoperable.

The purpose of this Required Action is to restrict the allowed outage time for an inoperable AC Source or equipment required by an LCO when in Conditions C, H, I or J. For Conditions I and J when the LCOs stated are initially not met, the maximum Completion Time is four hours or the remaining Completion Time allowed by the stated LCO, whichever is shorter.

M.1 and M.2

If a Required Action and associated Completion Time for Condition C, F, G, H, I, J, K or L are not met; or if a Required Action and associated Completion Time are not met for Required Action D.1 or D.3, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 12 hours and to MODE 5 within 84 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 (continued)

SURVEILLANCE REQUIREMENTS

SR 3.8.1.1

This SR ensures proper circuit continuity for the offsite AC electrical power supply to the onsite distribution network and availability of offsite AC electrical power. The breaker alignment verifies that each breaker is in its correct position to ensure that distribution buses and loads are connected to their power source, and that appropriate separation of offsite sources is maintained. The 7 day Frequency is adequate since breaker position is not likely to change without the operator being aware of it and because its status is displayed in the control room.

SR 3.8.1.2

This SR verifies adequate battery voltage when the KHU batteries are on float charge. This SR is performed to verify KHU battery OPERABILITY. The Frequency of once per 7 days is consistent with manufacturers recommendations and IEEE-450 (Ref. 8).

SR 3.8.1.3

This SR verifies the availability of the KHU associated with the underground emergency power path to start automatically and energize the underground power path. Utilization of either the auto-start or emergency start sequence assures the control function OPERABILITY by verifying proper speed control and voltage. Power path verification is included to demonstrate breaker OPERABILITY from the KHU onto the standby buses. This is accomplished by closing the Keowee Feeder Breakers (SK) to energize each deenergized standby bus. The 31 day Frequency is adequate based on operating experience to provide reliability verification without excessive equipment cycling for testing.

SR 3.8.1.4

This surveillance verifies the availability of the KHU associated with the overhead emergency power path. Utilization of either the auto-start or emergency start sequence assures the control function OPERABILITY by verifying proper speed control and voltage. The ability to supply the overhead emergency power path is satisfied by demonstrating the ability to synchronize (automatically or manually) the KHU with the grid system. The SR also requires that the underground power path be energized

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.4 (continued)

after removing the KHU from the overhead emergency power path. This surveillance can be satisfied by first demonstrating the ability of the KHU associated with the underground emergency path to energize the underground path then synchronizing the KHU to the overhead emergency power path. The SR is modified by a Note indicating that the requirement to energize the underground emergency power path is not applicable when the overhead disconnects are open for the KHU associated with the underground emergency power path or 2) when complying with Required Action D.1. The latter exception is necessary since Required Action D.1 continues to be applicable when both KHUs are inoperable.

The 31 day Frequency for this Surveillance was determined to be adequate based on operating experience to provide reliability verification without excessive equipment cycling for testing.

SR 3.8.1.5

This surveillance verifies OPERABILITY of the trip functions of each closed SL and each closed N breaker. Neither of these breakers have any automatic close functions; therefore, only the trip coils require verification. Cycling of each breaker demonstrates functional OPERABILITY and the coil monitor circuits verify the integrity of each trip coil. The 31 day frequency is based on operating experience.

This SR modified by a Note that states it is not required to be performed for an SL breaker when its standby bus is energized from a LCT via an isolated power path. This is necessary since the standby buses are required to be energized from a LCT by several Required Actions of Specification 3.8.1 and the breakers must remain closed to energize the standby buses from a LCT.

SR 3.8.1.6

Infrequently used source breakers are cycled to ensure OPERABILITY. The Standby breakers are to be cycled one breaker at a time to prevent inadvertent interconnection of two units through the standby bus breakers. Cycling the startup breakers verifies OPERABILITY of the breakers and associated interlock circuitry between the normal and

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.6 (continued)

startup breakers. This circuitry provides an automatic, smooth, and safe transfer of auxiliaries in both directions between sources. The 31 day Frequency for this Surveillance was determined to be adequate based on operating experience to provide reliability verification without excessive equipment cycling for testing.

This SR is modified by a Note which states the SR is not required to be performed for an S breaker when its standby bus is energized from a LCT via an isolated power path. This is necessary since the standby buses are required to be energized from a LCT by several Required Actions of Specification 3.8.1 and cycling the S breakers connects the standby buses with the main feeder buses which are energized from another source.

SR 3.8.1.7

The KHU tie breakers to the underground path, ACB3 and ACB4, are interlocked to prevent cross-connection of the KHU generators. The safety analysis utilizes two independent power paths for accommodating single failures in applicable accidents. Connection of both generators to the underground path compromises the redundancy of the emergency power paths. Installed test logic is used to verify a circuit to the close coil on one underground ACB does not exist with the other underground ACB closed. The 12 month Frequency for this surveillance is adequate based on operating experience to provide reliability verification without excessive equipment cycling for testing.

SR 3.8.1.8

Each KHU tie breaker to the underground emergency power path and tie breaker to the overhead emergency path, are interlocked to prevent the unit associated with the underground circuit from automatically connecting to the overhead emergency power path. The safety analysis utilizes two independent power paths for accommodating single failures in applicable accidents. Connection of both generators to the overhead emergency power path compromises the redundancy of the emergency power paths. Temporary test instrumentation is used to verify a circuit to the close coil on the overhead ACB does not exist with the Underground ACB closed. The 12 month Frequency for this Surveillance was determined to be adequate based on operating experience to provide reliability verification without excessive equipment cycling for testing.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.9

This surveillance verifies the KHUs' response time to an Emergency Start signal (normally performed using a pushbutton in the control room) to ensure ES equipment will have adequate power for accident mitigation. UFSAR Section 6.3.3.3 (Ref. 9) establishes the 23 second time requirement for each KHU to achieve rated frequency and voltage. Since the only available loads of adequate magnitude for simulating a accident is the grid, subsequent loading on the grid is required to verify the KHU's ability to assume rapid loading under accident conditions. Sequential block loads are not available to fully test this feature. This is the reason for the requirement to load the KHUs at the maximum practical rate. The 12 month Frequency for this SR is adequate based on operating experience to provide reliability verification without excessive equipment cycling for testing.

SR 3.8.1.10

A battery service test is a special test of the battery capability, as found, to satisfy the design requirements (battery duty cycle) of the DC electrical power system. The discharge rate and test length should correspond to the design duty cycle requirements as specified in Reference 4.

The Surveillance Frequency of 12 months is consistent with the recommendations of Regulatory Guide 1.32 (Ref. 6) and Regulatory Guide 1.129 (Ref. 7), which state that the battery service test should be performed with intervals between tests not to exceed 18 months.

SR 3.8.1.11

Visual inspection of the battery cells, cell plates, and battery racks provides an indication of physical damage or abnormal deterioration that could potentially degrade battery performance. The 12 month Frequency for this SR is consistent with manufacturers recommendations and IEEE-450 (Ref. 8), which recommends detailed visual inspection of cell condition and rack integrity on a yearly basis.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.12

Verification of cell to cell connection cleanliness, tightness, and proper coating with anti-corrosion grease provides an indication of any abnormal condition, and assures continued OPERABILITY of the battery. The 12 month frequency is based on engineering judgement and operational experience and is sufficient to detect cell connection degradation when it is properly coupled with other surveillances more frequently performed to detect abnormalities.

SR 3.8.1.13

The KHU underground ACBs have a control feature which will automatically close the KHU, that is pre-selected to the overhead path, into the underground path upon an electrical fault in the zone overlap region of the protective relaying. This circuitry prevents an electrical fault in the zone overlap region of the protective relaying from locking out both emergency power paths during dual KHU grid generation. In order to ensure this circuitry is OPERABLE, an electrical fault is simulated in the zone overlap region and the associated underground ACBs are verified to operate correctly. This surveillance is required on a 12 month Frequency. The 12 month Frequency is based on engineering judgement and provides reasonable assurance that the zone overlap protection circuitry is operating properly.

This SR is modified by a Note indicating the SR is only applicable when the overhead disconnects to the underground KHU are closed. When the overhead disconnects to the underground KHU are open, the circuitry preventing the zone overlap protective lockout of both KHUs is not needed.

SR 3.8.1.14

This surveillance verifies OPERABILITY of the trip functions of the SL and N breakers. This SR verifies each trip circuit of each breaker independently opens each breaker. Neither of these breakers have any automatic close functions; therefore, only the trip circuits require verification. The 18 month Frequency is based on engineering judgement and provides reasonable assurance that the SL and N breakers will trip when required.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.14 (continued)

The SR is modified by a Note indicating that the SR is not required for an SL breaker when its standby bus is energized by a LCT via an isolated power path. This is necessary since the standby buses are required to be energized from a LCT by several Required Actions of Specification 3.8.1 and the breakers must remain closed to energize the standby buses from a LCT.

SR 3.8.1.15

This surveillance verifies proper operation of the 230 kV switchyard circuit breakers upon an actual or simulated actuation of the Switchyard Isolation circuitry. This test causes an actual switchyard isolation (by actuation of degraded grid voltage protection) and alignment of KHUs to the overhead and underground emergency power paths. An 18 month Frequency minimizes the impact to the Station and the operating Units which are connected to the 230 kV switchyard. The effect of this SR is not significant because the generator red bus tie breakers and feeders from the Oconee 230 kV switchyard red bus to the system grid remain closed. Either Switchyard Isolation Channel causes full system realignment, which involves a complete switchyard realignment. To avoid excessive switchyard circuit breaker cycling, realignment and KHU emergency start functions, this SR need be performed only once each SR interval.

This SR is modified by a Note. This Note states the redundant breaker trip coils shall be verified on a STAGGERED TEST BASIS. Verifying the trip coils on a STAGGERED TEST BASIS precludes unnecessary breaker operation and minimizes the impact to the Station and the operating Units which are connected to the 230 kV switchyard.

SR 3.8.1.16

This SR verifies by administrative means that one KHU provides an alternate manual AC power source capability by manual or automatic KHU start with manual synchronize, or breaker closure, to energize its non-required emergency power path. That is, when the KHU to the overhead emergency power path is inoperable, the SR verifies by administrative means that the overhead emergency power path is

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.16 (continued)

OPERABLE. When the overhead emergency power path is inoperable, the SR verifies by administrative means that the KHU associated with the overhead emergency power path is OPERABLE.

This SR is modified by a Note indicating that the SR is only applicable when complying with Required Action C.2.2.4.

REFERENCES

1. UFSAR, Section 3.1.39
2. UFSAR, Chapter 16
3. 10 CFR 50.36
4. UFSAR, Chapter 6
5. UFSAR, Chapter 15
6 Regulatory Guide 1.32
7 Regulatory Guide 1.129
8 IEEE-450-1980
9. UFSAR, Section 6.3.3.3

# B 3.8 ELECTRICAL POWER SYSTEMS

## B 3.8.2 AC Sources – Shutdown

BASES

BACKGROUND

A description of the AC sources, except AC sources utilizing transformer CT-5, is provided in the Bases for LCO 3.8.1, "AC Sources – Operating." An additional source of AC power is available either directly from the 100 kV Central Tie Substation or from the combustion turbines at Lee Steam Station via a 100 kV transmission line connected to Transformer CT-5. This single 100 kV circuit is connected to the 100 kV transmission system through the substation at Central, located eight miles from Oconee. The Central Substation is connected to Lee Steam Station twenty-two miles away through a similar 100 kV line. This line can either be isolated from the balance of the transmission system to supply emergency power to Oconee from Lee Steam Station, or offsite power can be supplied directly from the 100 kV system from the Central Tie Substation. When CT-5 is energized from the 100 kV system, this is an acceptable offsite source for Oconee Units in MODES 5 and 6. When CT-5 is energized from an OPERABLE Lee Combustion Turbine (LCT) and isolated from the balance of the transmission system, this source is an acceptable emergency power source.

Located at Lee Steam Station are three 44.1 MVA combustion turbines. One of these three combustion turbines can be started in one hour and connected to the 100 kV line. Transformer CT-5 is sized to carry the engineered safeguards auxiliaries of one unit plus the shutdown loads of the other two units.

APPLICABLE SAFETY ANALYSES

The OPERABILITY of the minimum AC sources during MODES 5 and 6 and during movement of irradiated fuel assemblies ensures that:

a. The unit can be maintained in the shutdown or refueling condition for extended periods;

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and

c. Adequate AC electrical power is provided to mitigate events postulated during shutdown, such as a fuel handling accident.

BASES

APPLICABLE SAFETY ANALYSES (continued)

In general, when the unit is shut down, the Technical Specifications requirements ensure that the unit has the capability to mitigate the consequences of postulated accidents. However, assuming a single failure and concurrent loss of all offsite or all onsite power is not required. The rationale for this is based on the fact that many accidents that are analyzed in MODES 1, 2, 3, and 4 have no specific analyses in MODES 5 and 6. Worst-case bounding events are deemed not credible in MODES 5 and 6 because the energy contained within the reactor pressure boundary, reactor coolant temperature and pressure, and the corresponding stresses result in the probabilities of occurrence being significantly reduced or eliminated, and in minimal consequences. These deviations from accident analysis assumptions and design requirements during shutdown conditions are allowed by the LCO for required systems.

During MODES 1, 2, 3, and 4 various deviations from the analysis assumptions and design requirements are allowed within the Required Actions. This allowance is in recognition that certain testing and maintenance activities must be conducted provided an acceptable level of risk is not exceeded. During MODES 5 and 6, performance of a significant number of required testing and maintenance activities is also required. In MODES 5 and 6, the activities are generally planned and administratively controlled. Relaxations from MODE 1, 2, 3, and 4 LCO requirements are acceptable during shutdown MODES based on:

a. The fact that time in an outage is limited. This is a risk prudent goal as well as a utility economic consideration;

b. Requiring appropriate compensatory measures for certain conditions. These may include administrative controls, reliance on systems that do not necessarily meet typical design requirements applied to systems credited in operating MODE analyses, or both;

c. Prudent utility consideration of the risk associated with multiple activities that could affect multiple systems; and

d. Maintaining, to the extent practical, the ability to perform required functions (even if not meeting MODE 1, 2, 3, and 4 OPERABILITY requirements) with systems assumed to function during an event.

BASES

APPLICABLE SAFETY ANALYSES (continued)

In the event of an accident during shutdown, this LCO ensures the capability to support systems necessary to avoid immediate difficulty, assuming either a loss of all offsite power or a loss of all onsite emergency power sources and their associated emergency power paths.

The AC sources satisfy Criterion 3 of the 10 CFR 50.36 (Ref. 1).

LCO

One offsite source capable of supplying the onsite power distribution system(s) of LCO 3.8.9, "Distribution Systems – Shutdown," ensures that all required loads are powered from offsite power. An OPERABLE emergency power source, associated with a distribution system required to be OPERABLE by LCO 3.8.9, ensures a diverse power source is available to provide electrical power support, assuming a loss of the offsite source. Together, OPERABILITY of the required offsite source and emergency power source ensure the availability of sufficient AC sources to operate the unit in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents).

The qualified offsite source must be capable of maintaining rated frequency and voltage, and accepting required loads during an accident, while connected to the main feeder bus(es). Qualified offsite source are those that are described in the UFSAR and are part of the licensing basis for the unit.

An offsite source can be an offsite circuit available or connected through to the 230 kV switchyard to the startup transformer and to one main feeder bus. Additionally, the offsite source can be an offsite circuit available or connected though the 230 kV switchyard (525 kV switchyard for Unit 3) to a backcharged unit main step-up transformer and unit auxiliary transformer to one main feeder bus. Another alternative is the energized Central 100 kV switchyard available or connected through the 100 kV line and transformer CT-5 to one main feeder bus.

In MODES 5 or 6 and during movement of irradiated fuel, a Lee Combustion Turbine (LCT) energizing one standby bus via an isolated power path to one main feeder bus can be utilized as an emergency power source. The LCT is required to provide power within limits of voltage and frequency using the 100 kV transmission line electrically separated from the system grid and offsite loads energizing one or more standby buses through transformer CT-5. The required number of energized standby buses is based upon the requirements of LCO 3.8.9, "Distribution System – Shutdown."

BASES

LCO (continued)

An OPERABLE KHU must be capable of starting, accelerating to rated speed and voltage, and connecting to the main feeder bus(es). The sequence must be capable of being accomplished within 23 seconds after a manual emergency start initiation signal. An emergency power source must be capable of accepting required loads and must continue to operate until offsite power can be restored to the main feeder buses.

This LCO is modified by three Notes. Note 1 indicates that a unit startup transformer may be shared with a unit in MODES 5 and 6. Note 2 indicates that the requirements of Specification 5.5.19, "Lee Combustion Turbine Testing Program," shall be met when a Lee Combustion Turbine (LCT) is used for the emergency power requirements. Note 3 indicates that the required emergency power source and the required offsite power source shall not be susceptible to a failure disabling both sources.

The required emergency power source and required offsite source cannot be susceptible to a failure disabling both sources. If the required offsite source is the 230 kV switchyard and the startup transformer energizing the required main feeder bus(es), the KHU and its required underground emergency power path are required to be OPERABLE since it is not subject to a failure, such as an inoperable startup transformer, which simultaneously disables the offsite source. If the Central switchyard is serving as the required offsite source through the CT-5 transformer with a power path through only one standby bus, the KHU and its required underground emergency power path cannot be used as the emergency power source if the power path is through the same standby bus since a single failure of a standby bus would disable both sources. Conversely, if an LCT is being used as an emergency power source, the required offsite source must be an offsite circuit available or connected through the startup transformer or a backcharged unit main step-up transformer and the unit auxiliary transformer.

APPLICABILITY

The AC sources required to be OPERABLE in MODES 5 and 6 and during movement of irradiated fuel assemblies provide assurance that:

a. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel assemblies;

b. Systems needed to mitigate a fuel handling accident are available;

c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

BASES

APPLICABILITY (continued)

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

The AC power requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.1.

ACTIONS

A.1

An offsite source would be considered inoperable if it were not available to one required main feeder bus. Although two main feeder buses may be required by LCO 3.8.9, the one main feeder bus with offsite power available may be capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS and fuel movement. By the allowance of the option to declare features inoperable with no offsite power available, appropriate restrictions will be implemented in accordance with the affected required features LCO's ACTIONS.

A.2.1, A.2.2, A.2.3, A.2.4, B.1, B.2, B.3, and B.4

With the offsite source not available to all required features, the option would still exist to declare all required features inoperable. Since this option may involve undesired administrative efforts, the allowance for sufficiently conservative actions is made. With the required emergency power source inoperable, the minimum required diversity of AC power sources is not available. It is, therefore, required to suspend CORE ALTERATIONS, movement of irradiated fuel assemblies, and operations involving positive reactivity additions. The Required Action to suspend positive reactivity additions does not preclude actions to maintain or increase reactor vessel inventory provided the required SDM is maintained.

Suspension of these activities does not preclude completion of actions to establish a safe conservative condition. These actions minimize the probability or the occurrence of postulated events. It is further required to immediately initiate action to restore the required AC sources and to continue this action until restoration is accomplished in order to provide the necessary AC power to the unit safety systems.

BASES

ACTIONS

A.2.1, A.2.2, A.2.3, A.2.4, B.1, B.2, B.3, and B.4 (continued)

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required AC electrical power sources should be completed as quickly as possible in order to minimize the time during which the unit safety systems may be without sufficient power.

Pursuant to LCO 3.0.6, the Distribution System's ACTIONS are not entered even if all AC sources to it are inoperable, resulting in de-energization. Therefore, the Required Actions of Condition A are modified by a Note to indicate that when Condition A is entered with no AC power to any required main feeder bus, the ACTIONS for LCO 3.8.9 must be immediately entered. This Note allows Condition A to provide requirements for the loss of the offsite source, whether or not a train is de-energized. LCO 3.8.9 provides the appropriate restrictions for the situation involving a de-energized required main feeder bus.

SURVEILLANCE REQUIREMENTS

SR 3.8.2.1

SR 3.8.2.1 requires the SRs from LCO 3.8.1 that are necessary for ensuring the OPERABILITY of the AC sources in other than MODES 1, 2, 3, and 4. SR 3.8.1.7, SR 3.8.1.13, SR 3.8.1.14, SR 3.8.1.15 and SR 3.8.1.16 are not required to be met. SR 3.8.1.7 verifies both KHUs cannot be tied to the underground emergency power path simultaneously. This SR verifies train independence to prevent a single failure from disabling both KHUs. This SR is not required to be met in MODES 5 and 6 and during movement of irradiated fuel assemblies, because single failure protection is not required in these MODES. SR 3.8.1.13 requires verification that on an actual or simulated zone overlap signal each KHU's overhead tie breaker and underground tie breaker actuate to the correct position. This SR verifies redundancy between the KHU's in the ability to connect to the underground emergency power path. This redundancy is not required in MODES 5 and 6. SR 3.8.1.14 requires verification that each closed SL and closed N breaker opens on an actuation of each redundant trip coil. This SR verifies each trip circuit for each breaker independently opens each breaker. This SR is not required to be met in MODES 5 and 6 and during movement of irradiated fuel assemblies, because there is no requirement for the automatic transfer function to be OPERABLE when the Unit is in these MODES. SR 3.8.1.15 requires verification that each 230 kV

BASES

SURVEILLANCE REQUIREMENTS

3.8.2.1 (continued)

switchyard circuit breaker actuates to the correct position on an actual or simulated switchyard isolation actuation signal. This SR is not required to be met in MODES 5 and 6 and during movement of irradiated fuel assemblies, because there is no requirement for the switchyard isolation function to be OPERABLE when the Unit is in these MODES. SR 3.8.1.16 verifies that one KHU provides an alternate manual AC power source capability by manual or automatic KHU start with manual synchronize, or breaker closure, to energize its non-required emergency power path. This SR is not required to be met in MODES 5 and 6 and during movement of irradiated fuel assemblies, because there is no requirement for providing this capability when the Unit is in these MODES.

The SR is modified by two Notes. Note 1 indicates that SR requirements to energize both standby buses may be reduced to require energizing only one standby bus and one main feeder bus. Reduced OPERABILITY requirements associated with MODES 5 and 6 and during movement of irradiated fuel may permit a reduction in requirements for energizing portions of the AC distribution system. Note 2 indicates that the SR 3.8.1.4 requirement to energize the underground power path is not applicable since the performance of this portion of the SR is only appropriate when both emergency power paths are required to be OPERABLE.

REFERENCES

1. 10 CFR 50.36.

# B 3.8 ELECTRICAL POWER SYSTEMS

## B 3.8.3 DC Sources – Operating

BASES

BACKGROUND

The 125 VDC Vital I&C electrical power sources provide the AC emergency power system with control power. It also provides both motive power and control power for selected safety related equipment. Additionally, the 125 VDC Vital I&C electrical power sources provide DC electrical power through DC panelboards to the inverters, which in turn supply the AC Vital instrumentation power panelboards.

The 125 VDC Vital I&C electrical power system is a system consisting of six power sources shared by the three Oconee units. Each unit has its own two power sources with backup sources supplied to the unit's 125 VDC Vital Instrumentation distribution system from another unit using a network of isolating diode assemblies. This provides necessary redundancy and independence for the 125 VDC Vital I&C power sources. Each source consists of one 125 VDC battery, the associated battery charger for each battery, the distribution center, the associated control equipment, isolating transfer diodes and interconnecting cabling. Additionally, there is one standby battery charger shared between each unit's batteries, which provides backup service in the event that the preferred battery charger is out of service.

The 125 VDC I&C batteries of a unit are physically separated in separate enclosures from batteries of another unit to minimize their exposure to any damage. The battery chargers and associated DC distribution centers and switchgear of a unit are located in separate rooms from the battery chargers and associated DC distribution centers of another unit in the auxiliary building and physical separation is maintained between redundant equipment.

During normal operation, the 125 VDC Vital I&C loads are powered from the battery chargers with the batteries floating on the system. In case of loss of power to a battery charger, the associated DC loads are automatically powered from the 125 VDC Vital I&C battery. Each battery has adequate storage capacity to carry the required load continuously for at least 1 hour.

BASES

BACKGROUND (continued)

Each 125 VDC Vital I&C power source has ample power output capacity for the steady state operation of connected loads required during normal operation, while at the same time maintaining its battery bank fully charged. Each battery charger also has sufficient capacity to restore the battery from the design minimum charge to its fully charged state while supplying normal steady state loads.

The 230 kV switchyard 125 VDC Power System provides power to power circuit breakers, protective and control relays, indicating lights, annunciators, carrier equipment and other switchyard equipment requiring an uninterrupted power source.

The 230 kV switchyard 125 VDC Power System consists of two sources. Each source consists of one 125 VDC battery, the associated battery charger for each battery, distribution panel, and associated control equipment and interconnecting cabling. Redundant batteries are located in separate rooms and redundant chargers, distribution centers and panelboards are located on different walls of the 230 kV switchyard relay house. Additionally, there is one standby battery charger shared between the sources, which provides backup service in the event that the preferred battery charger is out of service.

During normal operation, the 230 kV 125 VDC loads are powered from the battery chargers with the batteries floating on the system. In case of loss of power to a battery charger, the associated DC load is automatically powered from the 230 kV 125 VDC battery. Each battery has adequate storage capacity to carry the required load continuously for at least 1 hour.

Each 230 kV 125 VDC power source has ample power output capacity for the steady state operation of connected loads required during normal operation, while at the same time maintaining its battery bank fully charged. Each battery charger also has sufficient capacity to restore the battery from the design minimum charge to its fully charged state while supplying normal steady state loads.

The 125 VDC Vital I&C power and 230 kV 125 VDC power distribution systems are described in more detail in the Bases for LCO 3.8.8, "Distribution System – Operating," and for LCO 3.8.9, "Distribution Systems – Shutdown."

BASES (continued)

APPLICABLE SAFETY ANALYSES

The initial conditions of accident and transient analyses in the UFSAR, Chapter 6 (Ref. 1) and Chapter 15 (Ref. 2), assume that Engineered Safeguards (ES) systems are OPERABLE. The 125 VDC Vital I&C electrical power system provides normal and emergency DC electrical power for the emergency auxiliaries, and control and switching during all MODES of operation.

The 230 kV switchyard 125 VDC Power System provides control power for circuit breaker operation in the 230 kV switchyard as well as DC power for degraded grid voltage protection circuits during all MODES of operation.

The OPERABILITY of the DC sources is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the unit. This includes maintaining the DC sources OPERABLE during accident conditions in the event of:

a. An assumed loss of all offsite AC power or all onsite AC power; and

b. A worst-case single failure.

The DC sources satisfy Criterion 3 of 10 CFR 50.36 (Ref. 3).

LCO

Each required 125 VDC electrical source consisting of one battery, associated battery charger, distribution center and the corresponding control equipment and interconnecting cabling supplying power to the associated panelboards is required to be OPERABLE to ensure the availability of the required power to shut down the reactor and maintain it in a safe condition after an anticipated transient or an accident.

For operation of any Oconee unit, three of four 125 VDC Vital I&C Sources capable of supplying the unit's DC distribution system shall be OPERABLE as follows:

Unit 1: 1CA, 1CB, 2CA, 2CB
Unit 2: 2CA, 2CB, 3CA, 3CB
Unit 3: 3CA, 3CB, 1CA, 1CB

and aligned to at least one panelboard provided that a power source is not the only source for two or more of the Unit's panelboards. The three of four requirement ensures that a single failure will not result in a loss of

BASES

LCO (continued)

power to more than one 125 VDC Vital I&C panelboard. This requirement ensures supported safety functions are not vulnerable to a single failure.

When any other unit is in MODES 1, 2, 3, or 4, two additional 125 VDC Vital I&C Sources are required to be OPERABLE as modified by LCO Note 2. When no other Unit is in MODES 1, 2, 3, or 4, one additional 125 VDC Vital I&C power source is required to be OPERABLE as modified by LCO Notes 2 and 3. These additional requirements ensure sufficient capacity and voltage for supported DC loads assuming a single failure.

The requirement that two 230 kV 125 VDC sources be OPERABLE ensures that supported safety functions are not vulnerable to a single failure.

The LCO is modified by three Notes. Note 1, which applies to Units 2 and 3 only, indicates that no single 125 VDC Vital I&C source shall be the only source for panelboards 1DIC and 1DID. This is necessary since vital I&C panelboards 1DIC and 1DID supply power for SK and SL breaker control, protective relaying for both standby buses, breaker control for both standby breakers for the three Oconee units, and retransfer to startup source logic circuits for the three Oconee units. The requirement that no single 125 VDC source be the only source of power for panelboards 1DIC and 1DID ensures than a single failure will not result in a loss of power to both panelboards. This requirement ensures supported safety functions are not vulnerable to a single failure.

Note 2 indicates that each additional 125 VDC Vital I&C source required by part b or part c of the LCO shall be connected to at least one panelboard associated with the unit where the source is physically located. For example, when applying the LCO requirements to Unit 1, an additional source from Unit 2 must be connected to at least one Unit 2 panelboard and an additional source from Unit 3 must be connected to at least one Unit 3 panelboard. If the additional sources are from Unit 3, each additional source need only be connected to at least one Unit 3 panelboard. Note 3 specifies that the additional 125 VDC Vital I&C power source required by LCO 3.8.3 part c shall not be a power source that is available to meet the three of four requirement of LCO 3.8.3 part a. This ensures that there is one source physically located on each unit not in MODES 1, 2, 3, or 4. For example, when applying the LCO requirements to Unit 1, the additional source cannot be a Unit 1 or Unit 2 power source since these are available to meet the three of four requirement. Therefore, a Unit 3 power source must be OPERABLE. Note 2 and 3 requirements are necessary to assure assumptions in the DC capacity and voltage drop analyses for the operating unit are valid.

BASES (continued)

APPLICABILITY

The DC electrical power sources are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure safe unit operation and to ensure that:

a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of transients and accidents; and

b. Adequate core cooling is provided, and containment integrity and other vital functions are maintained in the event of a postulated accident.

The DC electrical power requirements for MODES 5 and 6 are addressed in the Bases for LCO 3.8.4, "DC Sources – Shutdown."

ACTIONS

The ACTIONS are modified by a Note indicating that the Completion Times for Required Actions A through D are reduced when in Condition L of LCO 3.8.1. Condition L limits the Completion Time for restoring inoperable power sources to 4 hours when emergency power source(s) or offsite power source(s) are inoperable for extended time periods or for specific reasons.

A.1

With one of the required 125 VDC Vital I&C sources inoperable, the remaining sources are fully capable of providing adequate voltage to the four unit DC panelboards and will assure alignment of power to at least three panelboards. Three panelboards are necessary to shut down the operating unit and maintain it in a safe shutdown condition. However, overall reliability is reduced because an additional failure could result in the minimum required ES functions not being supported. Therefore, the inoperable source must be restored to OPERABLE status within 24 hours. Required Action A.1 is modified by a Note indicating that it is not applicable for up to 72 hours to perform an equalization charge after completion of a performance test or service test. This note allows a maximum Completion Time of 96 hours (24 hours for an inoperable battery due to performing a service test plus 72 hours to perform equalization charge).

The Completion Time for this Required Action is based on engineering judgment, taking into consideration the extent of degradation involved, the likelihood of events or failures which could challenge the system, and the time required to complete the equalization charge.

BASES

ACTIONS (continued)

B.1

In this condition, a single failure of a battery (or its associated equipment) could cause loss of more than one unit panelboard during an accident, so that required safety functions might not be supported. Specifically, if a single source were providing the only power source for panelboards DIA and DIB, single failure of the source would result in failure of both ES digital channels. Vulnerability of the ES digital channels to single failure for 24 hours is considered acceptable due to the limited scope of potential failures. Similarly, if the panelboards are isolated from their backup Unit (e.g., the Unit's DC system is isolated from the other Units), a single failure could result in loss of two or more panelboards so that required safety functions may not be supported. If the panelboards are isolated from their backup Unit when one of that Unit's batteries are inoperable (and the DC buses are cross tied), the remaining battery has the capacity to support all required loads, however, a single failure could result in loss of all four panelboards so that required safety functions may not be supported. Therefore, within 24 hours after such a condition arises, affected equipment shall be restored and aligned such that no single source is the only battery power supply for more than one 125 VDC Vital I&C panelboard for the unit under consideration. The 24 hour Completion Time is based on engineering judgement taking into consideration the time to complete the Required Action and the redundancy available in the 125 VDC Vital I&C System.

C.1

With a single source providing the only power supply for 125 VDC Vital I&C panelboards 1DIC and 1DID, a single failure of a battery (or its associated equipment) could cause loss of both panelboards, so that required automatic EPSL functions for all three units may not be supported. These panelboards provide primary and backup control power for the SK and SL breaker control power, standby bus protective relaying, standby breaker control power and retransfer to startup logic. Therefore, within 24 hours after such a condition arises, affected equipment shall be restored and aligned such that no single source is the only battery power supply for both DC panelboards 1DIC and 1DID.

The Completion Time is based on engineering judgement, provides a reasonable time to complete repairs and considers the redundancy available in the 125 VDC Vital I&C DC System.

This Condition is modified by a Note indicating that this ACTION is only applicable to Units 2 and 3. For Unit 1 the appropriate action is specified in ACTION B.

BASES

ACTIONS (continued)

D.1

With one of the required 230 kV switchyard DC power sources inoperable, the remaining source is fully capable of providing adequate voltage to the associated panelboards and is fully capable of powering the necessary panelboards. However, another failure of a DC source or panelboard could result in failure of the overhead emergency power path. In addition, in the event of grid voltage degradation the station and onsite emergency power sources could fail to separate from the grid. Therefore, the inoperable source must be restored to OPERABLE status within 24 hours. Required Action D.1 is modified by a Note indicating that it is not applicable for up to 72 hours to perform an equalization charge after completion of a performance test or service test. This note allows a maximum Completion Time of 96 hours (24 hours for an inoperable battery due to performing a service test plus 72 hours to perform equalization charge).

The Completion Time for this Required Action is based on engineering judgment, taking into consideration the extent of degradation involved, the likelihood of events or failures which could challenge the system, and the time required to complete the required actions.

E 1 and E.2

If the inoperable DC electrical power source cannot be restored to OPERABLE status within the required Completion Time, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 12 hours and to MODE 5 within 84 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.8.3.1

This Surveillance verifies that the distribution centers are functioning properly, with the correct circuit breaker alignment to the isolating transfer diodes. The correct breaker alignment ensures the appropriate separation and independence is maintained, and the appropriate voltage is available to each required isolating transfer diode. The verification of

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.3.1 (continued)

proper voltage availability on the distribution centers ensures that the required voltage is readily available for isolating transfer diodes connected to these distribution centers. The 7 day Frequency takes into account the redundant capability of the DC electrical power distribution systems, and other indications available in the control room that alert the operator to system malfunctions.

SR 3.8.3.2

Verifying battery terminal voltage while on float charge for the batteries helps to ensure the effectiveness of the charging system and the ability of the batteries to perform their intended function. Float charge is the condition in which the charger is supplying the continuous charge required to overcome the internal losses of a battery (or battery cell) and maintain the battery (or a battery cell) in a fully charged state. The voltage requirements are based on the nominal design voltage of the battery and are consistent with the initial voltages assumed in the battery sizing calculations. The 7 day Frequency is consistent with manufacturer recommendations and IEEE-450 (Ref. 5).

SR 3.8.3.3

Visual inspection of the battery cells, cell plates, and battery racks provides an indication of physical damage or abnormal deterioration that could potentially degrade battery performance. The presence of physical damage or deterioration does not necessarily represent a failure of this SR, provided an evaluation determines that the physical damage or deterioration does not affect the OPERABILITY of the battery (its ability to perform its design function).

The 12 month Frequency for this SR is consistent with IEEE-450 (Ref. 5), which recommends detailed visual inspection of cell condition and rack integrity on a yearly basis.

SR 3.8.3.4

Visual inspection of inter-cell, inter-rack, inter-tier, and terminal connections provide an indication of physical damage or abnormal deterioration that could indicate degraded battery condition. The

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.3.4 (continued)

anticorrosion material is used to help ensure good electrical connections and to reduce terminal deterioration. The visual inspection for corrosion is not intended to require removal of and inspection under each terminal connection.

The Surveillance Frequencies of 12 months are consistent with IEEE-450 (Ref. 5), which recommends cell to cell and terminal connection visual inspection on a yearly basis.

SR 3.8.3.5

A battery service test is a special test of the battery capability, as found, to satisfy the design requirements (battery duty cycle) of the DC electrical power system. The discharge rate and test length should correspond to the design duty cycle requirements as specified in Reference 4.

The Surveillance Frequency of 12 months is consistent with the recommendations of Regulatory Guide 1.32 (Ref. 6) and Regulatory Guide 1.129 (Ref. 7), which state that the battery service test should be performed during refueling operations, or at some other outage, with intervals between tests not to exceed 18 months.

SR 3.8.3.6

This SR requires battery capacity be verified in accordance with the Battery Discharge Testing Program. A battery performance discharge test is a test of constant current capacity of a battery, normally done in the as found condition, after having been in service, to detect any change in the capacity determined by the acceptance test.

The test is intended to determine overall battery degradation due to age and usage.

The Surveillance Frequencies for this test are in accordance with the Battery Discharge Testing Program and are consistent with the recommendations in IEEE-450 (Ref. 5). These periodic frequencies are based on the outcome of the previous battery capacity test.

BASES (continued)

REFERENCES

1. UFSAR, Chapter 6.
2. UFSAR, Chapter 15.
3. 10 CFR 50.36.
4. UFSAR, Chapter 8.
5. IEEE-450-1987.
6. Regulatory Guide 1.32, February 1977.
7. Regulatory Guide 1.129, December 1974.

# B 3.8 ELECTRICAL POWER SYSTEMS

## B 3.8.4 DC Sources – Shutdown

BASES

BACKGROUND

A description of the 125 VDC Vital I&C sources is provided in the Bases for LCO 3.8.3, "DC Sources – Operating."

APPLICABLE SAFETY ANALYSES

The initial conditions of Accidents and transients analyses in the UFSAR, Chapter 6 (Ref. 1) and Chapter 15 (Ref. 2), assume that Engineered Safeguard (ES) systems are OPERABLE. The 125 VDC Vital I&C electrical power system provides normal and emergency DC electrical power for the emergency auxiliaries, and control and switching during all MODES of operation.

Although the 230 kV Switchyard 125 VDC Power System provides control power for circuit breaker operation in the 230 kV switchyard as well as DC power for degraded grid voltage protection circuits during all MODES of operation, no credit is taken for these functions in MODES 5 and 6.

The OPERABILITY of the 125 VDC Vital I&C sources is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY.

The OPERABILITY of the minimum 125 VDC Vital I&C electrical power sources during MODES 5 and 6 and during movement of irradiated fuel assemblies ensures that:

a. The unit can be maintained in the shutdown or refueling condition for extended periods;

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and

c. Adequate DC electrical power is provided to mitigate events postulated during shutdown, such as a fuel handling accident.

The 125 VDC Vital I&C sources satisfy Criterion 3 of 10 CFR 50.36 (Ref. 3).

BASES (continued)

LCO

The 125 VDC Vital I&C electrical power sources, each source consisting of one battery, one battery charger, and the corresponding control equipment and interconnecting cabling within the source, are required to be OPERABLE to support required distribution systems required OPERABLE by LCO 3.8.9, "Distribution Systems – Shutdown" and shall include at least one of the unit's 125 VDC Vital I&C power sources. This ensures the availability of sufficient 125 VDC Vital I&C electrical power sources to operate the unit in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents).

APPLICABILITY

The 125 VDC Vital I&C electrical power sources required to be OPERABLE in MODES 5 and 6 and during movement of irradiated fuel assemblies, provide assurance that:

a. Required features to provide adequate coolant inventory makeup are available for the irradiated fuel assemblies in the core;

b. Required features needed to mitigate a fuel handling accident are available;

c. Required features necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

The 125 VDC Vital I&C electrical power requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.3.

ACTIONS

A.1, A.2.1, A.2.2, A.2.3, and A.2.4

If two or more 125 VDC Vital I&C panelboards are required by LCO 3.8.9, the remaining 125 VDC Vital I&C panelboards with 125 VDC Vital I&C power available may be capable of supporting sufficient systems to allow continuation of CORE ALTERATIONS and fuel movement. By allowing the option to declare required features inoperable with the associated 125 VDC Vital I&C power source(s) inoperable, appropriate restrictions

BASES

ACTIONS

A.1, A.2.1, A.2.2, A.2.3, and A.2.4 (continued)

will be implemented in accordance with the affected required features LCO ACTIONS. In many instances this option may involve undesired administrative efforts. Therefore, the allowance for sufficiently conservative actions is made (i.e., to suspend CORE ALTERATIONS, movement of irradiated fuel assemblies, and operations involving positive reactivity additions). The Required Action to suspend positive reactivity additions does not preclude actions to maintain or increase reactor vessel inventory, provided the required SDM is maintained.

Suspension of these activities shall not preclude completion of actions to establish a safe conservative condition. These actions minimize probability of the occurrence of postulated events. It is further required to immediately initiate action to restore the required 125 VDC Vital I&C electrical power sources and to continue this action until restoration is accomplished in order to provide the necessary 125 VDC Vital I&C electrical power to the unit safety systems.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required 125 VDC Vital I&C electrical power sources should be completed as quickly as possible in order to minimize the time during which the unit safety systems may be without sufficient power.

SURVEILLANCE REQUIREMENTS

SR 3.8.4.1

SR 3.8.4.1 requires performance of all Surveillances required by SR 3.8.3.1 through SR 3.8.3.6. Therefore, see the corresponding Bases for LCO 3.8.3 for a discussion of each SR.

This SR is modified by a Note. The reason for the Note is to preclude requiring the OPERABLE 125 VDC Vital I&C sources from being discharged below their capability to provide the required power supply or otherwise rendered inoperable during the performance of SRs. It is the intent that these SRs must still be capable of being met, but actual performance is not required.

BASES (continued)

REFERENCES

1. UFSAR, Chapter 6.
2. UFSAR, Chapter 15.
3. 10 CFR 50.36.

## B 3.8 ELECTRICAL POWER SYSTEMS

### B 3.8.5 Battery Cell Parameters

BASES

BACKGROUND

This LCO delineates the limits on electrolyte temperature, level, float voltage, and specific gravity for the KHU, 125 VDC Vital I&C, and 230 kV 125 VDC power source batteries. A discussion of these batteries and their OPERABILITY requirements is provided in the Bases for LCO 3.8.1, "AC Sources Operating, " LCO 3.8.2, "AC Sources – Shutdown, LCO 3.8.3, "DC Sources – Operating," and LCO 3.8.4, "DC Sources – Shutdown."

APPLICABLE SAFETY ANALYSES

The initial conditions of Accident and transient analyses in the UFSAR, Chapter 6 (Ref. 1) and Chapter 15 (Ref. 2), assume Engineered Safeguards (ES) systems are OPERABLE. The 125 VDC Vital I&C electrical power system provides normal and emergency DC electrical power for the emergency auxiliaries, and control and switching during all MODES of operation.

Although the 230 kV Switchyard 125 VDC Power System provides control power for circuit breaker operation in the 230 kV switchyard as well as DC power for degraded grid voltage protection circuits during all MODES of operation, no credit is taken for these functions in MODES 5 and 6.

Each Keowee Hydro Unit (KHU) includes a 125 VDC power source to supply power to DC auxiliary loads and the Keowee Emergency Start circuits.

The OPERABILITY of the DC sources is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the unit. This includes maintaining required DC sources OPERABLE during accident conditions, in the event of:

a. An assumed loss of all offsite AC power or all onsite AC power; and

b. A worst-case single failure.

Battery cell parameters satisfy Criterion 3 of 10 CFR 50.36 (Ref. 3).

BASES (continued)

LCO

Battery cell parameters must remain within acceptable limits to ensure availability of the required DC power to shut down the reactor and maintain it in a safe condition after a transient or a postulated accident. Electrolyte limits are conservatively established, allowing continued DC electrical system function even with Category A and B limits not met.

APPLICABILITY

The battery cell parameters are required solely for the support of the associated DC electrical power sources. Therefore, battery cell parameters are only required to be met when the DC power source is required to be OPERABLE. Refer to the Applicability discussion in Bases for LCO 3.8.1, LCO 3.8.2, LCO 3.8.3 and LCO 3.8.4.

ACTIONS

The ACTIONS Table is modified by a Note which indicates that separate Condition Entry is allowed for each battery. This is acceptable, since the Required Actions for each Condition provides appropriate compensatory actions for each inoperable DC source. Complying with the Required Actions for one inoperable DC source may allow for continued operation, and subsequent inoperable DC sources(s) are governed by separate Condition entry and application of associated Required Actions.

A second Note states that LCO 3.0.4 is not applicable. This is acceptable since a battery remains OPERABLE when one or more cells does not meet Category A or B limits but continues to meet Category C limits. Failure to meet Category C limits requires declaring the associated battery inoperable. LCO 3.0.4 requirements are applicable to the requirements of LCO 3.8.3, "DC Sources – Operating" for an inoperable battery.

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

With one or more cells in one or more batteries not within limits (i.e., Category A limits not met or Category B limits not met or Category A and B limits not met) but within the Category C limits specified in Table 3.8.5-1 in the accompanying LCO, the battery is degraded but there is still sufficient capacity to perform the intended function. Therefore, the affected battery is not required to be considered inoperable solely as a result of Category A or B limits not met, and continued operation is permitted for a limited period.

BASES

ACTIONS

A.1, A.2, and A.3 (continued)

The pilot cell electrolyte level and float voltage are required to be verified to meet the Category C limits within 1 hour (Required Action A.1). This check will provide a quick indication of the status of the remainder of the battery cells. One hour provides time to inspect the electrolyte level and to confirm the float voltage of the pilot cells. One hour is considered a reasonable amount of time to perform the required verification.

Verification that the Category C limits are met (Required Action A.2) provides assurance that during the time needed to restore the parameters to the Category A and B limits, the battery will still be capable of performing its intended function. A period of 24 hours is allowed to complete the initial verification because specific gravity measurements must be obtained for each connected cell. Taking into consideration both the time required to perform the required verification and the assurance that the battery cell parameters are not severely degraded, this time is considered reasonable. The verification is repeated at 7 day intervals until the parameters are restored to Category A and B limits. This periodic verification is consistent with the normal Frequency of pilot cell Surveillances.

Continued operation is only permitted for 90 days before battery cell parameters must be restored to within Category A and B limits. With the consideration that, while battery capacity is degraded, sufficient capacity exists to perform the intended function and to allow time to fully restore the battery cell parameters to normal limits, this time is acceptable prior to declaring the battery inoperable.

B.1

With the Required Action and associated Completion Time not met, or with one or more batteries with one or more battery cell parameters outside the Category C limit for any connected cell, or with the average electrolyte temperature of representative cells falling below 60°F, sufficient capacity to supply the maximum expected load requirement is not assured and the corresponding DC electrical power source must be declared inoperable immediately.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.8.5.1

This SR verifies that Category A battery cell parameters are consistent with IEEE-450 (Ref. 4), which recommends regular battery inspections including voltage, specific gravity, and electrolyte temperature of pilot cells.

SR 3.8.5.2

The quarterly inspection of specific gravity and voltage is consistent with IEEE-450 (Ref. 4).

SR 3.8.5.3

This Surveillance verification that the average temperature of representative cells is ≥ 60°F is consistent with a recommendation of IEEE-450 (Ref. 4), which states that the temperature of electrolytes in representative cells should be determined on a quarterly basis.

Lower than normal temperatures act to inhibit or reduce battery capacity. This SR ensures that the operating temperatures remain within an acceptable operating range. This limit is based on plant specific calculations.

Table 3.8.5-1

This table delineates the limits on electrolyte level, float voltage, and specific gravity for three different categories. The meaning of each category is discussed below.

Category A defines the normal parameter limit for each designated pilot cell in each battery. The cells selected as pilot cells are those whose temperature, voltage and electrolyte specific gravity are considered to approximate the state of charge of the entire battery.

The Category A limits specified for electrolyte level are based on manufacturer recommendations and are consistent with the guidance in IEEE-450 (Ref. 4), with the extra ¼ inch allowance above the high water level indication for operating margin to account for temperatures and charge effects. In addition to this allowance, footnote a to Table 3.8.5-1 permits the electrolyte level to be above the specified maximum level during equalizing charge, provided it is not overflowing. These limits

BASES

SURVEILLANCE REQUIREMENTS

TABLE 3.8.5-1 (continued)

ensure that the plates suffer no physical damage and that adequate electron transfer capability is maintained in the event of transient conditions. IEEE-450 (Ref. 4) recommends that electrolyte level readings should be made only after the battery has been at float charge for at least 72 hours.

The Category A limit specified for float voltage is ≥ 2.13 V per cell. This value is based on a recommendation of IEEE-450 (Ref. 4), which states that prolonged operation of cells < 2.13 V can reduce the life expectancy of cells.

The Category A limit specified for specific gravity for each pilot cell is ≥ 1.200 (0.015 below the manufacturer fully charged nominal specific gravity or a battery charging current that had stabilized at a low value). This value is characteristic of a charged cell with adequate capacity. According to IEEE-450 (Ref. 4), the specific gravity readings are based on a temperature of 77°F (25°C).

The specific gravity readings are corrected for actual electrolyte temperature and level. For each 3°F (1.67°C) above 77°F (25°C), 1 point (0.001) is added to the reading; 1 point is subtracted for each 3°F below 77°F. The specific gravity of the electrolyte in a cell increases with a loss of water due to electrolysis or evaporation.

Category B defines the normal parameter limits for each connected cell. The term "connected cell" excludes any battery cell that may be jumpered out.

The Category B limits specified for electrolyte level, float voltage and specific gravity are the same as those specified for Category A and have been discussed above. In addition to that limit, it is required that the specific gravity for each connected cell must be no less than 0.010 below the average of all connected cells. This limit ensures that the effect of a highly charged or new cell does not mask overall degradation of the battery.

Category C defines the limits for each connected cell. These values, although reduced, provide assurance that sufficient capacity exists to perform the intended function and maintain a margin of safety. When any battery parameter is outside the Category C limits, the assurance of sufficient capacity described above no longer exists and the battery must be declared inoperable.

BASES

SURVEILLANCE REQUIREMENTS

Table 3.8.5-1 (continued)

The Category C limits specified for electrolyte level (above the top of the plates and not overflowing) ensure that the plates suffer no physical damage and maintain adequate electron transfer capability. The Category C limits for float voltage is based on IEEE-450 (Ref. 4), which states that a cell voltage of 2.07 V or below, under float conditions and not caused by elevated temperature of the cell, indicates internal cell problems and may require cell replacement.

The Category C limits for specific gravity is the same as the limits specified for Category A and has been discussed above.

The footnotes to Table 3.8.5-1 are applicable to Category A, B, and C specific gravity. Footnote (b) to Table 3.8.5-1 requires the above mentioned correction for electrolyte level and temperature, with the exception that level correction is not required when battery float current is < 2 amps on float charge. This current provides, in general, an indication of overall battery condition.

Because of specific gravity gradients that are produced during the recharging process, delays of several days may occur while waiting for the specific gravity to stabilize. A stabilized charger current is an acceptable alternative to specific gravity measurement for determining the state of charge. This phenomenon is discussed in IEEE-450 (Ref. 4). Footnote (c) to Table 3.8.5-1 allows the float (charger) current to be used as an alternate to specific gravity for up to 7 days following a battery recharge. When battery float current is verified in lieu of specific gravity, the specific gravity of each connected cell shall be measured prior to expiration of the 7 day allowance. Within 7 days each connected cell's specific gravity must be measured to confirm the state of charge. Following a minor battery recharge (such as equalizing charge that does not follow a deep discharge) specific gravity gradients are not significant, and confirming measurements may be made in less than 7 days.

REFERENCES

1. UFSAR, Chapter 6.

2. UFSAR, Chapter 15.

3. 10 CFR 50.36.

4. IEEE-450-1995.

## B 3.8 ELECTRICAL POWER SYSTEMS

### B 3.8.6 Vital Inverters – Operating

BASES

BACKGROUND

The inverters are the preferred source of power for the 120 VAC Vital Instrumentation panelboards because of the stability and reliability they achieve. The function of the inverter is to provide AC electrical power to the 120 VAC Vital Instrumentation panelboard. The panelboards can be powered from an alternate AC source or from the 125 VDC Vital I&C batteries through a 125 VDC Vital I&C Power Panelboard and the inverters. The inverters provide an uninterrupted power source for the instrumentation and controls for the Reactor Protective System (RPS) and the Engineered Safeguards (ES) System.

APPLICABLE SAFETY ANALYSES

The initial conditions of Accident and transient analyses in the UFSAR, Chapter 6 (Ref. 1), and Chapter 15 (Ref. 2), assume Engineered Safeguards (ES) systems are OPERABLE. The inverters are designed to provide the required capacity, capability, redundancy, and reliability to ensure the availability of necessary power to the RPS and ES instrumentation and controls so that the fuel, Reactor Coolant System, and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for Section 3.2, Power Distribution Limits; Section 3.4, Reactor Coolant System (RCS); and Section 3.6, Containment Systems.

The OPERABILITY of the inverters is consistent with the initial assumptions of the accident analyses and is based on meeting the design basis of the unit. This includes maintaining required 120 VAC Vital Instrumentation panelboards OPERABLE during accident conditions in the event of:

a. An assumed loss of all offsite AC electrical power or all onsite AC electrical power; and

b. A worst-case single failure.

Inverters are a part of the distribution system and, as such, satisfy Criterion 3 of 10 CFR 50.36 (Ref. 3).

BASES (continued)

LCO

The inverters ensure the availability of AC electrical power for the systems instrumentation required to shut down the reactor and maintain it in a safe condition after a transient or a postulated accident.

Maintaining the required inverters OPERABLE ensures that the redundancy incorporated into the design of the RPS and ES instrumentation and controls is maintained. The four inverters ensure an uninterrupted supply of AC electrical power to the 120 VAC Vital Instrumentation panelboards even if the 4.16 kV buses are de-energized.

OPERABLE inverters require the associated Vital Instrumentation panelboards to be powered by the inverter with output voltage and frequency within tolerances, and power input to the inverter from a 125 VDC Vital I&C source.

APPLICABILITY

The inverters are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure that:

a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of transients and accidents; and

b. Adequate core cooling is provided, and containment OPERABILITY and other vital functions are maintained in the event of a postulated accident.

Inverter requirements for MODES 5 and 6 are covered in the Bases for LCO 3.8.7, "Inverters – Shutdown."

ACTIONS

A.1

With a required inverter inoperable, its associated 120 VAC Vital Instrumentation panelboard becomes inoperable until it is manually re-energized from its alternate regulated voltage source.

For this reason, Note 1 has been included for Required Action A.1 requiring entry into the Conditions and Required Actions of LCO 3.8.8, "Distribution Systems – Operating." This ensures the vital bus is re-energized within either 4 or 24 hours. Required Action A.1 allows 7 days to fix the inoperable inverter and return it to service. The 7 day limit is

BASES

ACTIONS

A.1 (continued)

based upon engineering judgment, taking into consideration the time required to repair an inverter and the additional risk to which the unit is exposed because of the inverter inoperability. This has to be balanced against the risk of an immediate shutdown, along with the potential challenges to safety systems such a shutdown might entail. When the 120 VAC Vital Instrumentation panelboard is powered from its regulated voltage source, it is relying upon interruptible AC electrical power sources (offsite and onsite). The uninterruptible inverter source to the 120 VAC Vital Instrumentation panelboards is the preferred source for powering instrumentation trip setpoint devices.

Required Action A.1 is also modified by Note 2 which indicates that the Completion Time is reduced when in Condition L of LCO 3.8.1. Condition L limits the Completion Time for restoring an inoperable vital inverter to 4 hours when emergency power source(s) or offsite power source(s) are inoperable for extended time periods or for specific reasons.

B.1 and B.2

If the Required Action and associated Completion Time are 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 12 hours and to MODE 5 within 84 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.8.6.1

This Surveillance verifies that the inverters are functioning properly with all required circuit breakers closed and 120 VAC Vital Instrumentation panelboards energized from the inverter. The verification of proper voltage and frequency output ensures that the required power is readily available for the instrumentation of the RPS and ES connected to the 120 VAC Vital Instrumentation panelboards. The 7 day Frequency takes into account the redundant capability of the inverters and other indications available in the control room that alert the operator to inverter malfunctions.

BASES (continued)

REFERENCES

1. UFSAR, Chapter 6.
2. UFSAR, Chapter 15.
3. 10 CFR 50.36.

## B 3.8 ELECTRICAL POWER SYSTEMS

### B 3.8.7 Vital Inverters – Shutdown

BASES

BACKGROUND

A description of the inverters is provided in the Bases for LCO 3.8.6, "Inverters – Operating."

APPLICABLE SAFETY ANALYSES

The initial conditions of Accident and transient analyses in the UFSAR, Chapter 6 (Ref. 1) and Chapter 15 (Ref. 2), assume Engineered Safeguards systems are OPERABLE. The DC to AC inverters are designed to provide the required capacity, capability, redundancy, and reliability to ensure the availability of necessary power to the Reactor Protection System and Engineered Safeguards (ES) System instrumentation and controls so that the fuel, Reactor Coolant System, and containment design limits are not exceeded.

The OPERABILITY of the inverters is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY.

The OPERABILITY of the minimum inverters to each 120 VAC Vital Instrumentation panelboards during MODES 5 and 6 ensures that:

a. The unit can be maintained in the shutdown or refueling condition for extended periods;

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and

c. Adequate power is available to mitigate events postulated during shutdown, such as a fuel handling accident.

The inverters were previously identified as part of the distribution system and, as such, satisfy Criterion 3 of 10 CFR 50.36 (Ref. 3).

BASES (continued)

LCO

The inverters ensure the availability of electrical power for the instrumentation for systems required to shut down the reactor and maintain it in a safe condition after a transient or accident. The battery powered inverters provide uninterruptible supply of AC electrical power to the 120 VAC Vital Instrumentation panelboards even if the 4.16 kV buses are de-energized. OPERABILITY of the inverters requires that the 120 VAC Vital Instrumentation panelboard be powered by the inverter. This ensures the availability of sufficient inverter power sources to operate the unit in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents).

APPLICABILITY

The inverters required to be OPERABLE in MODES 5 and 6, and during movement of irradiated fuel assemblies provide assurance that:

a. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel in the core;

b. Systems needed to mitigate a fuel handling accident are available;

c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

Inverter requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.6.

ACTIONS

A.1, A.2.1, A.2.2, A.2.3, and A.2.4

If two or more 120 VAC Vital Instrumentation panelboards are required by LCO 3.8.9, "Distribution Systems – Shutdown," the remaining OPERABLE inverters may be capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS, fuel movement, and operations with a potential for positive reactivity additions. The Required Action to suspend positive reactivity additions does not preclude actions to maintain or increase reactor vessel inventory, provided the required SDM is maintained. By the allowance of the option to declare required features inoperable with the associated inverter(s)

BASES

ACTIONS

A.1, A.2.1, A.2.2, A.2.3, and A.2.4 (continued)

inoperable, appropriate restrictions will be implemented in accordance with the affected required features LCOs' Required Actions. In many instances, this option may involve undesired administrative efforts. Therefore, the allowance for sufficiently conservative actions is made (i.e., to suspend CORE ALTERATIONS, movement of irradiated fuel assemblies, and operations involving positive reactivity additions).

Suspension of these activities shall not preclude completion of actions to establish a safe conservative condition. These actions minimize the probability of the occurrence of postulated events. It is further required to immediately initiate action to restore the required inverters and to continue this action until restoration is accomplished in order to provide the necessary inverter power to the unit safety systems.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required inverters should be completed as quickly as possible in order to minimize the time the unit safety systems may be without power or powered from an alternate regulated voltage source.

SURVEILLANCE REQUIREMENTS

SR 3.8.7.1

This Surveillance verifies that the inverters are functioning properly with all required circuit breakers closed and 120 VAC Vital Instrumentation panelboards energized from the inverter. The verification of proper voltage and frequency output ensures that the required power is readily available for the instrumentation connected to the 120 VAC Vital Instrumentation panelboards. The 7 day Frequency takes into account the redundant capability of the inverters and other indications available in the control room that alert the operator to inverter malfunctions.

REFERENCES

1. UFSAR, Chapter 6.

2. UFSAR, Chapter 15.

3. 10 CFR 50.36.

# B 3.8 ELECTRICAL POWER SYSTEMS

## B 3.8.8 Distribution Systems – Operating

BASES

BACKGROUND

The onsite AC, DC, and AC vital electrical power distribution systems are divided into redundant and independent AC, DC, and AC vital electrical power distribution buses and panelboards.

The electrical power distribution system consists of two 4.16 kV main feeder buses each connected to three 4.16 kV Engineered Safeguards (ES) power strings, and secondary 600 V load centers; and 600 V and 208 V motor control centers. Both main feeder buses can be connected to the offsite sources or the emergency power sources. Upon a loss of power to the normal unit auxiliary transformer, the main feeder buses are transferred to the startup transformer powered from either the offsite sources through the 230 kV switchyard or the overhead emergency power path. If power is not available from the startup transformer, the main feeder buses are transferred to the standby buses powered from either the underground emergency power path or a Lee combustion turbine using a 100 kV transmission line separated from the system grid and offsite loads. Control power for the 4.16 kV breakers is supplied from the 125 VDC Vital I&C batteries. Control power for the circuit breakers in the 230 kV switchyard is provided from the 230 kV Switchyard 125 VDC batteries. Additionally, power to grid voltage protection circuits are also provided from the 230 kV switchyard 125 VDC batteries. Additional description of this system may be found in the Bases for LCO 3.8.1, "AC Sources – Operating," and the Bases for LCO 3.8.3, "DC Sources – Operating."

The 120 VAC Vital Instrumentation panelboards are normally powered from the inverters. The alternate power supply for the vital panelboards is a regulated voltage source and its use is governed by LCO 3.8.6, "Inverters – Operating." Each regulated voltage source is powered from a non-safety related non-load shed source.

There are four 125 VDC Vital I&C panelboards supplying power to DC loads. Each 125 VDC I&C panelboard is connected to two 125 VDC Vital I&C sources through isolating transfer diodes. Upon a loss of power from either source, power is supplied to the panelboard through the redundant source. There are two 230 kV switchyard 125 VDC sources each supplying power to three required DC panelboards.

BASES (continued)

APPLICABLE SAFETY ANALYSES

The initial conditions of accidents and transient analyses in the UFSAR, Chapter 6 (Ref. 1) and Chapter 15 (Ref. 2), assume ES systems are OPERABLE. The AC, DC, and AC vital electrical power distribution systems are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ES systems so that the fuel, Reactor Coolant System, and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for Section 3.2, Power Distribution Limits; Section 3.4, Reactor Coolant System (RCS); and Section 3.6, Containment Systems.

The OPERABILITY of the AC, DC, and AC vital electrical power distribution systems is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the unit. This includes maintaining power distribution systems OPERABLE during accident conditions in the event of:

a. An assumed loss of all offsite power or all onsite AC electrical power; and

b. A worst-case single failure.

The distribution systems satisfy Criterion 3 of the 10 CFR 50.36 (Ref. 4).

LCO

The AC, DC, and AC vital electrical power distribution systems are required to be OPERABLE. To be considered OPERABLE the AC Distribution System must include two energized main feeder buses capable of being automatically powered by a Keowee Hydro Unit. Each main feeder bus is considered OPERABLE if it is energized and connected to at least two ES power strings. Each of the three ES power strings is required to be energized. The three ES power Strings consist of the following:

| | | |
|---|---|---|
| 1A) Switchgear TC | 1B) Switchgear TD | 1C) Switchgear TE |
| 2A) Load Center X8 | 2B) Load Center X9 | 2C) Load Center X10 |
| 3A) 600V MCC XS1 and 1, 2, 3XSF | 3B) 600V MCC XS2 | 3C) 600V MCC XS3 |
| 4A) 208V MCC XS1 and 1, 2, 3XSF | 4B) 208V MCC XS2 | 4C) 208V MCC XS3 |

BASES

LCO (continued)

Each string is considered OPERABLE if it is energized by at least one main feeder bus except when MCC 1, 2, or 3XSF is powered from load center OXSF. These MCCs would not be available during a DBA when powered from load center OXSF and therefore are considered inoperable.

An OPERABLE 125 VDC Vital I&C Distribution System must include energized 125 VDC Vital I&C panelboards DIA, DIB, DIC, and DID. Additionally, for Units 2 and 3 only, Vital I&C panelboards 1DIC and 1DID shall be energized.

To be considered OPERABLE, 230 kV switchyard 125 VDC panelboards DYA, DYB, DYC, DYE, DYF, and DYG must be energized.

An OPERABLE 120 VAC Vital Instrumentation Distribution System must include energized 120 VAC Vital Instrumentation panelboards KVIA, KVIB, KVIC, and KVID.

These distribution systems ensure the availability of AC, DC, and AC vital electrical power for the systems required to shut down the reactor and maintain it in a safe condition after a transient or accident.

Maintaining the AC, DC, and AC vital electrical power distribution systems OPERABLE ensures that the redundancy incorporated into the design of ES is not defeated. Therefore, a single failure within any system or within the electrical power distribution systems will not prevent safe shutdown of the reactor.

An OPERABLE AC electrical power distribution system requires the associated buses, ES power strings, load centers, and motor control centers to be energized to their proper voltages. OPERABLE 125 VDC Vital I&C panelboards require the panelboards to be energized to their proper voltage from either a battery or charger. OPERABLE 120 VAC Vital Instrumentation panelboards require the panelboards to be energized to their proper voltage from the associated inverter via inverted DC voltage or alternate regulated voltage source.

APPLICABILITY

The electrical power distribution systems are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure that:

a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of accident or transients; and

BASES

APPLICABILITY (continued)

b. Adequate core cooling is provided, and containment OPERABILITY and other vital functions are maintained in the event of a postulated DBA.

Electrical power distribution system requirements for MODES 5 and 6 are covered in the Bases for LCO 3.8.9, "Distribution Systems – Shutdown."

ACTIONS

The ACTIONS are modified by a Note indicating that the Completion Times for Required Actions A through F are reduced when in Condition L of LCO 3.8.1. Condition L limits the Completion Time for restoring inoperable power sources to 4 hours when emergency power source(s) or offsite power source(s) are inoperable for extended time periods or for specific reasons.

A.1 and B.1

With one Main Feeder bus inoperable or not connected to two ES power strings or one ES power string inoperable, the remaining portion of the AC electrical power distribution system is capable of supporting the minimum safety functions necessary to shut down the reactor and maintain it in a safe shutdown condition, assuming no single failure. The overall reliability is reduced, however, because a single failure in the remaining portion of the power distribution systems could result in the minimum required ES functions not being supported. Therefore, the required AC buses, ES power strings, load centers, and motor control centers must be restored to OPERABLE status within 24 hours.

Condition A and B's worst scenario is one main feeder bus and one ES power string without AC power. In this Condition, the unit is more vulnerable to a complete loss of AC power. It is, therefore, imperative that the unit operator's attention be focused on minimizing the potential for loss of power to the remaining bus or ES power strings by stabilizing the unit, and on restoring power to the affected bus or ES power string. The 24 hour time limit before requiring a unit shutdown in this Condition is acceptable because of:

a. The potential for decreased safety if the unit operator's attention is diverted from the evaluations and actions necessary to restore power to the affected train to the actions associated with taking the unit to shutdown within this time limit; and

b. The potential for an event in conjunction with a single failure of a redundant component.

BASES

ACTIONS (continued)

C.1

With one of the unit's 125 VDC Vital I&C panelboard inoperable, the remaining 125 VDC Vital I&C panelboards are capable of supporting the minimum safety functions necessary to shutdown the reactor and maintain it in a safe shutdown condition, assuming no additional failure. The overall reliability is reduced, however, because an additional failure in the remaining 125 VDC Vital I&C panelboards could result in the minimum required ES functions not being supported. Therefore, the 125 VDC Vital I&C panelboard must be restored to OPERABLE status within 24 hours by powering the bus from a battery or charger.

Condition C represents one of the unit's 125 VDC Vital I&C panelboard without adequate 125 VDC Vital I&C power; potentially with both the batteries significantly degraded and the associated chargers nonfunctioning. In this situation, the unit is significantly more vulnerable to a complete loss of all 125 VDC Vital I&C power. It is, therefore, imperative that the operator's attention focus on stabilizing the unit, minimizing the potential for loss of power to the remaining panelboard(s) and restoring power to the affected panelboard(s).

This 24 hour limit is longer than Completion Times allowed for some of the components that are without power. Utilizing the LCO 3.0.6 exception to LCO 3.0.2 for components without adequate 125 VDC Vital I&C power, which would have Required Action Completion Times shorter than 24 hours, is acceptable because of:

a. The potential for decreased safety by requiring a change in unit conditions (i.e., requiring a shutdown) while allowing stable operations to continue;

b. The potential for decreased safety by requiring entry into numerous applicable Conditions and Required Actions for components without DC power and not providing sufficient time for the operators to perform the necessary evaluations and actions to restore power to the affected panelboard(s); and

c. The potential for an event in conjunction with a single failure of a redundant component.

BASES

ACTIONS (continued)

D.1

If a required 230 kV switchyard 125 VDC panelboard or combination of required panelboards which are not redundant to each other are inoperable, the required panelboard(s) shall be restored to OPERABLE status within 24 hours. Loss of the remaining distribution center or a redundant panelboard could result in failure of the overhead emergency power path. In addition, in the event of grid degradation, the station and onsite emergency power sources could fail to separate from the grid.

Condition D is modified by two Notes. Note 1 indicates that Separate Condition entry is allowed for each 230 kV switchyard 125 VDC power panelboard. Note 2 indicates that Condition D is not applicable to the following loss of function combinations: DYA and DYE, DYB and DYF, and DYC and DYG.

The 24 hour Completion Time is based on engineering judgement taking into consideration the time to complete the required action, the redundancy available in the 230 kV switchyard 125 VDC system, the redundancy available in the emergency power paths, and the infrequency of an actual grid system degradation.

E.1

With either panelboard 1DIC inoperable or panelboard 1DID inoperable, a single failure of the remaining panelboard would result in failure of control power for the S, SK, and SL breakers, standby bus protective relaying, and retransfer to startup logic. Within 24 hours after such a condition arises, the inoperable panelboard shall be restored. The Completion Time is based on engineering judgement taking into consideration the time to complete the required action and the redundancy available in the Vital I&C DC System and AC electrical power system.

This Condition is modified by a Note indicating that it is only applicable to Units 2 and 3. For Unit 1 the appropriate action is specified in ACTION C.

BASES

ACTIONS (continued)

F.1 and F.2

With one 120 VAC Vital Instrumentation power panelboard inoperable, the remaining three OPERABLE 120 VAC Vital Instrumentation power panelboards are capable of supporting the minimum safety functions necessary to shut down the unit and maintain it in the safe shutdown condition. Overall reliability is reduced, however, since an additional single failure could result in the minimum required functions not being supported. Therefore, the inoperable 120 VAC Vital Instrumentation power panelboard must be restored to OPERABLE status within 4 or 24 hours dependent upon which panelboard is inoperable. The Completion Time for restoring panelboard KVIA or KVIB is limited to 4 hours since these panelboards power the digital Engineered Safeguards Protective System (ESPS) channels and they cannot actuate without power. The Completion Time for restoring KVIC or KVID is 24 hours.

Condition F represents one 120 VAC Vital Instrumentation panelboard without power; potentially both the 125 VDC Vital I&C source and the alternate AC source are nonfunctioning. In this situation the unit is significantly more vulnerable to a complete loss of all 120 VAC Vital Instrumentation panelboards. It is, therefore, imperative that the operator's attention focus on stabilizing the unit, minimizing the potential for loss of power to the remaining 120 VAC Vital Instrumentation panelboards and restoring power to the affected 120 VAC Vital Instrumentation panelboard.

The 4 hour and 24 hour limits are longer than Completion Times allowed for some of the components that are without adequate vital AC power. Utilizing the LCO 3.0.6 exception to LCO 3.0.2 for components without adequate vital AC power, that would have the Required Action Completion Times shorter than 4 hours or 24 hours if declared inoperable, is acceptable because of:

a. The potential for decreased safety by requiring a change in unit conditions (i.e., requiring a shutdown) and not allowing stable operations to continue;

b. The potential for decreased safety by requiring entry into numerous applicable Conditions and Required Actions for components without adequate vital AC power and not providing sufficient time for the operators to perform the necessary evaluations and actions for restoring power to the affected train; and

c. The potential for an event in conjunction with a single failure of a redundant component.

BASES

ACTIONS

F.1 and F.2 (continued)

The digital ESPS channels are powered from KVIA and KVIB, and cannot actuate without power. The 4 hour Completion Time takes into account the importance to safety of restoring the 120 VAC Vital Instrumentation panelboards to OPERABLE status, the redundant capability afforded by the other OPERABLE 120 VAC Vital Instrumentation panelboards, and the low probability of an accident occurring during this period.

Panelboards KVIC and KVID supply some loads which trip upon loss of power. For example, RPS channels and ES analog channels go to a tripped state upon loss of power. The 24 hour Completion Time takes into account the importance to safety of restoring the 120 VAC Vital Instrumentation panelboards to OPERABLE status, the redundant capability afforded by the other OPERABLE 120 VAC Vital Instrumentation panelboards, and the low probability of an accident occurring during this period.

G.1 and G.2

If the Required Action and associated Completion Time are 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 12 hours and to MODE 5 within 84 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.

H.1

Condition H corresponds to a level of degradation in the electrical distribution system that causes a required safety function to be lost. When more than one Condition is entered, and this results in the loss of a required safety function, the plant is in a condition outside the accident analysis. Therefore, no additional time is justified for continued operation.

LCO 3.0.3 must be entered immediately to commence a controlled shutdown.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.8.8.1

This Surveillance verifies that the main feeder buses are functioning properly, with the correct circuit breaker alignment. The correct breaker alignment ensures the appropriate separation and independence is maintained, and the appropriate voltage is available to each required bus. The verification of proper voltage availability on the buses ensures that the required voltage is readily available for motive as well as control functions for critical system loads connected to these buses. The 7 day Frequency takes into account the redundant capability of the AC, DC, and AC vital electrical power distribution systems, and other indications available in the control room that alert the operator to system malfunctions.

SR 3.8.8.2

This Surveillance verifies that the required AC, DC, and AC vital electrical power distribution systems are functioning properly, with the correct circuit breaker alignment. The correct breaker alignment ensures the appropriate separation and independence is maintained, and the appropriate voltage is available to each ES power string and panelboard. The verification of voltage availability on the ES power strings, and panelboards ensures that voltage is readily available for motive as well as control functions for critical system loads connected to the ES power strings, and panelboards. Verification of voltage availability may be accomplished by observing alarm conditions, status lights or by confirming proper operation of a component supplied from each ES power string or panelboard. The 7 day Frequency takes into account the redundant capability of the AC, DC, and AC vital electrical power distribution systems, and other indications available in the control room that alert the operator to system malfunctions.

REFERENCES

1. UFSAR, Chapter 6.
2. UFSAR, Chapter 15.
3. Regulatory Guide 1.93, December 1974.
4. 10 CFR 50.36.

## B 3.8 ELECTRICAL POWER SYSTEMS

### B 3.8.9 Distribution Systems – Shutdown

BASES

BACKGROUND

A description of the AC, DC and AC vital electrical power distribution systems is provided in the Bases for LCO 3.8.8, "Distribution Systems – Operating."

APPLICABLE SAFETY ANALYSES

The initial conditions of accident and transient analyses in the UFSAR, Chapter 6 (Ref. 1) and Chapter 15 (Ref. 2), assume Engineered Safeguards (ES) systems are OPERABLE. The AC, DC, and AC vital electrical power distribution systems are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ES systems so that the fuel, Reactor Coolant System, and containment design limits are not exceeded.

The OPERABILITY of the AC, DC, and AC vital electrical power distribution systems is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY.

The OPERABILITY of the minimum AC, DC, and AC vital electrical power distribution systems during MODES 5 and 6, and during movement of irradiated fuel assemblies ensures that:

a. The unit can be maintained in the shutdown or refueling condition for extended periods;

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and

c. Adequate power is provided to mitigate events postulated during shutdown, such as a fuel handling accident.

The AC and DC electrical power distribution systems satisfy Criterion 3 of 10 CFR 50.36 (Ref. 3).

BASES (continued)

LCO

Various combinations of portions of systems, equipment, and components are required OPERABLE by other LCOs, depending on the specific plant condition. Implicit in those requirements is the required OPERABILITY of necessary support required features. This LCO explicitly requires energization of the portions of the electrical distribution system necessary to support OPERABILITY of required systems, equipment, and components – all specifically addressed in each LCO and implicitly required via the definition of OPERABILITY.

Maintaining these portions of the distribution system energized ensures the availability of sufficient power to operate the unit in a safe manner to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents).

APPLICABILITY

The AC and DC electrical power distribution buses, ES power strings and panelboards required to be OPERABLE in MODES 5 and 6, and during movement of irradiated fuel assemblies, provide assurance that:

a. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel in the core;

b. Systems needed to mitigate a fuel handling accident are available;

c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

The AC, DC, and AC vital electrical power distribution buses, ES power strings and panelboards requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.8.

BASES (continued)

ACTIONS

A.1, A.2.1, A.2.2, A.2.3, A.2.4, and A.2.5

Although redundant required equipment may require redundant buses, ES power strings and panelboards of electrical power distribution systems to be OPERABLE, a reduced set of OPERABLE distribution buses, ES power strings and panelboards may be capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS and fuel movement. By allowing the option to declare required equipment associated with an inoperable distribution buses, ES power strings and panelboards inoperable, appropriate restrictions are implemented in accordance with the affected distribution buses, ES power strings and panelboards LCO's Required Actions. In many instances, this option may involve undesired administrative efforts. Therefore, the allowance for sufficiently conservative actions is made (i.e., to suspend CORE ALTERATIONS, movement of irradiated fuel assemblies, and operations involving positive reactivity additions).

Suspension of these activities does not preclude completion of actions to establish a safe conservative condition. These actions minimize the probability of the occurrence of postulated events. It is further required to immediately initiate action to restore the required AC and DC electrical power distribution buses, ES power strings and panelboards and to continue this action until restoration is accomplished in order to provide the necessary power to the unit safety systems.

Notwithstanding performance of the above conservative Required Actions, a required decay heat removal (DHR) subsystem may be inoperable. In this case, Required Actions A.2.1 through A.2.4 do not adequately address the concerns relating to coolant circulation and heat removal. Pursuant to LCO 3.0.6, the DHR ACTIONS would not be entered. Therefore, Required Action A.2.5 is provided to direct declaring DHR inoperable, which results in taking the appropriate DHR actions.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required distribution buses, ES power strings and panelboards should be completed as quickly as possible in order to minimize the time the unit safety systems may be without power.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.8.9.1

This Surveillance verifies that the required main feeder buses are functioning properly, with all the required main feeder buses energized. The verification of proper voltage availability on the buses, ES power strings and panelboards ensures that the required power is readily available for motive as well as control functions for critical system loads connected to these buses. The 7 day Frequency takes into account the capability of the electrical power distribution buses, ES power strings and panelboards, and other indications available in the control room that alert the operator to system malfunctions.

SR 3.8.9.2

This Surveillance verifies that the required AC, DC, and AC vital electrical power distribution systems are functioning properly, with the correct circuit breaker alignment. The correct breaker alignment ensures the appropriate separation and independence is maintained, and the appropriate voltage is available to each ES power strings and panelboards. The verification of voltage availability on the ES power strings, and panelboards ensures that voltage is readily available for motive as well as control functions for critical system loads connected to the ES power strings, and panelboards. Verification of voltage availability may be accomplished by observing alarm conditions, status lights or by confirming proper operation of a component supplied from each ES power string or panelboard. The 7 day Frequency takes into account the redundant capability of the AC, DC, and AC vital electrical power distribution systems, and other indications available in the control room that alert the operator to system malfunctions.

REFERENCES

1. UFSAR, Chapter 6.
2. UFSAR, Chapter 14.
3. 10 CFR 50.36.

# B 3.9 REFUELING OPERATIONS

## B 3.9.1 Boron Concentration

BASES

BACKGROUND

The limit on the boron concentrations of the Reactor Coolant System (RCS) and the refueling canal during refueling ensures that the reactor remains subcritical during MODE 6. Refueling boron concentration is the soluble boron concentration in the coolant in each of these volumes having direct access to the reactor core during refueling.

The soluble boron concentration offsets the core reactivity and is measured by chemical analysis of a representative sample of the coolant in each of the volumes. The refueling boron concentration limit is specified in the COLR. Unit procedures ensure the specified boron concentration in order to maintain an overall core reactivity of $k_{eff} \leq 0.95$ during fuel handling, with control rods and fuel assemblies assumed to be in the most adverse configuration (least negative reactivity) allowed by unit procedures.

Oconee Design Criteria (Ref.1), require that two independent reactivity control systems of different design principles be provided. One of these systems must be capable of holding the reactor core subcritical under cold conditions. The Chemical Addition System serves as the system capable of maintaining the reactor subcritical in cold conditions by maintaining the boron concentration.

The reactor is brought to shutdown conditions before beginning operations to open the reactor vessel for refueling. After the RCS is cooled and depressurized and the vessel head is unbolted, the head is removed. The refueling canal is then flooded with borated water from the borated water storage tank.

The pumping action of the DHR System in the RCS, and the natural circulation due to thermal driving heads in the reactor vessel mix the concentrations in the RCS and refueling canal above the COLR limit. The DHR System is in operation during refueling (see LCO 3.9.4, "DHR and Coolant Circulation – High Water Level and LCO 3.9.5, DHR and Coolant Circulation – Low Water Level") to provide forced circulation in the RCS and assist in maintaining the boron concentrations in the RCS, and the refueling canal above the COLR limit.

BASES (continued)

APPLICABLE SAFETY ANALYSES

During refueling operations, the reactivity condition of the core is consistent with the initial conditions assumed for the boron dilution accident in the accident analysis and is conservative for MODE 6. The boron concentration limit specified in the COLR is based on the core reactivity at the beginning of each fuel cycle (the end of refueling) and includes an uncertainty allowance.

The required boron concentration and the unit refueling procedures that demonstrate the correct fuel loading plan (including full core mapping) ensure the $k_{eff}$ of the core will remain $\leq 0.95$ during the refueling operation.

The RCS boron concentration satisfies Criterion 2 of 10 CFR 50.36 (Ref. 2).

LCO

The LCO requires that a minimum boron concentration be maintained in the RCS and the refueling canal while in MODE 6. The boron concentration limit specified in the COLR ensures a core $k_{eff}$ of $\leq 0.95$ is maintained during fuel handling operations with CONTROL RODS and fuel assemblies assumed to be in the most adverse configuration (least negative reactivity) allowed by unit procedures.

Violation of the LCO results in uncertainty with respect to the degree of subcriticality and could lead to an inadvertent criticality during MODE 6.

APPLICABILITY

This LCO is applicable in MODE 6 to ensure that the fuel in the reactor vessel will remain subcritical.

Above MODE 6, LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," LCO 3.1.5, "Safety Rod Position Limits," and LCO 3.2.1, "Regulating Rod Position Limits," ensure that an adequate amount of negative reactivity is available to shut down the reactor and to maintain it subcritical.

ACTIONS

A.1 and A.2

Continuation of CORE ALTERATIONS or positive reactivity additions (including actions to reduce boron concentration) is contingent upon maintaining the unit in compliance with the LCO. If the boron concentration of the RCS or the refueling canal is less than its limit, all operations involving CORE ALTERATIONS or positive reactivity additions must be suspended immediately.

BASES

ACTIONS

A.1 and A.2 (continued)

Suspension of CORE ALTERATIONS and positive reactivity additions shall not preclude moving a component to a safe position.

A.3

In addition to immediately suspending CORE ALTERATIONS and positive reactivity additions, action to restore the concentration must be initiated immediately.

One means of complying with the action is to initiate boration of the affected volume. In determining the required combination of boration flow rate and concentration, there is no unique Design Basis Event that must be satisfied. The only requirement is to restore the boron concentration to its required value as soon as possible. In order to raise the boron concentration as soon as possible, the operator should begin boration with the best source available for unit conditions.

Once actions have been initiated, they must be continued until the boron concentration is restored. The restoration time depends on the amount of boron that must be injected to reach the required concentration.

SURVEILLANCE REQUIREMENTS

SR 3.9.1.1

This SR ensures the coolant boron concentration in the RCS and the refueling canal is within the COLR limits. The boron concentration of the coolant in each volume is determined every 72 hours by chemical analysis.

The Frequency is based on industry experience, which has shown 72 hours to be adequate.

REFERENCES

1. UFSAR, Section 3.1

2. 10 CFR 50.36.

## B 3.9 REFUELING OPERATIONS

### B 3.9.2 Nuclear Instrumentation

BASES

BACKGROUND

The source range neutron flux monitors are used during refueling operations to monitor the core reactivity condition. The installed source range neutron flux monitors are part of the Nuclear Instrumentation (NI) System. These detectors are located external to the reactor vessel and detect neutrons leaking from the core. The use of portable detectors is permitted, provided the LCO requirements are met.

The installed source range neutron flux monitors are fission chambers. The detectors monitor the neutron flux in counts per second. The instrument range covers six decades of neutron flux. The detectors also provide continuous visual indication in the control room to alert operators to a possible reactivity excursion. The NI System is designed in accordance with the criteria presented in Reference 1. If used, portable detectors should be functionally equivalent to the installed NI source range monitors.

APPLICABLE SAFETY ANALYSES

An OPERABLE source range neutron flux monitor is required to provide indication to alert the operator to unexpected changes in core reactivity, such as by a boron dilution accident or an improperly loaded fuel assembly. The safety analysis of the uncontrolled boron dilution accident is described in Reference 2. The analysis of the uncontrolled boron dilution accident shows that the core will remain subcritical, and there is sufficient time for the operator to take corrective action.

The source range neutron flux monitors satisfy Criterion 3 of 10 CFR 50.36 (Ref. 3).

LCO

This LCO requires one source range neutron flux monitor OPERABLE to ensure that monitoring capability is available to detect changes in core reactivity. One additional source range neutron flux monitor shall be OPERABLE during CORE ALTERATIONS and during positive reactivity additions. This additional requirement ensures redundant monitoring capability when positive reactivity changes are made to the core. Continuous indication must be available for each required neutron flux monitor.

BASES

LCO (continued)

The use of portable detectors is permitted for purpose of meeting this LCO. If used, portable detectors should be functionally equivalent to the installed source range monitors and satisfy applicable Surveillance Requirements.

APPLICABILITY

In MODE 6, the source range neutron flux monitors must be OPERABLE to determine changes in core reactivity. There is no other direct means available to check core reactivity levels. In MODES 2, 3, 4, and 5, these same installed source range detectors and circuitry are also required to be OPERABLE by LCO 3.3.9, "Source Range Neutron Flux."

In MODE 1, the neutron flux level is above the indicated range of the monitors. Thus, they are no longer relied upon for power level monitoring. Hence, there are no requirements for source range monitors in MODE 1.

ACTIONS

A.1 and A.2

With only one required source range neutron flux monitor OPERABLE during CORE ALTERATIONS or positive reactivity additions, redundancy has been lost. Since these instruments are the only direct means of monitoring core reactivity conditions, CORE ALTERATIONS and positive reactivity additions must be suspended immediately. Performance of Required Action A.1 shall not preclude completion of movement of a component to a safe position.

B.1

With no required source range neutron flux monitor OPERABLE, action to restore a monitor to OPERABLE status shall be initiated immediately. Once initiated, action shall be continued until a source range neutron flux monitor is restored to OPERABLE status or until the Applicability is exited.

B.2

With no required source range neutron flux monitor OPERABLE, there is no direct means of detecting changes in core reactivity. However, since CORE ALTERATIONS and positive reactivity additions are not to be

BASES

ACTIONS

B.2 (continued)

made in accordance with Required Actions A.1 and A.2, the core reactivity condition is stabilized until the source range neutron flux monitors are restored to OPERABLE status. This stabilized condition is verified by performing SR 3.9.1.1 to ensure that the required boron concentration exists.

The Completion Time of 4 hours is sufficient to obtain and analyze a reactor coolant sample for boron concentration. The Frequency of once per 12 hours ensures that unplanned changes in boron concentration would be identified. The 12 hour Frequency is reasonable, considering the low probability of a change in core reactivity during this time period.

SURVEILLANCE REQUIREMENTS

SR 3.9.2.1

SR 3.9.2.1 is the performance of a CHANNEL CHECK, which is a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that the indication channel(s) should be consistent with core conditions. Changes in fuel loading and core geometry can result in significant differences between source range channels, but each channel should be consistent with its local conditions. When in MODE 6 with one channel OPERABLE, a CHANNEL CHECK is still required. However, in this condition, a redundant source range instrument may not be available for comparison. The CHANNEL CHECK provides verification that the OPERABLE source range channel is energized and indicates a value consistent with current unit status.

The Frequency of 12 hours is consistent with the CHANNEL CHECK Frequency specified similarly for the same instruments in LCO 3.3.9.

SR 3.9.2.2

SR 3.9.2.2 is the performance of a CHANNEL CALIBRATION every 18 months. This SR is modified by a Note stating that neutron detectors are excluded from the CHANNEL CALIBRATION. The CHANNEL CALIBRATION for the source range nuclear instrument is a complete check and re-adjustment of the channel, from the pre-amplifier input to the indicator. The 18 month Frequency is based on the need to perform this Surveillance during the conditions that apply during a unit outage. Industry experience has shown these components usually pass the Surveillance when performed at the 18 month Frequency.

BASES (continued)

REFERENCES

1. UFSAR, Section 3.1.
2. UFSAR, Section 15.4.
3. 10 CFR 50.36.

# B 3.9 REFUELING OPERATIONS

## B 3.9.3 Containment Penetrations

BASES

BACKGROUND

During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, a release of fission product radioactivity within containment will be restricted from escaping to the environment when the LCO requirements are met. In MODES 1, 2, 3, and 4, this is accomplished by maintaining containment OPERABLE as described in LCO 3.6.1, "Containment." In MODE 6, the potential for containment pressurization as a result of an accident is not likely; therefore, requirements to isolate the containment from the outside atmosphere can be less stringent. In order to make this distinction, the penetration requirements are referred to as "containment closure" rather than "containment OPERABILITY." Containment closure means that specified escape paths are closed or capable of being closed. Since there is no significant potential for containment pressurization, the Appendix J leakage criteria and tests are not required.

The containment serves to contain fission product radioactivity that may be released from the reactor core following an accident, such that offsite radiation exposures are maintained within the requirements of 10 CFR 100. Additionally, the containment provides radiation shielding from the fission products that may be present in the containment atmosphere following accident conditions.

The containment equipment hatch, which is part of the containment pressure boundary, provides a means for moving large equipment and components into and out of containment. During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, the equipment hatch must be held in place by at least four bolts. Good engineering practice dictates that the bolts required by this LCO be approximately equally spaced.

The containment air locks, which are also part of the containment pressure boundary, provide a means for personnel access during MODES 1, 2, 3, and 4 unit operation in accordance with LCO 3.6.2, "Containment Air Locks." Each air lock has a door at both ends. The doors are normally interlocked to prevent simultaneous opening when containment OPERABILITY is required. During periods of unit shutdown

BASES

BACKGROUND (continued)

when containment OPERABILITY is not required, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment ingress and egress is necessary. During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, containment closure is required; therefore, the door interlock mechanism may remain disabled, but one air lock door must always remain closed. Placement of a temporary cover plate in the emergency air lock is an acceptable means for providing containment closure.

The temporary cover plate is installed and sealed against the inner emergency air lock door flange gasket. The temporary cover plate is visually inspected to ensure that no gaps exist. All cables, hoses and service air piping run through the sleeves on the temporary cover plate will also be installed and sealed. The sleeves will also be inspected to ensure that no gaps exist. Leak testing is not required prior to beginning fuel handling operations. Therefore, visual inspection of the temporary cover plate over the emergency air lock satisfies the requirement that the air lock be closed, which constitutes operability for this requirement.

The requirements on containment penetration closure ensure that a release of fission product radioactivity within containment will be restricted from escaping to the environment. The closure restrictions are sufficient to restrict fission product radioactivity release from containment due to a fuel handling accident during refueling.

The Reactor Building Purge System includes a supply penetration and exhaust penetration. During MODES 1, 2, 3, and 4, two valves in each of the supply and exhaust penetrations are secured in the closed position. The system is not subject to a Specification in MODE 5.

In MODE 6, large air exchanges are necessary to support refueling operations. The purge system is used for this purpose, and two valves in each penetration flow path may be closed on a unit vent high radiation signal.

Other containment penetrations that provide direct access from containment atmosphere to outside atmosphere must be isolated on at least one side. Isolation may be achieved by a closed automatic isolation valve, non-automatic power operated valve, manual isolation valve, blind flange, or equivalent. Equivalent isolation methods may include use of a material that can provide a temporary, atmospheric pressure ventilation barrier for the containment penetration(s) during fuel movements.

BASES (continued)

APPLICABLE SAFETY ANALYSES

During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, the most severe radiological consequences result from a fuel handling accident. The fuel handling accident is a postulated event that involves damage to irradiated fuel (Ref. 2). A minimum fuel transfer canal water level and the minimum decay time of 72 hours prior to CORE ALTERATIONS ensure that the release of fission product radioactivity subsequent to a fuel handling accident results in doses that are within the guideline values specified in 10 CFR 100. The design basis for fuel handling accidents has historically separated the radiological consequences from the containment capability. The NRC staff has treated the containment capability for fuel handling conditions as a logical part of the "primary success path" to mitigate fuel handling accidents, irrespective of the assumptions used to calculate the radiological consequences of such accidents (Ref. 2).

Containment penetrations satisfy Criterion 3 of 10 CFR 50.36.

LCO

This LCO reduces the consequences of a fuel handling accident in containment by limiting the potential escape paths for fission product radioactivity from containment. The LCO requires any penetration providing direct access from the containment atmosphere to the outside atmosphere to be closed except for the OPERABLE containment purge and exhaust penetrations. For the OPERABLE containment purge and exhaust penetrations, this LCO ensures that these penetrations are isolable by the RB purge isolation signal.

This LCO is modified by a note indicating that an emergency air lock door is not required to be closed when a temporary cover plate is installed.

APPLICABILITY

The containment penetration requirements are applicable during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment because this is when there is a potential for a fuel handling accident. In MODES 1, 2, 3, and 4, containment penetration requirements are addressed by LCO 3.6.1. In MODES 5 and 6, when CORE ALTERATIONS or movement of irradiated fuel assemblies within containment are not being conducted, the potential for a fuel handling accident does not exist. Therefore, under these conditions no requirements are placed on containment penetration status.

BASES (continued)

ACTIONS

A.1 and A.2

With the containment equipment hatch, air locks, or any containment penetration that provides direct access from the containment atmosphere to the outside atmosphere not in the required status, including the Containment Purge and Exhaust Isolation System not capable of automatic actuation when the purge and exhaust valves are open, the unit must be placed in a condition in which the isolation function is not needed. This is accomplished by immediately suspending CORE ALTERATIONS and movement of irradiated fuel assemblies within containment. Performance of these actions shall not preclude moving a component to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.9.3.1

This Surveillance demonstrates that each of the containment penetrations required to be in its closed position is in that position. Also the Surveillance will demonstrate that each open penetration's valve operator has motive power, which will ensure each valve is capable of being closed.

The Surveillance is performed every 7 days during the CORE ALTERATIONS or movement of irradiated fuel assemblies within the containment. The Surveillance interval is selected to be commensurate with the normal duration of time to complete fuel handling operations.

As such, this Surveillance ensures that a postulated fuel handling accident that releases fission product radioactivity within the containment will not result in a release of fission product radioactivity to the environment.

SR 3.9.3.2

This Surveillance demonstrates that each containment purge supply and exhaust isolation valve that is not locked, sealed or otherwise secured in the isolation position actuates to its isolation position on an actual or simulated high radiation signal. The frequency requires the isolation capability of the reactor building purge valves to be verified functional once each refueling outage prior to CORE ALTERATIONS or movement of irradiated fuel assemblies within containment. This ensures that this

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.3.2 (continued)

function is verified prior to CORE ALTERATIONS or movement of irradiated fuel assemblies within containment. This Surveillance will ensure that the valves are capable of closing after a postulated fuel handling accident to limit a release of fission product radioactivity from the containment.

REFERENCES

1. UFSAR, Section 15.11.

2. NRC letter to RG & E dated December 7, 1995, R.E. Ginna Nuclear Power Plant Conversion to Improved Standard Technical Specifications - Resolutions of Ginna Design Basis for Refueling Accidents.

## B 3.9 REFUELING OPERATIONS

### B 3.9.4 Decay Heat Removal (DHR) and Coolant Circulation – High Water Level

BASES

BACKGROUND

The purposes of the DHR Loops in MODE 6 are to remove decay heat and sensible heat from the Reactor Coolant System (RCS), to provide mixing of borated coolant, to provide sufficient coolant circulation to minimize the effects of a boron dilution accident, and to prevent boron stratification. Heat may be removed from the RCS by two methods. In the first method, heat is removed by circulating reactor coolant through the LPI heat exchanger(s), where the heat is transferred to the Low Pressure Service Water (LPSW) System via the LPI heat exchanger(s). The coolant is then returned to the reactor vessel via the core flood tank injection nozzles. Operation of a DHR Loop for normal cooldown or decay heat removal is manually accomplished from the control room. The heat removal rate is adjusted by control of the flow of reactor coolant through the LPI heat exchanger(s), bypassing the heat exchanger(s) and throttling of LPSW through the heat exchangers. Mixing of the reactor coolant is maintained by this continuous circulation of reactor coolant through the DHR Loop.

In the second method, the "B" spent fuel cooling pump may be lined up to take suction on the transfer canal, or the reactor vessel, and discharge to the spent fuel pool. Heat is removed by circulating reactor coolant through the spent fuel pool (SFP) heat exchanger, where the heat is transferred to the Recirculated Cooling Water (RCW) System. The coolant is then returned to the reactor vessel via the fuel transfer canal and the fuel transfer tubes. When using this method the remaining SF cooling trains operate normally.

APPLICABLE SAFETY ANALYSES

If the reactor coolant temperature is not maintained below 200°F, boiling of the reactor coolant could result. This could lead to inadequate cooling of the reactor fuel as a result of a loss of coolant in the reactor vessel. Additionally, boiling of the reactor coolant could lead to a reduction in boron concentration in the coolant due to boron plating out on components near the areas of the boiling the activity, and because of the possible addition of water to reactor vessel with a lower

BASES

APPLICABLE SAFETY ANALYSES (continued)

boron concentration than is required to keep the reactor subcritical. The loss of reactor coolant and the reduction in boron concentration in the reactor coolant would eventually challenge the integrity of the fuel cladding, which is a fission product barrier. One loop of DHR is required to be operational in MODE 6, with the water level ≥ 21.34 feet above the top of the reactor vessel flange, to prevent this challenge. The LCO does permit de-energizing the DHR pump for short durations under the condition that the boron concentration is not diluted. This conditional de-energizing of the DHR pump does not result in a challenge to the fission product barrier. The DHR loop satisfies Criteria 4 of 10 CFR 50.36 (Ref. 2).

LCO

Only one DHR loop is required for decay heat removal in MODE 6 with a water level ≥ 21.34 feet above the top of the reactor vessel flange. Only one DHR Loop is required to be operable because the volume of water above the reactor vessel flange provides backup decay heat removal capability. At least one DHR loop must be OPERABLE and in operation to provide:

a. Removal of decay heat;

b. Mixing of borated coolant to minimize the possibility of criticality; and

c. Indication of reactor coolant temperature.

To be considered OPERABLE, a DHR loop must include a pump, a heat exchanger, valves, piping, instruments, and controls to ensure an OPERABLE flow path and to determine the temperature. The flow path starts in one of the RCS hot legs and is returned to reactor vessel via either one or both of the Core Flood tank injection nozzles when using an LPI pump or the fuel transfer canal and a fuel transfer tube from the SFP when using a SFP Cooling System Pump. The BWST recirculation crossover line through valves LP-40 and LP-41 may be part of a flow path if it provides adequate decay heat removal capability.

Additionally, to be considered OPERABLE, a DHR loop must be capable of being manually aligned (remote or local) in the DHR mode for removal of decay heat. Operation of one loop can maintain the reactor coolant temperature as required. The LCO is modified by a Note that allows the required DHR loop to be removed from operation for up to 1 hour in an 8 hour period, provided no operation that would cause reduction of the RCS boron concentration is in progress. Boron concentration reduction

BASES

LCO (continued)

is prohibited because uniform concentration distribution cannot be ensured without forced circulation, etc. This allowance permits operations such as core mapping, alterations or maintenance in the vicinity of the reactor vessel hot leg nozzles and RCS to LPI isolation valve testing. During this 1 hour period, decay heat is removed by natural convection.

APPLICABILITY

One DHR loop must be OPERABLE and in operation in MODE 6 with the water level ≥ 21.34 ft above the top of the reactor vessel flange, to provide decay heat removal. The 21.34 ft level was selected because it corresponds to the 21.34 ft requirement established for fuel movement in the fuel handling accident analysis. Requirements for the DHR Loops in other MODES are covered by LCOs in Section 3.4, Reactor Coolant System (RCS), and Section 3.5, Emergency Core Cooling Systems (ECCS). DHR loop requirements in MODE 6, with the water level < 21.34 ft above the reactor vessel flange, are located in LCO 3.9.5, "Decay Heat Removal (DHR) and Coolant Circulation-Low Water Level.

ACTIONS

A.1

If DHR loop requirements are not met, there will be no forced circulation to provide mixing to establish uniform boron concentrations. Reduced boron concentrations can occur by adding water with a lower boron concentration than that contained in the RCS. Therefore, actions that reduce boron concentration shall be suspended immediately.

A.2

If DHR loop requirements are not met, actions shall be taken immediately to suspend the loading of irradiated fuel assemblies in the core. With no forced circulation cooling, decay heat removal from the core occurs by natural convection to the heat sink provided by the water above the core. A minimum refueling water level 21.34 feet above the reactor vessel flange provides an adequate available heat sink. Suspending any operation that would increase decay heat load, such as loading an irradiated fuel assembly, is prudent under this condition.

BASES

ACTIONS (continued)

A.3

If DHR loop requirements are not met, actions shall be initiated immediately in order to satisfy DHR loop requirements.

Restoration of one decay heat removal loop is required because this is the only active method of removing decay heat. Dissipation of decay heat through natural convection should not be relied upon for an extended period of time. Reliance on natural convection can lead to boiling which results in inventory loss. Sustained inventory loss can eventually result in inadequate decay heat removal from the core with subsequent release of fission products from the core to the reactor building atmosphere. The immediate Completion Time reflects the importance of restoring an adequate heat cooling loop.

A.4

If DHR loop requirements are not met, all containment penetrations providing direct access from the containment atmosphere to outside atmosphere shall be closed within 4 hours.

If no means of decay heat removal can be restored, the core decay heat could raise temperatures and cause boiling in the core which could result in uncovering the core and the release of radioactivity to the reactor building atmosphere. Closure of penetrations providing access to the outside atmosphere will prevent uncontrolled release of radioactivity to the environment.

SURVEILLANCE REQUIREMENTS

SR 3.9.4.1

This Surveillance demonstrates that the DHR loop is in operation and circulating reactor coolant. Verification includes flow rate, temperature, or pump status monitoring, which help assure that forced flow is providing heat removal. The Frequency of 12 hours is sufficient, considering the flow, temperature, pump control, and alarm indications available to the operator in the control room for monitoring the DHR System.

REFERENCES

1. 10 CFR 50.36.

## B 3.9 REFUELING OPERATIONS

### B 3.9.5 Decay Heat Removal (DHR) and Coolant Circulation – Low Water Level

BASES

BACKGROUND

The purposes of the DHR Loops in MODE 6 are to remove decay heat and sensible heat from the Reactor Coolant System (RCS), to provide mixing of borated coolant, to provide sufficient coolant circulation to minimize the effects of a boron dilution accident, and to prevent boron stratification. Heat is removed from the RCS by circulating reactor coolant through the LPI heat exchanger(s), where the heat is transferred to the Low Pressure Service Water (LPSW) System via the LPI heat exchanger. The coolant is then returned to the reactor vessel via the core flood tank injection nozzles. Operation of a DHR Loop for normal cooldown/decay heat removal is manually accomplished from the control room. The heat removal rate is adjusted by control of the flow of reactor coolant through the LPI heat exchanger(s), bypassing the heat exchanger(s) and by throttling of LPSW through the heat exchangers. Mixing of the reactor coolant is maintained by this continuous circulation of reactor coolant through the DHR Loop.

APPLICABLE SAFETY ANALYSES

If the reactor coolant temperature is not maintained below 200°F, boiling of the reactor coolant could result. This could lead to inadequate cooling of the reactor fuel due to resulting loss of coolant in the reactor vessel. Additionally, boiling of the reactor coolant could lead to a reduction in boron concentration in the coolant due to boron plating out on components near the areas of the boiling activity, and because of the possible addition of water to the reactor vessel with a lower boron concentration than is required to keep the reactor subcritical. The loss of reactor coolant and the reduction of boron concentration in the reactor coolant would eventually challenge the integrity of the fuel cladding, which is a fission product barrier. Without a large water inventory to provide a backup means of heat removal, two loops of DHR are required to be OPERABLE, and one is required to be in operation, to prevent this challenge.

The DHR Loops satisfy Criterion 4 of 10 CFR 50.36 (Ref. 1).

BASES (continued)

LCO

In MODE 6, with the water level < 21.34 ft above the top of the reactor vessel flange, two DHR loops must be OPERABLE. Additionally, one DHR loop must be in operation to provide:

a. Removal of decay heat;

b. Mixing of borated coolant to minimize the possibility of criticality; and

c. Indication of reactor coolant temperature.

To be considered OPERABLE, a DHR loop must consist of a pump, a heat exchanger, valves, piping, instruments, and controls to ensure an OPERABLE flow path and to determine the temperature. The flow path starts in one of the RCS hot legs and is returned to reactor vessel via one or both of the Core Flood tank injection nozzles. The BWST recirculation crossover line through valves LP-40 and LP-41 may be part of a flow path if it provides adequate decay heat removal capability.

Both pumps may be aligned to the Borated Water Storage Tank to support filling or draining of the refueling transfer canal or performance of required testing.

To be considered OPERABLE, each DHR loop must be capable of being manually aligned (remote or local) in the DHR mode for removal of decay heat. Operation of one DHR loop can maintain the reactor coolant temperature as required.

APPLICABILITY

Two DHR loops are required to be OPERABLE, and one in operation in MODE 6, with the water level < 21.34 ft above the top of the reactor vessel flange, to provide decay heat removal. Requirements for the LPI System in other MODES are covered by LCOs in Section 3.4, Reactor Coolant System (RCS), and Section 3.5, Emergency Core Cooling Systems (ECCS). DHR loop requirements in MODE 6, with the water level ≥ 21.34 ft above the top of the reactor vessel flange, are located in LCO 3.9.4, "Decay Heat Removal (DHR) and Coolant Circulation – High Water Level."

BASES (continued)

ACTIONS

A.1 and A.2

With fewer than the required loops OPERABLE, action shall be immediately initiated and continued until the DHR loop is restored to OPERABLE status or until ≥ 21.34 ft of water level is established above the reactor vessel flange. When the water level is established at ≥ 21.34 ft above the reactor vessel flange, the Applicability will change to that of LCO 3.9.4, and only one DHR loop is required to be OPERABLE and in operation. An immediate Completion Time is necessary due to the increased risk of operating without a large available inventory.

B.1

If no DHR loop is in operation or no DHR loop is OPERABLE, there may be no forced circulation to provide mixing to establish uniform boron concentrations. Reduced boron concentration can occur by adding water with a lower boron concentration than that contained in the RCS. Therefore, actions that reduce boron concentration shall be suspended immediately.

B.2

If no DHR loop is in operation or no DHR loop is OPERABLE, actions shall be initiated immediately and continued without interruption to restore one DHR loop to OPERABLE status and operation. Since the unit is in Conditions A and B concurrently, the restoration of two OPERABLE DHR loops and one operating DHR loop should be accomplished expeditiously.

B.3

If no DHR loop is in operation, all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere must be closed within 4 hours. With the DHR loop requirements not met, the potential exists for the coolant to boil and release radioactive gas to the containment atmosphere. Closing the containment penetrations that are open to the outside atmosphere prevents the uncontrolled release of radioactivity.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.9.5.1

This Surveillance demonstrates that one DHR loop is in operation. The flow rate is determined by the operator as that necessary to provide adequate decay heat removal capability.

The Frequency of 12 hours is sufficient, considering the flow, temperature, pump control, and alarm indications available to the operator to monitor the DHR Loops in the control room.

SR 3.9.5.2

Verification that each required pump is OPERABLE ensures that an additional DHR pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the required pump. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

REFERENCES

1. 10 CFR 50.36.

## B 3.9 REFUELING OPERATIONS

### B 3.9.6 Fuel Transfer Canal Water Level

BASES

BACKGROUND

The movement of irradiated fuel assemblies or performance of CORE ALTERATIONS, except during latching and unlatching of CONTROL ROD drive shafts, within containment requires a minimum water level of 21.34 ft above the top of the reactor vessel flange. During refueling, this maintains sufficient water level in the fuel transfer canal, and the spent fuel pool. Sufficient water is necessary to retain iodine fission product activity in the water in the event of a fuel handling accident (Refs. 1 and 2). Sufficient iodine activity would be retained to limit offsite doses from the accident within 10 CFR 100 limits, as provided by the guidance of Reference 3.

APPLICABLE SAFETY ANALYSES

During CORE ALTERATIONS and during movement of irradiated fuel assemblies, the water level in the fuel transfer canal is an initial condition design parameter in the analysis of the fuel handling accident in containment postulated by Regulatory Guide 1.25 (Ref. 1). A minimum water level of 23 ft (Regulatory Position C.1.c of Ref. 1) allows a decontamination factor (DF) of 100 (Regulatory Position C.1.g of Ref. 1) to be used in the accident analysis for iodine. This relates to the assumption that 99% of the total iodine released from the pellet to cladding gap of all the dropped fuel assembly rods is retained by the fuel transfer canal water. The fuel pellet to cladding gap is assumed to contain 10% of the total fuel rod iodine inventory (Ref. 1).

The fuel handling accident analysis inside containment is described in Reference 2. Since the minimum water level of 21.34 feet is less than 23 feet, the assumed iodine DF must be less than 100, according to Ref. 1, and calculated with comparable conservatism. An experimental test program described in WCAP-7828 (Ref. 4) evaluated the extent of removal of iodine released from a damaged irradiated fuel assembly. Using the analytical results from the test program described in WCAP-7828, with a water depth of 21.34 feet, a comparable DF of 89 was determined. With a minimum water level of 21.34 ft, and a minimum decay time of 72 hours prior to fuel handling, the analysis and test programs demonstrate that the iodine release due to a postulated fuel handling accident is adequately captured by the water, and offsite doses are maintained within allowable limits (Ref. 3).

Fuel Transfer Canal water level satisfies Criterion 2 of 10 CFR 50.36.

BASES (continued)

LCO

A minimum fuel transfer canal water level of 21.34 ft above the reactor vessel flange is required to ensure that the radiological consequences of a postulated fuel handling accident inside containment are within acceptable limits as provided by 10 CFR 100.

APPLICABILITY

LCO 3.9.6 is applicable during CORE ALTERATIONS, except during latching and unlatching of CONTROL ROD drive shafts, and when moving irradiated fuel assemblies within the containment. The LCO minimizes the possibility of a fuel handling accident in containment that is beyond the assumptions of the safety analysis. If irradiated fuel is not present in containment, there can be no significant radioactivity release as a result of a postulated fuel handling accident. Requirements for fuel handling accidents in the spent fuel pool are covered by LCO 3.7.11, "Fuel Storage Pool Water Level."

ACTIONS

A.1 and A.2

With a water level of < 21.34 ft above the top of the reactor vessel flange, all operations involving CORE ALTERATIONS or movement of irradiated fuel assemblies shall be suspended immediately to ensure that a fuel handling accident cannot occur.

The suspension of CORE ALTERATIONS and fuel movement shall not preclude completion of movement of a component to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.9.6.1

Verification of a minimum water level of 21.34 ft above the top of the reactor vessel flange ensures that the design basis for the postulated fuel handling accident analysis during refueling operations is met. Water at the required level above the top of the reactor vessel flange limits the consequences of damaged fuel rods that are postulated to result from a postulated fuel handling accident inside containment (Ref. 2).

The Frequency of 24 hours is based on engineering judgment and is considered adequate in view of the large volume of water and the normal procedural controls of valve positions, which make significant unplanned level changes unlikely.

BASES (continued)

REFERENCES

1. Regulatory Guide 1.25, March 23, 1972.
2. UFSAR Section 15.11.2.2.
3. 10 CFR 100.10.
4. WCAP-7828, December 1971

## B 3.9 REFUELING OPERATIONS

### B 3.9.7 Unborated Water Source Isolation Valves

BASES

**BACKGROUND**

During MODE 6 operations, all isolation valves for reactor makeup water sources containing unborated water that are connected to the Reactor Coolant System (RCS) must be closed to prevent unplanned boron dilution of the reactor coolant. The isolation valves must be secured in the closed position.

The Coolant Storage System is capable of supplying borated and unborated water to the RCS through various flow paths. Since a positive reactivity addition made by reducing the boron concentration is inappropriate during MODE 6, isolation of all unborated water sources prevents an unplanned boron dilution.

**APPLICABLE SAFETY ANALYSES**

The possibility of an inadvertent boron dilution event (Ref. 1) occurring during MODE 6 refueling operations is precluded by adherence to this LCO, which requires that potential dilution sources be isolated. Closing the required valves during refueling operations prevents the flow of unborated water to the filled portion of the RCS. The valves are used to isolate unborated water sources. These valves have the potential to indirectly allow dilution of the RCS boron concentration in MODE 6. By isolating unborated water sources when in MODE 6, a boron dilution event as analyzed in the UFSAR is prevented.

The RCS boron concentration satisfies Criterion 2 of 10 CFR 50.36 (Ref. 2).

**LCO**

This LCO requires that flow paths to the RCS from unborated water sources be isolated to prevent unplanned boron dilution during MODE 6 and thus avoid a reduction in SDM.

**APPLICABILITY**

In MODE 6, this LCO is applicable to prevent an inadvertent boron dilution event by ensuring isolation of all sources of unborated water to the RCS.

BASES

APPLICABILITY (continued)

For all other applicable MODES, the boron dilution accident was analyzed and was found to be capable of being mitigated. The boron dilution event is applicable in MODES 1 and 6.

ACTIONS

The ACTIONS table has been modified by a Note that allows separate Condition entry for each unborated water source isolation valve.

A.1

Continuation of CORE ALTERATIONS is contingent upon maintaining the unit in compliance with this LCO. With any valve used to isolate unborated water sources not secured in the closed position, all operations involving CORE ALTERATIONS must be suspended immediately. The Completion Time of "immediately" for performance of Required Action A.1 shall not preclude completion of movement of a component to a safe position.

Condition A has been modified by a Note to require that Required Action A.3 be completed whenever Condition A is entered.

A.2

Preventing inadvertent dilution of the reactor coolant boron concentration is dependent on maintaining the unborated water isolation valves secured closed. Securing the valves in the closed position ensures that the valves cannot be inadvertently opened. The Completion Time of "immediately" requires an operator to initiate actions to close an open valve and secure the isolation valve in the closed position immediately. Once actions are initiated, they must be continued until the valves are secured in the closed position.

A.3

Due to the potential of having diluted the boron concentration of the reactor coolant, SR 3.9.1.1 (verification of boron concentration) must be performed whenever Condition A is entered to demonstrate that the required boron concentration exists. The Completion Time of 4 hours is sufficient to obtain and analyze a reactor coolant sample for boron concentration.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.9.7.1

These valves are to be secured closed to isolate possible dilution paths. The likelihood of a significant reduction in the boron concentration during MODE 6 operations is remote due to the large mass of borated water in the fuel transfer canal and the fact that all unborated water sources are isolated, precluding a dilution. The boron concentration is checked every 72 hours during MODE 6 under SR 3.9.1.1. This Surveillance demonstrates that the valves are closed through a system walkdown. The 31 day Frequency is based on engineering judgment and is considered reasonable in view of other administrative controls that will ensure that the valve opening is an unlikely possibility.

REFERENCES

1. UFSAR, Section 15.4.1.

2. 10 CFR 50.36.

# B 3.10 STANDBY SHUTDOWN FACILITY

## B 3.10.1 Standby Shutdown Facility (SSF)

BASES

BACKGROUND

The Standby Shutdown Facility (SSF) is designed as a standby system for use under certain emergency conditions. The system provides additional "defense in-depth" protection for the health and safety of the public by serving as a backup to existing safety systems. The SSF is provided as an alternate means to achieve and maintain the unit in MODE 3 with average RCS temperature $\geq$ 525°F (unless the initiating event causes the unit to be driven to a lower temperature) following 10 CFR 50 Appendix R fire, sabotage, turbine building flood, station blackout (SBO) and tornado missile events, and is designed in accordance with criteria associated with these events. In that the SSF is a backup to existing safety systems, the single failure criterion is not required. Failures in the SSF systems will not cause failures or inadvertent operations in other plant systems. The SSF requires manual activation and can be activated if emergency systems are not available.

The SSF is designed to maintain the reactor in a safe shutdown condition for a period of 72 hours following 10 CFR 50 Appendix R fire, turbine building flood, sabotage, SBO, or tornado missile events. This is accomplished by re-establishing and maintaining Reactor Coolant Pump Seal cooling; assuring natural circulation and core cooling by maintaining the primary coolant system filled to a sufficient level in the pressurizer while maintaining sufficient secondary side cooling water; and maintaining the reactor subcritical by isolating all sources of Reactor Coolant System (RCS) addition except for the Reactor Coolant Makeup System which supplies makeup of a sufficient boron concentration.

The main components of the SSF are the SSF Auxiliary Service Water (ASW) System, SSF Portable Pumping System, SSF Reactor Coolant (RC) Makeup System, SSF Power System, and SSF Instrumentation.

The SSF ASW System is a high head, high volume system designed to provide sufficient steam generator (SG) inventory for adequate decay heat removal for three units during a loss of normal AC power in conjunction with the loss of the normal and emergency feedwater systems. One motor driven SSF ASW pump, located in the SSF, serves all three units. The ASW pump suction supply is lake water from the embedded Unit 2 condenser circulating water (CCW) piping.

BASES

BACKGROUND (continued)

The SSF ASW System is used to provide adequate cooling to maintain single phase RCS natural circulation flow in MODE 3 with an average RCS temperature ≥ 525°F (unless the initiating event causes the unit to be driven to a lower temperature). In order to maintain single phase RCS natural circulation flow, at least 5 of the 9 Bank 2, Group B pressurizer heaters must be OPERABLE. These heaters are needed to compensate for ambient heat loss from the pressurizer. As long as the temperature in the pressurizer is maintained, RCS pressure will also be maintained. This will preclude hot leg voiding and ensure adequate natural circulation cooling.

The SSF Portable Pumping System, which includes a submersible pump and a flow path capable of taking suction from the intake canal and discharging into the Unit 2 CCW line, is designed to provide a backup supply of water to the SSF in the event of loss of CCW and subsequent loss of CCW siphon flow. The SSF Portable Pumping System is installed manually according to procedures.

The SSF RC Makeup System is designed to supply makeup to the RCS in the event that normal makeup systems are unavailable. An SSF RC Makeup Pump located in the Reactor Building of each unit supplies makeup to the RCS should the normal makeup system flow and seal cooling become unavailable. The system is designed to ensure that sufficient borated water is provided from the spent fuel pools to allow the SSF to maintain all three units in MODE 3 with average RCS temperature ≥ 525°F (unless the initiating event causes the unit to be driven to a lower temperature) for approximately 72 hours. An SSF RC Makeup Pump is capable of delivering borated water from the Spent Fuel Pool to the RC pump seal injection lines. A portion of this seal injection flow is used to makeup for reactor coolant pump seal leakage while the remainder flows into the RCS to makeup for other RCS leakage (non LOCA).

The SSF Power System includes 4160 VAC, 600 VAC, 208 VAC, 120 VAC and 125 VDC power. It consists of switchgear, a load center, motor control centers, panelboards, remote starters, batteries, battery chargers, inverters, a diesel generator (DG), relays, control devices, and interconnecting cable supplying the appropriate loads.

The AC power system consists of 4160 V switchgear OTS1; 600 V load center OXSF; 600 V motor control centers XSF, 1XSF, 2XSF, 3XSF; 208 V motor control centers 1XSF, 1XSF-1, 2XSF, 2XSF-1, 3XSF, 3XSF-1; 120 V panelboards KSF, KSFC.

BASES

BACKGROUND (continued)

The DC power system consists of two 125 VDC batteries and associated chargers, two 125 VDC distribution centers (DCSF, DCSF-1), and a DC power panelboard (DCSF). Only one battery and associated charger is required to be operable and connected to the 125 VDC distribution center to supply the 125 VDC loads. In this alignment, which is normal, the battery is floated on the distribution center and is available to assure power without interruption upon loss of its associated battery charger or AC power source. The other 125 VDC battery and its associated charger are in a standby mode and are not normally connected to the 125 VDC distribution center. However, they are available via manual connection to the 125 VDC distribution center to supply SSF loads, if required.

The SSF Power System is provided with standby power from a dedicated DG. The SSF DG and support systems consists of the diesel generator, fuel oil transfer system, air start system, diesel engine service water system, as well as associated controls and instrumentation. This SSF DG is rated for continuous operation at 3500 kW, 0.8 pf, and 4160 VAC. The SSF electrical design load does not exceed the continuous rating of the DG. The auxiliaries required to assure proper operation of the SSF DG are supplied entirely from the SSF Power System. The SSF DG is provided with manual start capability from the SSF only. It uses a compressed air starting system with four air storage tanks. An independent fuel system, complete with a separate underground storage tank, duplex filter arrangement, a fuel oil transfer pump, and a day tank, is supplied for the DG.

The SSF Power System OPERABILITY is supported by portions of the SSF HVAC System, consisting of the SSF Air Conditioning (AC) and Ventilation Systems. The SSF AC System, which includes the HVAC service water system and AC equipment (fan motors, compressors, condensors, and coils) must be OPERABLE to support SSF Power System OPERABILITY. The SSF AC System is designed to maintain the SSF Control Room, Computer Room, and Battery Rooms within their design temperature range. Elevated temperatures in the SSF Control Room and Computer Room could cause the SSF Power System to fail during an accident which requires operation of the SSF. Since the SSF HVAC service water pumps perform a redundant function, only one of two are required to be OPERABLE for the SSF HVAC service water system to be considered OPERABLE. The SSF Ventilation System, which supplies outside air to the Switchgear, Pump, HVAC and Diesel Generator Rooms, is composed of the following four subsystems: Constant Ventilation, Summer Ventilation, On-line Ventilation, and Diesel Generator Engine Ventilation. These ventilation systems work together to provide cooling to the various rooms of the SSF under both standby

BASES

BACKGROUND (continued)

and on-line modes. The Diesel Generator Engine Ventilation fan is required for OPERABILITY of the SSF Power System. The six fans associated with the other three ventilation systems may or may not be required for SSF OPERABILITY dependent upon outside air temperature. If one of these ventilation fans fail, an engineering evaluation must be performed to determine if any of the SSF Systems or instrumentation are inoperable.

SSF Instrumentation is provided to monitor RCS pressure, RCS Loop A and B temperature (hot leg and cold leg), pressurizer water level, and SG A and B water level. Indication is displayed on the SSF control panel.

APPLICABLE SAFETY ANALYSES

The SSF serves as a backup for existing safety systems to provide an alternate and independent means to achieve and maintain one, two, or three Oconee units in MODE 3 with average RCS temperature $\geq$ 525°F (unless the initiating event causes the unit to be driven to a lower temperature) for up to 72 hours following 10 CFR 50 Appendix R fire, a turbine building flood, sabotage, SBO, or tornado missile events (Refs. 1, 6, 7, and 8).

The OPERABILITY of the SSF is consistent with the assumptions of the Oconee Probabilistic Risk Assessment (Ref. 2). Therefore, the SSF satisfies Criterion 4 of 10 CFR 50.36 (Ref. 3).

LCO

The SSF Instrumentation in Table B 3.10.1-1 and the following SSF Systems shall be OPERABLE:

a. SSF Auxiliary Service Water System;

b. SSF Portable Pumping System;

c. SSF Reactor Coolant Makeup System; and

d. SSF Power System.

An OPERABLE SSF ASW System includes five pressurizer heaters capable of being powered from the SSF, and an SSF ASW pump, piping, instruments, and controls to ensure a flow path capable of taking suction from the Unit 2 condenser circulating water (CCW) line and discharging into the secondary side of each SG. An OPERABLE SSF Portable Pumping System includes an SSF submersible pump and a flow

BASES

LCO (continued)

path capable of taking suction from the intake canal and discharging into the Unit 2 CCW line. An OPERABLE Reactor Coolant Makeup System includes an SSF RC makeup pump, piping, instruments, and controls to ensure a flow path capable of taking suction from the spent fuel pool and discharging into the RCS. An OPERABLE SSF Power System includes the SSF DG, diesel support systems, 4160 VAC, 600 VAC, 208 VAC, 120 VAC, and 125 VDC systems. Only one 125 VDC SSF battery and its associated charger are required to be OPERABLE to support OPERABILITY of the 125 VDC system.

APPLICABILITY

The SSF System is required in MODES 1, 2, and 3 to provide an alternate means to achieve and maintain the unit in MODE 3 with average RCS temperature $\geq$ 525°F (unless the initiating event causes the unit to be driven to a lower temperature) following 10 CFR 50 Appendix R fire, turbine building flood, sabotage, SBO and tornado missile events. The safety function of the SSF is to achieve and maintain the unit in MODE 3 with average RCS temperature $\geq$ 525°F (unless the initiating event causes the unit to be driven to a lower temperature); therefore, this LCO is not applicable in MODES 4, 5, or 6.

ACTIONS

The exception for LCO 3.0.4, provided in the Note of the Actions, permits entry into MODES 1, 2, and 3 with the SSF not OPERABLE. This is acceptable because the SSF is not required to support normal operation of the facility or to mitigate a design basis accident.

A.1, B.1, C.1, D.1, and E.1

With one or more of the SSF Systems inoperable or the required SSF instrumentation of Table B 3.10.1-1 inoperable, the SSF is in a degraded condition and the system(s) or instrumentation must be restored to OPERABLE status within 7 days. The 7 day Completion Time is based on the low probability of an event occurring which would require the SSF to be utilized.

F.1

If the Required Action and associated Completion Time of Condition A, B, C, D, or E are not met when SSF Systems or Instrumentation are inoperable due to maintenance, the unit may continue to operate provided that the SSF is restored to OPERABLE status within 45 days from discovery of initial inoperability.

BASES

ACTIONS

F.1 (continued)

This Completion Time is modified by a Note that indicates that the SSF shall not be in Condition F for more than a total of 45 days in a calendar year. This includes the 7 day Completion Time that leads to entry into Condition F. For example, if the SSF ASW System is inoperable for 10 days, the 45 day special inoperability period is reduced to 35 days. If the SSF ASW System is inoperable for 6 days, Condition A applies and there is no reduction in the 45 day allowance. The limit of 45 days per calendar year minimizes the number and duration of extended outages associated with exceeding the 7 day Completion Time of a Condition.

G.1 and G.2

If the Required Action and associated Completion Time of Condition F are not met or if the Required Action and associated Completion Time of Condition A, B, C, D, or E are not met for reasons other than Condition F, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 12 hours and MODE 4 within 84 hours. The allowed Completion Times are appropriate, to reach the required unit conditions from full power conditions in an orderly manner and without challenging plant systems, considering a three unit shutdown may be required.

SURVEILLANCE REQUIREMENTS

SR 3.10.1.1

Performance of the CHANNEL CHECK once every 7 days for each required instrumentation channel ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel with a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; therefore, it is key in verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION. This SR is modified by a Note to indicate that it is not applicable to the SSF RCS temperature instrument channels, which are common to the RPS RCS temperature instrument channels and are normally aligned through a transfer isolation device to each Unit control room. The instrument string to the SSF control room is checked and calibrated every 18 months.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.10.1.1 (continued)

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

The Frequency is based on unit operating experience that demonstrates channel failure is rare.

SR 3.10.1.2

Verifying battery terminal voltage while on float charge for the batteries helps to ensure the effectiveness of the charging system and the ability of the batteries to perform their intended function. Float charge is the condition in which the charger is supplying the continuous charge required to overcome the internal losses of a battery (or battery cell) and maintain the battery (or a battery cell) in a fully charged state. The voltage requirements are based on the nominal design voltage of the battery and are consistent with the initial voltages assumed in the battery sizing calculations. The 7 day Frequency is consistent with manufacturer recommendations and IEEE-450 (Ref. 4).

SR 3.10.1.3 and 3.10.1.4

SR 3.10.1.3 provides verification that the level of fuel oil in the day tank is at or above the level at which fuel oil is automatically added. The level is expressed as an equivalent volume in gallons. The day tank is sized based on the amount of fuel oil required to successfully start the DG and to allow for orderly shutdown of the DG upon loss of fuel oil from the main storage tank.

SR 3.10.1.4 provides verification that there is an adequate inventory of fuel oil in the storage tanks to support SSF DG operation for 72 hours at full load. The 72 hour period is sufficient time to place the unit in a safe shutdown condition.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.10.1.3 and 3.10.1.4 (continued)

The 31 day Frequency for these SRs is adequate to assure that a sufficient supply of fuel oil is available, since low level alarms are provided and unit operators would be aware of any large uses of fuel oil during this period.

SR 3.10.1.5

The SR requires the DG to start (normal or emergency) from standby conditions and achieve required voltage and frequency. Standby conditions for a DG means that the diesel engine coolant and oil are being continuously circulated and temperature is being maintained consistent with manufacturer recommendations. This SR is modified by a Note to indicate that all DG starts for this Surveillance may be preceded by an engine prelube period and followed by a warmup period prior to loading. This minimizes wear on moving parts that do not get lubricated when the engine is running.

The 31 day Frequency is consistent with Regulatory Guide 1.9 (Ref. 5). This Frequency provides adequate assurance of DG OPERABILITY, while minimizing degradation resulting from testing.

SR 3.10.1.6

This Surveillance ensures that sufficient air start capacity for the SSF DG is available, without the aid of the refill compressor. The SSF DG air start system is equipped with four air storage tanks. Each set of two tanks will provide sufficient air to start the SSF DG a minimum of three successive times without recharging. The pressure specified in this SR is intended to reflect the lowest value at which the three starts can be accomplished.

The 31 day Frequency takes into account the capacity, capability, redundancy, and diversity of the AC sources.

SR 3.10.1.7

This Surveillance demonstrates that the fuel oil transfer pump automatically starts and transfers fuel oil from the underground fuel oil storage tank to the day tank. This is required to support continuous operation of SSF DG. This Surveillance provides assurance that the fuel

BASES

SURVEILLANCE REQUIREMENTS

SR 3.10.1.7 (continued)

oil transfer pump is OPERABLE, the fuel oil piping system is intact, the fuel delivery piping is not obstructed, and the controls and control systems for automatic fuel transfer systems are OPERABLE.

The 92 day Frequency is considered acceptable based on operating experience.

SR 3.10.1.8

A sample of fuel oil is required to be obtained from the SSF day tank and underground fuel oil storage tank in accordance with the Diesel Fuel Oil Testing Program in order to ensure that fuel oil viscosity, water, and sediment are within the limits of the Diesel Fuel Oil Testing Program.

The 92 day Frequency is considered acceptable based on operating experience related to diesel fuel oil quality.

SR 3.10.1.9

This Surveillance verifies that the SSF DG is capable of synchronizing with the offsite electrical system and accepting loads greater than or equal to the equivalent of the maximum expected accident loads. A minimum run time of 60 minutes is required to stabilize electrical loads, while minimizing the time that the DG is connected to the offsite source.

Although no power factor requirements are established by this SR, the DG is normally operated at a power factor between 0.8 lagging and 1.0. The 0.8 value is the design rating of the machine, while the 1.0 is an operational limitation to ensure circulating currents are minimized. The load band is provided to avoid routine overloading of the DG. Routine overloading may result in more frequent teardown inspections in accordance with vendor recommendations in order to maintain DG OPERABILITY.

The normal 92 day Frequency for this Surveillance is consistent with Regulatory Guide 1.9 (Ref. 3).

BASES

SURVEILLANCE REQUIREMENTS

SR 3.10.1.9 (continued)

This SR is modified by three Notes. Note 1 indicates that diesel engine runs for this Surveillance may include gradual loading, as recommended by the manufacturer, so that mechanical stress and wear on the diesel engine are minimized. Note 2 states that momentary transients because of changing bus loads do not invalidate this test. Similarly, momentary power factor transients above the limit will not invalidate the test. Note 3 indicates that all DG starts for this Surveillance may be preceded by an engine prelube period and followed by a warmup period prior to loading. This minimizes wear on moving parts that do not get lubricated.

SR 3.10.1.10

Visual inspection of the battery cells, cell plates, and battery racks provides an indication of physical damage or abnormal deterioration that could potentially degrade battery performance.

The presence of physical damage or deterioration does not necessarily represent a failure of this SR, provided an evaluation determines that the physical damage or deterioration does not affect the OPERABILITY of the battery (its ability to perform its design function).

The 12 month Frequency for this SR is consistent with IEEE-450 (Ref. 4), which recommends detailed visual inspection of cell condition and rack integrity on a yearly basis.

SR 3.10.1.11

Visual inspection of battery cell to cell and terminal connections provides an indication of physical damage that could potentially degrade battery performance. The anti-corrosion material is used to help ensure good electrical connections and to reduce terminal deterioration. The visual inspection for corrosion is not intended to require removal of and inspection under each terminal connection.

The limits established for this SR must be no more than 20% above the resistance as measured during installation or not above the ceiling value established by the manufacturer.

The Surveillance Frequency for these inspections is 12 months. This Frequency is considered acceptable based on operating experience related to detecting corrosion trends.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.10.1.12

A battery service test is a special test of the battery capability, as found, to satisfy the design requirements (battery duty cycle) of the DC electrical power system. The discharge rate and test length correspond to the design duty cycle requirements. The design basis discharge time for the SSF battery is one hour.

The Surveillance Frequency for this test is 12 months. This Frequency is considered acceptable based on operating experience.

SR 3.10.1.13

CHANNEL CALIBRATION is a complete check of the instrument channel, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift to ensure that the instrument channel remains operational between successive tests. CHANNEL CALIBRATION shall find that measurement errors and bistable setpoint errors are within the assumptions of the setpoint analysis. CHANNEL CALIBRATIONS must be performed consistent with the assumptions of the setpoint analysis. This Frequency is justified by the assumption of an 18 month calibration interval to determine the magnitude of equipment drift in the setpoint analysis.

SR 3.10.1.14

Inservice Testing of the SSF valves demonstrates that the valves are mechanically OPERABLE and will operate when required. These valves are required to operate to ensure the required flow path.

The specified Frequency is in accordance with the IST Program requirements. Operating experience has shown that these components usually pass the SR when performed at the IST Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.10.1.15

This SR requires the SSF pumps to be tested in accordance with the IST Program. The IST verifies the required flow rate at a discharge pressure to verify OPERABILITY. The SR is modified by a note indicating that it is not applicable to the SSF submersible pump.

The specified Frequency is in accordance with the IST Program requirements. Operating experience has shown that these components usually pass the SR when performed at the IST Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.10.1.16

This SR requires the SSF submersible pump to be tested on a 2 year Frequency and verifies the required flow rate at a discharge pressure to verify OPERABILITY.

The specified Frequency is based on the pump being not QA grade and on operating experience that has shown it usually passes the SR when performed at the 2 year Frequency.

REFERENCES

1. UFSAR, Section 9.6.

2. Oconee Probabilistic Risk Assessment.

3. 10 CFR 50.36.

4. IEEE-450-1987.

5. Regulatory Guide 1.9, Rev. 0, December 1974.

6. NRC Letter from L. A. Wiens to H. B. Tucker, "Safety Evaluation Report on Effect of Tornado Missiles on Oconee Emergency Feedwater System," dated July 28, 1989.

7. NRC Letter from L. A. Wiens to J. W. Hampton, "Safety Evaluation for Station Blackout (10 CFR 50.63) - Oconee Nuclear Station, Units 1, 2, and 3," dated March 10, 1992.

BASES

REFERENCES (continued)

8. NRC Letter from L. A. Wiens to J. W. Hampton, "Supplemental Safety Evaluation for Station Blackout (10 CFR 50.63) - Oconee Nuclear Station, Units 1, 2, and 3," dated December 10, 1992.

Table B 3.10.1-1 (page 1 of 1)
SSF Instrumentation

| FUNCTION | REQUIRED CHANNELS PER UNIT |
|---|---|
| 1. Reactor Coolant System Pressure | 1 |
| 2. Reactor Coolant System Temperature (Tc) | 1/Loop |
| 3. Reactor Coolant System Temperature (Th) | 1/Loop |
| 4. Pressurizer Water Level | 1 |
| 5. Steam Generator A & B Water Level | 1/SG |

## B 3.10 STANDBY SHUTDOWN FACILITY

### B 3.10.2 Standby Shutdown Facility (SSF) Battery Cell Parameters

BASES

| | |
|---|---|
| BACKGROUND | This LCO delineates the limits on electrolyte temperature level, float voltage, and specific gravity for the SSF Power System batteries. A discussion of these batteries and their OPERABILITY requirements is provided in the Bases for LCO 3.10.1, "Standby Shutdown Facility (SSF)." |
| APPLICABLE SAFETY ANALYSES | The SSF serves as a backup for existing safety systems to provide an alternate and independent means to achieve and maintain one, two, or three Oconee units in MODE 3 with average RCS temperature ≥ 525°F (unless the initiating event causes the unit to be driven to a lower temperature) for up to 72 hours following a 10 CFR 50 Appendix R fire event, a turbine building flood, sabotage, SBO, or tornado missile events (Refs. 1, 5, 6, and 7).<br><br>The OPERABILITY of the SSF DC system is consistent with the assumptions of the Oconee Probabilistic Risk Assessment (Ref. 2). Therefore, the SSF battery cell parameters satisfy Criterion 4 of 10 CFR 50.36 (Ref. 3). |
| LCO | The SSF Battery cell parameters must remain within acceptable limits to ensure availability of the required SSF Power System DC power to shut down the reactor and maintain it in a safe condition after a 10 CFR 50 Appendix R fire, turbine building flood, sabotage, SBO, or tornado missile events. Electrolyte limits are conservatively established, allowing continued DC electrical system function even with Category A and B limits not met. |
| APPLICABILITY | The SSF battery cell parameters are required solely for the support of the associated SSF power system battery. Therefore, battery cell parameters are only required to be met when the SSF DC power source is required to be OPERABLE. Refer to the Applicability discussion in Bases for LCO 3.10.1. |

BASES (continued)

ACTIONS

The ACTIONS Table is modified by a Note which indicates that LCO 3.0.4 is not applicable. This is acceptable since a battery remains OPERABLE when one or more cells does not meet Category A or B limits but continues to meet Category C limits. Failure to meet Category C limits requires declaring the SSF battery inoperable.

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

With one or more cells in a required SSF battery not within limits (i.e., Category A limits not met or Category B limits not met or Category A and B limits not met) but within the Category C limits specified in Table 3.10.2-1 in the accompanying LCO, the battery is degraded but there is still sufficient capacity to perform the intended function. Therefore, the affected battery is not required to be considered inoperable solely as a result of Category A or B limits not met, and continued operation is permitted for a limited period.

The pilot cell electrolyte level and float voltage are required to be verified to meet the Category C limits within 1 hour (Required Action A.1). This check will provide a quick indication of the status of the remainder of the battery cells. One hour provides time to inspect the electrolyte level and to confirm the float voltage of the pilot cells. One hour is considered a reasonable amount of time to perform the required verification.

Verification that the Category C limits are met (Required Action A.2) provides assurance that during the time needed to restore the parameters to the Category A and B limits, the battery will still be capable of performing its intended function. A period of 24 hours is allowed to complete the initial verification because specific gravity measurements must be obtained for each connected cell. Taking into consideration both the time required to perform the required verification and the assurance that the battery cell parameters are not severely degraded, this time is considered reasonable. The verification is repeated at 7 day intervals until the parameters are restored to Category A and B limits. This periodic verification is consistent with the normal Frequency of pilot cell Surveillances.

Continued operation is only permitted for 90 days before battery cell parameters must be restored to within Category A and B limits. With the consideration that, while battery capacity is degraded, sufficient capacity exists to perform the intended function and to allow time to fully restore the battery cell parameters to normal limits, this time is acceptable prior to declaring the battery inoperable.

BASES

ACTIONS (continued)

B.1

With the Required Action and associated Completion Time not met, or with the required SSF battery with one or more battery cell parameters outside the Category C limit for any connected cell, or with the average electrolyte temperature of representative cells falling below 60°F, sufficient capacity to supply the maximum expected load requirement is not assured and the SSF Power System must be declared inoperable immediately.

SURVEILLANCE REQUIREMENTS

SR 3.10.2.1

This SR verifies that Category A battery cell parameters are consistent with IEEE-450 (Ref. 4), which recommends regular battery inspections including voltage, specific gravity, and electrolyte temperature of pilot cells.

SR 3.10.2.2

The quarterly inspection of specific gravity and voltage is consistent with IEEE-450 (Ref. 4).

SR 3.10.2.3

This Surveillance verification that the average temperature of representative cells is ≥ 60°F is consistent with a recommendation of IEEE-450 (Ref. 4), which states that the temperature of electrolytes in representative cells should be determined on a quarterly basis.

Lower than normal temperatures act to inhibit or reduce battery capacity. This SR ensures that the operating temperatures remain within an acceptable operating range. This limit is based on plant specific calculations.

Table 3.10.2-1

This table delineates the limits on electrolyte level, float voltage, and specific gravity for three different categories. The meaning of each category is discussed below.

Category A defines the normal parameter limit for each designated pilot cell in each battery. The cells selected as pilot cells are those whose

BASES

SURVEILLANCE REQUIREMENTS

Table 3.10.2-1 (continued)

temperature, voltage and electrolyte specific gravity are considered to approximate the state of charge of the entire battery.

The Category A limits specified for electrolyte level are based on manufacturer recommendations and are consistent with the guidance in IEEE-450 (Ref. 4), with the extra 1/4 inch allowance above the high water level indication for operating margin to account for temperatures and charge effects. In addition to this allowance, footnote (a) to Table 3.10.2-1 permits the electrolyte level to be above the specified maximum level during equalizing charge, provided it is not overflowing. These limits ensure that the plates suffer no physical damage and that adequate electron transfer capability is maintained in the event of transient conditions. IEEE-450 (Ref. 4) recommends that electrolyte level readings should be made only after the battery has been at float charge for at least 72 hours.

The Category A limit specified for float voltage is ≥ 2.13 V per cell. This value is based on a recommendation of IEEE-450 (Ref. 4), which states that prolonged operation of cells < 2.13 V can reduce the life expectancy of cells.

The Category A limit specified for specific gravity for each pilot cell is ≥ 1.200 (0.015 below the manufacturer fully charged nominal specific gravity or a battery charging current that had stabilized at a low value). This value is characteristic of a charged cell with adequate capacity. According to IEEE-450 (Ref. 4), the specific gravity readings are based on a temperature of 77°F (25°C).

The specific gravity readings are corrected for actual electrolyte temperature and level. For each 3°F (1.67°C) above 77°F (25°C), 1 point (0.001) is added to the reading; 1 point is subtracted for each 3°F below 77°F. The specific gravity of the electrolyte in a cell increases with a loss of water due to electrolysis or evaporation.

Category B defines the normal parameter limits for each connected cell. The term "connected cell" excludes any battery cell that may be jumpered out.

The Category B limits specified for electrolyte level, float voltage, and specific gravity are the same as those specified for Category A and have been discussed above. In addition, it is required that the specific gravity for each connected cell must be no less than 0.010 below the average of all

BASES

SURVEILLANCE REQUIREMENTS

TABLE 3.10.2-1 (continued)

connected cells. This limit ensures that the effect of a highly charged or new cell does not mask overall degradation of the battery.

Category C defines the limits for each connected cell. These values, although reduced, provide assurance that sufficient capacity exists to perform the intended function and maintain a margin of safety. When any battery parameter is outside the Category C limits, the assurance of sufficient capacity described above no longer exists and the battery must be declared inoperable.

The Category C limits specified for electrolyte level (above the top of the plates and not overflowing) ensure that the plates suffer no physical damage and maintain adequate electron transfer capability. The Category C limits for float voltage is based on IEEE-450 (Ref. 4), which states that a cell voltage of 2.07 V or below, under float conditions and not caused by elevated temperature of the cell, indicates internal cell problems and may require cell replacement.

The Category C limit for specific gravity is the same as that specified for Category A and has been discussed above.

The footnotes to Table 3.10.2-1 are applicable to Category A, B, and C specific gravity. Footnote (b) to Table 3.10.2-1 requires the above mentioned correction for electrolyte level and temperature, with the exception that level correction is not required when battery float current is < 2 amps on float charge. This current provides, in general, an indication of overall battery condition.

Because of specific gravity gradients that are produced during the recharging process, delays of several days may occur while waiting for the specific gravity to stabilize. A stabilized charger current is an acceptable alternative to specific gravity measurement for determining the state of charge. This phenomenon is discussed in IEEE-450 (Ref. 4). Footnote (c) to Table 3.10.2-1 allows the float (charger) current to be used as an alternate to specific gravity for up to 7 days following a battery recharge. When battery float current is verified in lieu of specific gravity, the specific gravity of each connected cell shall be measured prior to expiration of the 7 day allowance. Within 7 days each connected cell's specific gravity must be measured to confirm the state of charge. Following a minor battery recharge (such as equalizing charge that does not follow a deep discharge) specific gravity gradients are not significant, and confirming measurements may be made in less that 7 days.

BASES (continued)

REFERENCES

1. UFSAR, Section 9.6.

2. Oconee Probabilistic Risk Assessment.

3. 10 CFR 50.36.

4. IEEE-450-1980.

5. NRC Letter from L. A. Wiens to J. W. Hampton, "Safety Evaluation for Station Blackout (10 CFR 50.63) - Oconee Nuclear Station, Units 1, 2, and 3," dated March 10, 1992.

6. NRC Letter from L. A. Wiens to J. W. Hampton, "Supplemental Safety Evaluation for Station Blackout (10 CFR 50.63) - Oconee Nuclear Station, Units 1, 2, and 3," dated December 10, 1992.

7. NRC Letter from L. A. Wiens to H. B. Tucker, "Safety Evaluation Report on Effect of Tornado Missiles on Oconee Emergency Feedwater System," dated July 28, 1989.