Tennessee Valley Authority, Sequoyah Nuclear Plant, P.O. Box 2000, Soddy Daisy, Tennessee 37384

November 29, 2017

10 CFR 50.4
10 CFR 50.71(e)

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

Sequoyah Nuclear Plant, Units 1 and 2
Renewed Facility Operating License Nos. DPR-77 and DPR-79
NRC Docket Nos. 50-327 and 50-328

Subject: **Revisions to the Sequoyah Nuclear Plant Units 1 and 2 Technical Specification Bases**

References: TVA Letter to NRC, "Revisions to the Sequoyah Nuclear Plant Units 1 and 2 Technical Specification Bases," dated June 10, 2016.

Pursuant to the SQN Technical Specification 5.5.12, "Technical Specifications (TS) Bases Control Program," these changes to the SQN TS Bases are submitted in accordance with 10 CFR 50.71(e). The previous revisions of the SQN TS Bases were submitted in the referenced letter. The enclosure to this letter provides a description of the TS Bases revisions with attachments of the updated pages.

There are no new regulatory commitments contained in this letter. If you have any questions, please contact Michael McBrearty at (423) 843-7170.

I certify that I am duly authorized by TVA, and that, to the best of my knowledge and belief, the information contained herein accurately presents changes made since the previous submittal, necessary to reflect information and analyses submitted to the Commission or prepared pursuant to Commission requirements.

Respectfully,

Anthony L. Williams
Site Vice President
Sequoyah Nuclear Plant

Enclosure:

Description of Revisions for the Sequoyah Nuclear Plant (SQN), Units 1 and 2 Technical Specification Bases

cc (Enclosure):

NRC Regional Administrator – Region II
NRC Senior Resident Inspector – Sequoyah Nuclear Plant

ENCLOSURE

DESCRIPTION OF REVISIONS FOR THE SEQUOYAH NUCLEAR PLANT (SQN), UNITS 1 AND 2 TECHNICAL SPECIFICATION BASES

Revision 48 to the SQN, Units 1 and 2 Technical Specification (TS) Bases was approved on May 24, 2016, and implemented on May 25, 2016. This Bases change was associated with corrective action program condition report number (CR#) 1169888, wherein it was identified that performance of a high point vent using a specific vent path of the emergency core cooling system was not currently analyzed to ensure 100 percent equivalent flow would exists with the vent path open. To address this condition, Technical Specification (TS) Bases for Surveillance Requirement (SR) 3.5.2.3 was revised to permit ultrasonic testing of piping in lieu of and/or in addition to venting pump casings and accessible high point vents.

Revision 49 to the SQN, Units 1 and 2 TS Bases was approved on July 1, 2016, and implemented on August 22, 2016. Bases changes were made to TS Bases 2.1.1 Reactor Core Safety Limits, 3.1.3 Moderator Temperature Coefficient, 3.2.2 Heat Flux Hot Channel Factors, 3.2.3 Axial Flux Difference, and 3.2.4 Quadrant Power Tilt Ratio. These changes provided clarification to improve reader understanding, corrections of inconsistencies and editorial enhancements.

Revision 50 to the SQN, Units 1 and 2 TS Bases was approved on October 7, 2016, and implemented on October 25, 2016. The Bases change was developed to address CR# 1173123 that identified the exclusion statement for TS 3.6.3, "Containment Isolation Valves," Conditions E, F, and G described in TS Bases 3.6.3 Conditions A, B, and C was poorly worded and could be misapplied if Conditions A, B, or C existed. The unintended exclusion statement "except for containment purge isolation valve, containment vacuum relief isolation valve, or shield building bypass leakage not within leakage limit" was associated with SQN TS Amendment Nos. 334 (Unit 1) and 327 (Unit 2) that were approved by NRC on September 30, 2015. These amendments, 334 and 327, were for conversion of the TSs to improved TSs. TVA did not request a change to the original exclusions in the before mentioned amendments. The original development of the exclusions were a part of Amendment Nos. 323 (Unit 1) and 315 (Unit 2) approved by NRC on April 13, 2009. The revised Bases exclusion statement restores the intended amendments 323 and 315 language, "except for inoperable containment vacuum relief isolation valve(s), or shield building bypass or containment purge isolation valve leakage not within limit,".

Revision 51 to the SQN, Units 1 and 2 TS Bases was approved on March 8, 2016, and implemented on October 25, 2016. This Bases revision is associated with TS Change 16-01, "Extend the Allowed Completion Time to Restore Essential Raw Cooling Water System Train to OPERABLE Status from 72 hours to 7 days," for SQN Units 1 and 2, Amendment Nos. 336 and 329 approved on September 29, 2016. These amendments added a new Condition A to TS 3.7.8, Essential Raw Cooling Water (ERCW) System, to extend the allowed completion time to restore ERCW System train to OPERABLE status from 72 hours to 7 days for planned maintenance when the opposite unit is defueled or in Mode 6 following defueled under certain restrictions. TS Bases 3.7.8, "Essential Raw Cooling Water (ERCW) System," was revised to describe the newly approved Condition A, modifying notes, and associated required actions for Shutdown Board maintenance.

Revision 52 to the SQN, Units 1 and 2 TS Bases was approved on November 23, 2016, and implemented on December 23, 2016. Condition Report no. 1099380 identifies that detailed information, previously in the Bases, describing the support function of the auxiliary control air system and the ERCW system for the auxiliary feedwater system was removed in the conversion of the TSs to improved TSs, SQN TS Amendment Nos. 334 (Unit 1) and 327 (Unit 2). This Base change to TS Bases 3.7.5, “Auxiliary Feedwater,” restore the information that had been previously approved by NRC in letter, “Auxiliary Air Supply To Auxiliary Feedwater System (TAC 76080/76081) (TS 90-10),” dated March 23, 1990, and Amendment Nos. 155 (Unit 1), 182 (Unit 1), and 174 (Unit 2) .

Attachments:

1. Sequoyah Nuclear Plant, Unit 1, Technical Specification Bases - Changed Pages
2. Sequoyah Nuclear Plant, Unit 2, Technical Specification Bases - Changed Pages

ATTACHMENT 1

SEQUOYAH NUCLEAR PLANT, UNIT 1,
TECHNICAL SPECIFICATION BASES
CHANGED PAGES

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AMENDMENT LISTING

Amendments	Date and Revision
Amendment 316 issued by NRC	09/20/07 (R320)
Amendment 317 issued by NRC	09/28/07 (R321)
Bases Revision	12/12/07 (BR-31)
Amendment 318 issued by NRC	04/02/08 (R322)
Bases Revision	08/29/08 (BR-32)
Amendment 319 issued by NRC	08/29/08 (R323)
Bases Revision	08/28/08 (BR-33)
Amendment 320 issued by NRC	09/24/08
Amendment 321 issued by NRC	10/28/08
Amendment 322 issued by NRC	12/04/08
Amendment 323 issued by NRC	04/13/09
Amendment 324 issued by NRC	06/12/09
Bases Revision	06/12/09 (BR-34)
Amendment 325 issued by NRC	08/14/09
Amendment 326 issued by NRC	01/28/10
Amendment 327 issued by NRC	02/02/10
Bases Revision	03/25/10 (BR-35)
Bases Revision	05/27/10 (BR-36)
Amendment 328 issued by NRC	12/21/10
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Bases Revision	03/04/14 (BR-44)
Amendment 333 issued by NRC	09/29/14
Amendment 334 issued by NRC	09/30/15 (ITS)
Bases Revision	10/23/15 (BR-45)
Bases Revision	10/23/15 (BR-46)
Amendment 335 issued by NRC	11/30/15
Bases Revision	04/27/16 (BR-47)
Bases Revision	05/24/16 (BR-48)
Bases Revision	07/01/16 (BR-49)
Amendment 336 issued by NRC	09/29/16
Bases Revision	09/29/16 (BR-51)
Amendment 337 issued by NRC	10/03/16 (License Only)
Bases Revision	10/07/16 (BR-50)
Bases Revision	11/23/16 (BR-52)
Amendment 338 issued by NRC	03/27/17

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BACKGROUND (continued)

confidence level that the minimum DNBR for the limiting rod is greater than or equal to the DNBR limit. This DNBR uncertainty derived from the SCD analysis, combined with the applicable DNB critical heat flux correlation limit, establishes the statistical DNBR design limit which must be met in plant safety analysis using values of input parameters without adjustment for uncertainty.

Operation above the maximum local linear heat generation rate for fuel melting could result in excessive fuel pellet temperature and cause melting of the fuel at its centerline. 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. The melting point of uranium dioxide varies slightly with burnup. As uranium is depleted and fission products produced, the net effect is a decrease in the melting point. Fuel centerline temperature is not a directly measurable parameter during operation. The maximum local fuel pin centerline temperature is maintained by limiting the local linear heat generation rate in the fuel. The local linear heat generation rate in the fuel is limited so that the maximum fuel centerline temperature will not exceed the acceptance criteria in the safety analysis.

The curves provided in Figure 2.1.1-1 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature for which the minimum DNBR is no less than the safety analysis DNBR limit, or the average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid.

These lines are bounding for all fuel types. The curves provided in Figure 2.1.1-1 are based upon enthalpy rise hot channel factors that result in acceptable DNBR performance of each fuel type. Acceptable DNBR performance is assured by operation within the DNB-based Limiting Safety Limit System Settings (Reactor Trip System trip limits). The plant trip setpoints are verified to be less than the limits defined by the safety limit lines provided in Figure 2.1.1-1 converted from power to delta-temperature and adjusted for uncertainty.

The limiting heat flux conditions for DNB are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion assuming the axial power imbalance or Delta-I (ΔI) is within the limits of the f_1 (ΔI) function of the Overtemperature Delta Temperature trip. When the axial power imbalance exceeds the tolerance (or deadband) of the $f_1(\Delta I)$ trip reset function, the Overtemperature Delta Temperature trip setpoint is reduced by the values

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.3.2

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

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

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

b. If the 300 ppm Surveillance limit is not met, it is possible that the EOL limit on MTC could be reached before the planned EOL. Because the MTC changes slowly with core depletion, the Frequency of 14 effective full power days is sufficient to avoid exceeding the EOL limit on MTC.

c. If the measured MTC at 60 ppm is more positive than the 60 ppm Surveillance limit, the EOL limit on MTC will not be exceeded because of the gradual manner in which MTC changes with core burnup. The 60 ppm Surveillance is only performed if the 300 ppm Surveillance limit was not met (see note b). If the 60 ppm Surveillance limit is met, no further Surveillance of EOL MTC is required for the remainder of the fuel cycle. If the 60 ppm Surveillance limit is not met, then Surveillance of EOL MTC is required for the remainder of the fuel cycle as described in note b.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 11.

2. UFSAR, Chapter 15.

3. BAW 10169P-A, "B&W Safety Analysis Methodology for Recirculating Steam Generator Plants," October 1989.

4. UFSAR, Section 15.2.1.

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BACKGROUND (continued)

The COLR provides peaking factor limits that ensure that the design basis value of the DNB is met for normal operation, operational transients, and any transient condition arising from events of moderate frequency. The DNB design basis precludes DNB and is met by limiting the minimum local DNB heat flux ratio to the design limit value using an NRC approved critical heat flux correlation. All DNB limited transient events are assumed to begin with an $F_{\Delta H}(X,Y)$ value that satisfies the LCO requirements.

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

APPLICABLE SAFETY ANALYSES

Limits on $F_{\Delta H}(X,Y)$ preclude core power distributions that exceed the following fuel design limits:

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

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

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

d. Fuel design limits required by GDC 26 (Ref. 2) for the condition when control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn.

For transients that may be DNB limited, the Reactor Coolant System flow and $F_{\Delta H}(X,Y)$ are the core parameters of most importance. The limits on $F_{\Delta H}(X,Y)$ ensure that the DNB design basis is met for normal operation, operational transients, and any transients arising from events of moderate frequency. The DNB design basis is met by limiting the minimum local DNB heat flux ratio to the design limit value using an NRC approved critical heat flux correlation. This value provides a high degree of assurance that the hottest fuel rod in the core does not experience a DNB.

The allowable $F_{\Delta H}(X,Y)$, $F_{\Delta H}$ min margin and $f_1(\Delta I)$ min margin, increase with decreasing power level. This functionality in $F_{\Delta H}(X,Y)$ is

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APPLICABLE SAFETY ANALYSES (continued)

included in the analyses that provide the Reactor Core Safety Limits (SLs) of SL 2.1.1. Therefore, any DNB events in which the calculation of the core limits is modeled implicitly use this variable value of $F_{\Delta H}(X,Y)$ in the analyses. Likewise, all transients that may be DNB limited are assumed to begin with $F_{\Delta H}$ min margin and f_1 (ΔI) min margin.

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

The fuel is protected in part by Technical Specifications, which ensure that the initial conditions assumed in the safety and accident analyses remain valid. The following LCOs ensure this: LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," LCO 3.1.6, "Control Bank Insertion Limits," LCO 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor $F_{\Delta H}(X,Y)$," and LCO 3.2.1, "Heat Flux Hot Channel Factor ($F_Q(X,Y,Z)$)."

$F_{\Delta H}(X,Y)$ and $F_Q(X,Y,Z)$ are indirectly measured periodically using the movable incore detector system. Measurements are generally taken with the core at, or near, steady state conditions. Core monitoring and control under transient conditions (Condition 1 events) are accomplished by operating the core within the limits of the LCOs on AFD, QPTR, and Bank Insertion Limits.

$F_{\Delta H}(X,Y)$ satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO states that $F_{\Delta H}(X,Y)$ shall be less than the limits provided in the COLR. This LCO relationship must be satisfied even if the core is operating at limiting conditions. This requires adjustment to the measured $F_{\Delta H}(X,Y)$ to account for limiting conditions and the differences between design and measured conditions. The adjustments are accounted for by comparing $F\Delta HR^M(X,Y)$ to the limits BHDES(X,Y) and BRDES(X,Y). Therefore, if the $F_{\Delta H}$ min margin is ≥ 0 and $f_1(\Delta I)$ min margin ≥ 0 the LCO is satisfied.

APPLICABILITY

The $F_{\Delta H}(X,Y)$ limits must be maintained in MODE 1 to preclude core power distributions from exceeding the fuel design limits for DNBR and PCT. Applicability in other modes is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the coolant to require a limit on the distribution of core power. Specifically, the design bases events that are sensitive to

BASES

APPLICABLITY (continued)

$F_{\Delta H}(X,Y)$ in other modes (MODES 2 through 5) have significant margin to DNB, and therefore, there is no need to restrict $F_{\Delta H}(X,Y)$ in these modes.

ACTIONS

The % $F_{\Delta H}$ margin is based on the relationship between $F\Delta HR^M(X,Y)$ and the limit, BHDES (X,Y), as follows:

$$\% F_{\Delta H}\ Margin = \left(1 - \frac{F\Delta\ HR^M\ (X,Y)}{BHDES(X,Y)}\right) x\ 100\%$$

If the reactor core is ″operating as designed″, then $F\Delta HR^M(X,Y)$ is less than BHDES (X,Y) and calculation of %$F_{\Delta H}$ margin is not required. If the %$F_{\Delta H}$ margin is less than zero, then $F\Delta HR^M(X,Y)$ is greater than BHDES (X, Y) and the $F_{\Delta H}(X,Y)$ limits may not be adequate to prevent exceeding the initial DNB conditions assumed for transients such as a LOFA. BHDES (X,Y) represents the maximum allowable design radial peaking factors which ensures that the initial condition DNB will be preserved for operation within the LCO limits, and includes allowances for calculational and measurement uncertainties. The $F_{\Delta H}$ min margin is the minimum for all core locations examined.

Condition A is modified by a Note that requires that Required Actions A.3 and A.5 must be completed whenever Condition A is entered. If $F_{\Delta H}$ min margin < 0 is restored to within limits prior to completion of the THERMAL POWER reduction in Required Action A.1, compliance with Required Actions A.3 and A.5 must be met.

However, if power is reduced below 50% RTP, Required Action A.5 requires that another determination of $F_{\Delta H}$ min margin must be verified prior to exceeding 50% RTP, prior to exceeding 75% RTP, and within 24 hours after reaching or exceeding 95% RTP.

A.1 and A.2

If the value of $F_{\Delta H}$ min margin is not restored to within its specified limit either by adjusting a misaligned rod or by reducing THERMAL POWER, the alternative option is to reduce allowable THERMAL POWER from RTP by at least RRH% (where RRH = Thermal power reduction required to compensate for each 1% that $F_{\Delta H}(X,Y)$ exceeds its limit) multiplied by the $F_{\Delta H}$ min margin in accordance with Required Action A.1 and reduce the Power Range Neutron Flux - High trip setpoints, as specified in TS Table 3.3.1-1 by ≥ RRH% multiplied times the $F_{\Delta H}$ min margin in accordance with Required Action A.2. Reducing allowable RTP by at least RRH% multiplied by the $F_{\Delta H}$ min margin increases the DNB

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ACTIONS (continued)

margin and does not likely cause the DNBR limit to be violated in steady state operation. The reduction in trip setpoints ensures that continuing operation remains at an acceptable low power level with adequate DNBR margin. The allowed Completion Time of 2 hours for Required Action A.1 provides an acceptable time to reach the required power level from full power operation without allowing the plant to remain in an unacceptable condition for an extended period of time.

The allowed Completion Time of 72 hours to reset the trip setpoints per Required Action A.2 recognizes that, once power is reduced, the safety analysis assumptions are satisfied and there is no urgent need to reduce the trip setpoints. This is a sensitive operation that may inadvertently trip the Reactor Protection System.

A.3

Once the allowable power level has been reduced by at least RRH% multiplied by the $F_{\Delta H}$ min margin per Required Action A.1, an incore flux map (SR 3.2.2.1) must be obtained and the $F_{\Delta H}$ min margin is verified ≥ 0 at the lower power level. The unit is provided 22 additional hours to perform this task over and above the 2 hours allowed by Action A.1. The Completion Time of 24 hours is acceptable because of the increase in the DNB margin, which is obtained at lower power levels, and the low probability of having a DNB limiting event within this 24 hour period. Additionally, operating experience has indicated that this Completion Time is sufficient to obtain the incore flux map, perform the required calculations, and evaluate $F_{\Delta H}$ min margin.

A.4

If the value of $F\Delta HR^M(X,Y)$ is not restored to within its specified limit, Overtemperature ΔT K1 (OTΔT K1) term is required to be reduced by at least TRH multiplied by the FΔH min margin. The value of TRH is provided in the COLR. Completing Required Action A.4 ensures protection against the consequences of transients since this adjustment limits the peak transient power level which can be achieved during an anticipated operational occurrence. Also, completing Required Action A.4 within the allowed Completion Time of 48 hours is sufficient considering the small likelihood of a limiting transient in this time period.

BASES

APPLICABLE SAFETY ANALYSES (continued)

A Condition 2 event significantly affected by AFD is the Uncontrolled RCCA Bank Withdrawal at Power Event (Ref. 2). Condition 2 accidents, simulated to begin from within the AFD limits are used to confirm the adequacy of the Overpower ΔT and Overtemperature ΔT trip setpoints.

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

LCO

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

Signals are available to the operator from the Nuclear Instrumentation System (NIS) excore neutron detectors (Refs. 1 and 3). Separate signals are taken from the top and bottom detectors. The AFD is defined as the difference in normalized flux signals between the top and bottom excore detectors in each detector well. For convenience, this flux difference is converted to provide flux difference units expressed as a percentage and labeled as %Δ flux or %ΔI.

The AFD limits are provided in the COLR. The AFD limits resulting from analysis of core power distributions relative to the initial condition peaking limits comprise a power-dependent envelope of acceptable AFD values. During steady-state operation, the core normally is controlled to a target AFD within a narrow (approximately ± 5% AFD) band. However, the limiting AFD values may be somewhat greater than the extremes of the normal operating band.

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

APPLICABILITY

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

ACTIONS

A.1

As an alternative to restoring the AFD to within its specified limits, Required Action A.1 requires a THERMAL POWER reduction to < 50% RTP. This places the core in a condition for which the value of the AFD is not important in the applicable safety analyses. A Completion

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 QUADRANT POWER TILT RATIO (QPTR)

BASES

BACKGROUND

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

The power density at any point in the core must be limited so that the fuel design criteria are maintained. Together, LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.2.4, and LCO 3.1.6, "Control Bank Insertion Limits," provide limits on process variables that characterize and control the three dimensional power distribution of the reactor core. Control of these variables ensures that the core operates within the fuel design criteria and that the power distribution remains within the bounds used in the safety analyses.

APPLICABLE SAFETY ANALYSES

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

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

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

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

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

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

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

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.2.2

Verifying the correct alignment for manual, power operated, and automatic valves in the ECCS flow paths provides assurance that the proper flow paths will exist for ECCS 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. A valve that receives an actuation signal is allowed to be in a nonaccident position provided the valve will automatically reposition within the proper stroke time. 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 Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.5.2.3

With the exception of the operating centrifugal charging pump, the ECCS pumps are normally in a standby, nonoperating mode. As such, flow path piping has the potential to develop voids and pockets of entrained gases. Maintaining the piping from the ECCS pumps to the RCS full of water ensures that the system will perform properly, injecting its full capacity into the RCS upon demand. ECCS piping is verified full of water by venting and/or ultrasonic testing (UT) pump casings and accessible high point vents. This will also prevent water hammer, pump cavitation, and pumping of noncondensible gas (e.g., air, nitrogen, or hydrogen) into the reactor vessel following an SI signal or during shutdown cooling.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.5.2.4

Periodic surveillance testing of ECCS pumps to detect gross degradation caused by impeller structural damage or other hydraulic component problems is required by the ASME Code. This type of testing may be accomplished by measuring the pump developed head at only one point of the pump characteristic curve. This verifies both that the measured performance is within an acceptable tolerance of the original pump baseline performance and that the performance at the test flow is greater than or equal to the performance assumed in the plant safety analysis. SRs are specified in the Inservice Testing Program of the ASME Code.

BASES

ACTIONS (continued)

In the event the isolation valve leakage results in exceeding the overall containment leakage rate, Note 4 directs entry into the applicable Conditions and Required Actions of LCO 3.6.1.

Note 5 limits the number of open containment purge lines to no more than one set of supply valves and one set of exhaust valves.

A.1 and A.2

Condition A is applicable to penetration flow paths with two containment isolation valves, and penetration flow paths with only one containment isolation valve and a closed system. The closed system must meet the requirements of Reference 3.

In the event one containment isolation valve in one or more penetration flow paths is inoperable, except for inoperable containment vacuum relief isolation valve(s), or shield building bypass or containment purge isolation valve leakage not within limit, 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 manual valve, a blind flange, and a check valve with flow through the valve secured. For a penetration flow path 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 Completion Time specified for each Category of containment isolation valve identified in Table B 3.6.3-1, Containment Isolation Valve Completion Times. The Completion Time is justified in Reference 4.

For affected penetration flow paths that cannot be restored to OPERABLE status within the specified 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 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 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

BASES

ACTIONS (continued)

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.

Required Action A.2 is modified by two Notes. Note 1 applies to isolation devices 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. Note 2 applies to isolation devices that are locked, sealed, or otherwise secured in position and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since the function of locking, sealing, or securing components is to ensure that these devices are not inadvertently repositioned. Therefore, the probability of misalignment of these devices once they have been verified to be in the proper position, is small.

<u>B.1</u>

With two containment isolation valves in one or more penetration flow paths inoperable, except for inoperable containment vacuum relief isolation valve(s), or shield building bypass or containment purge isolation valve leakage not within limit, 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 automatic valve, a closed manual valve, and a blind flange. 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 control 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.

BASES

ACTIONS (continued)

C.1

In the event one containment isolation valve in two or more penetration flow paths is inoperable except for inoperable containment vacuum relief isolation valve(s), or shield building bypass or containment purge isolation valve leakage not within limit, all but one of the affected penetration flow path(s) 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 manual valve, a blind flange, and a check valve with flow through the valve secured. For a penetration flow path isolated in accordance with Required Action C.1, the device used to isolate the penetration should be the closest available one to containment. Required Action C.1 must be completed within 4 hours. For subsequent containment isolation valve inoperabilities, the Required Action and Completion Time continue to apply to each additional containment isolation valve inoperability, with the Completion Time based on each subsequent entry into the Condition consistent with Note 2 to the ACTIONS Table (e.g., for each entry into the Condition). Each containment isolation valve(s) that is (are) declared inoperable for subsequent Condition C entries shall meet the Required Action and Completion Time. For the penetration flow paths isolated in accordance with Required Action C.1, the affected penetration(s) 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 that the penetrations requiring isolation following an accident are isolated. The 4 hour Completion Time 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.

D.1 and D.2

In the event two or more pairs of containment purge lines are open, all but one penetration flow paths 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 manual valve, and a blind flange. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1. The 1 hour Completion Time 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

LCO (continued)

Each motor-driven auxiliary feedwater pump (one Train A and one Train B) supplies flow paths to two steam generators. Each flow path contains an automatic air-operated level control valve (LCV). The LCVs have the same train designation as the associated pump and are provided trained air. The turbine driven auxiliary feedwater pump supplies flow paths to all four steam generators. Each of these flow paths contains an automatic opening (non-modulating) air-operated LCV, two of which are designated as Train A, receive A-train air, and provide flow to the same steam generators that are supplied by the B-train motor-driven auxiliary feedwater pump. The remaining two LCVs are designated as Train B, receive B-train air, and provide flow to the same steam generators that are supplied by the A-train motor-driven pump. This design provides the required redundancy to ensure that at least two steam generators receive the necessary flow assuming any single failure. It can be seen from the description provided above that the loss of a single train of air (A or B) will not prevent the auxiliary feedwater system from performing its intended safety function and is no more severe than the loss of a single auxiliary feedwater pump. Therefore, the loss of a single train of auxiliary air only affects the capability of a single motor-driven auxiliary feedwater pump because the turbine-driven pump is still capable of providing flow to two steam generators that are separate from the other motor-driven pump.

Two redundant steam sources are required to be operable to ensure that at least one source is available for the steam-driven auxiliary feedwater (AFW) pump operation following a feedwater or main steam line break. This requirement ensures that the plant remains within its design basis (i.e., AFW to two intact steam generators) given the event of a loss of the No.1 steam generator because of a main steam line or feedwater line break and a single failure of the B-train motor driven AFW pump. The two redundant sources must be aligned such that No.1 steam generator source is open and operable and the No.4 steam generator source is closed and operable.

For instances where one train of emergency raw cooling water (ERCW) is declared inoperable in accordance with technical specifications, the AFW turbine-driven pump is considered operable since it is supplied by both trains of ERCW. Similarly, the AFW turbine-driven pump is considered operable when one train of the AFW loss of power start function is declared inoperable in accordance with Technical Specifications because both 6.9 kilovolt shutdown board logic trains supply this function. This position is consistent with American National Standards Institute/ANS 58.9 requirements (i.e., postulation of the failure of the opposite train is not required while relying on the TS limiting condition for operation).

BASES

LCO (continued)

The LCO is modified by a Note indicating that one AFW train, which includes a motor driven pump, is required to be OPERABLE in MODE 4. This is because of the reduced heat removal requirements and short period of time in MODE 4 during which the AFW is required and the insufficient steam available in MODE 4 to power the turbine driven AFW pump.

APPLICABILITY

In MODES 1, 2, and 3, the AFW System is required to be OPERABLE in the event it is called upon to function when MFW is lost. In addition, the AFW System is required to supply enough makeup water to replace the steam generator secondary inventory, lost as the unit cools to MODE 4 conditions.

In MODE 4 the AFW System may be used for heat removal via the steam generators.

In MODE 5 or 6, the steam generators are not normally used for heat removal, and the AFW System is not required.

ACTIONS

A Note prohibits the application of LCO 3.0.4.b to an inoperable AFW train. There is an increased risk associated with entering a MODE or other specified condition in the Applicability with an AFW train inoperable and the provisions of LCO 3.0.4.b, which allow entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, should not be applied in this circumstance.

A.1

If the turbine driven AFW train is inoperable due to one inoperable steam supply, or if a turbine driven pump is inoperable for any reason while in MODE 3 immediately following refueling, action must be taken to restore the inoperable equipment to an OPERABLE status within 7 days. The 7 day Completion Time is reasonable, based on the following reasons:

a. For the inoperability of the turbine driven AFW pump due to one inoperable steam supply, the 7 day Completion Time is reasonable since there is a redundant steam supply line for the turbine driven pump and the turbine driven train is still capable of performing its specified function for most postulated events.

b. For the inoperability of a turbine driven AFW pump while in MODE 3 immediately subsequent to a refueling, the 7 day Completion Time is reasonable due to the minimal decay heat levels in this situation.

BASES

ACTIONS (continued)

c. For both the inoperability of the turbine driven pump due to one inoperable steam supply and an inoperable turbine driven AFW pump while in MODE 3 immediately following a refueling outage, the 7 day Completion Time is reasonable due to the availability of redundant OPERABLE motor driven AFW pumps, and due to the low probability of an event requiring the use of the turbine driven AFW pump.

Condition A is modified by a Note which limits the applicability of the Condition for an inoperable turbine driven AFW pump in MODE 3 to when the unit has not entered MODE 2 following refueling. Condition A allows one AFW train to be inoperable for 7 days vice the 72 hour Completion Time in Condition B. This longer Completion Time is based on the reduced decay heat following refueling and prior to the reactor being critical.

B.1

With one of the required AFW trains (pump or flow path) inoperable in MODE 1, 2, or 3 for reasons other than Condition A, action must be taken to restore OPERABLE status within 72 hours. This Condition includes the loss of two steam supply lines to the turbine driven AFW pump. The 72 hour Completion Time is reasonable, based on the redundant capabilities afforded by the AFW System, the time needed for repairs, and the low probability of a DBA occurring during this time period.

C.1 and C.2

With one of the required motor driven AFW trains (pump or flow path) inoperable and the turbine driven AFW train inoperable due to one inoperable steam supply, action must be taken to restore the affected equipment to OPERABLE status within 48 hours. Assuming no single active failures when in this condition, the accident (a feedline break (FLB) or main steam line break (MSLB)) could result in the loss of the remaining steam supply to the turbine driven AFW pump due to the faulted steam generator (SG).

BASES

ACTIONS (continued)

The 48 hour Completion Time is reasonable based on the fact that the remaining motor driven AFW train is capable of providing 100% of the AFW flow requirements, and the low probability of an event occurring that would challenge the AFW system.

D.1 and D.2

When Required Action A.1, B.1, C.1, or C.2 cannot be completed within the required Completion Time, or if two AFW trains are inoperable in MODE 1, 2, or 3 for reasons other than Condition C, 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 6 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.

In MODE 4 with two AFW trains inoperable, operation is allowed to continue because only one motor driven pump AFW train is required in accordance with the Note that modifies the LCO. Although not required, the unit may continue to cool down and initiate RHR.

E.1

If all three AFW trains are inoperable in MODE 1, 2, or 3, the unit is in a seriously degraded condition with no safety related means for conducting a cooldown, and only limited means for conducting a cooldown with nonsafety related equipment. 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 one AFW train to OPERABLE status.

Required Action E.1 is modified by a Note indicating that all required MODE changes are suspended until one AFW train is restored to OPERABLE status. In this case, LCO 3.0.3 is not applicable because it could force the unit into a less safe condition.

BASES

ACTIONS (continued)

F.1

In MODE 4, either the reactor coolant pumps or the RHR loops can be used to provide forced circulation. This is addressed in LCO 3.4.6, "RCS Loops - MODE 4." With the required AFW train inoperable, action must be taken to immediately restore the inoperable train to OPERABLE status. The immediate Completion Time is consistent with LCO 3.4.6.

SURVEILLANCE REQUIREMENTS

SR 3.7.5.1

Verifying the correct alignment for manual, power operated, and automatic valves in the AFW System water and steam supply flow paths provides assurance that the proper flow paths will exist for AFW 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 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 being mispositioned are in the correct position.

The SR is modified by a Note that states one or more AFW trains may be considered OPERABLE during alignment and operation for steam generator level control, if it is capable of being manually (i.e., remotely or locally, as appropriate) realigned to the AFW mode of operation, provided it is not otherwise inoperable. This exception allows the system to be out of its normal standby alignment and temporarily incapable of automatic initiation without declaring the train(s) inoperable. Since AFW may be used during startup, shutdown, hot standby operations, and hot shutdown operations for steam generator level control, and these manual operations are an accepted function of the AFW System, OPERABILITY (i.e., the intended safety function) continues to be maintained.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.7.5.2

Verifying that each AFW pump's developed head at the flow test point is greater than or equal to the required developed head ensures that AFW pump performance has not degraded during the cycle. Flow and differential head are normal tests of centrifugal pump performance required by the ASME Code (Ref 2). Because it is undesirable to introduce cold AFW into the steam generators while they are operating,

BASES

SURVEILLANCE REQUIREMENTS (continued)

this testing is performed 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 detect incipient failures by indicating abnormal performance. Performance of inservice testing discussed in the ASME Code (Ref. 2) (only required at 3 month intervals) satisfies this requirement.

This SR is modified by a Note indicating that the SR should be deferred until suitable test conditions are established. This deferral is required because there is insufficient steam pressure to perform the test.

SR 3.7.5.3

This SR verifies that AFW can be delivered to the appropriate steam generator in the event of any accident or transient that generates an ESFAS, or in the event the CSTs become depleted, by demonstrating that each automatic valve in the flow path actuates to its correct position on an actual or simulated actuation signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

The SR is modified by two Notes. Note 1 states that one or more AFW trains may be considered OPERABLE during alignment and operation for steam generator level control, if it is capable of being manually (i.e., remotely or locally, as appropriate) realigned to the AFW mode of operation, provided it is not otherwise inoperable. This exception allows the system to be out of its normal standby alignment and temporarily incapable of automatic initiation without declaring the train(s) inoperable. Since AFW may be used during startup, shutdown, hot standby operations, and hot shutdown operations for steam generator level control, and these manual operations are an accepted function of the AFW System, OPERABILITY (i.e., the intended safety function) continues to be maintained. Note 2 states that the SR is only required to be met in MODES 1, 2, and 3. It is not required to be met in MODE 4, since the AFW train is only required for the purposes of removing decay heat when the SG is relied upon for heat removal. The operation of the AFW train is by manual means and automatic startup of the AFW train is not required.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.5.4

This SR verifies that the AFW pumps will start in the event of any accident or transient that generates an ESFAS by demonstrating that each AFW pump starts automatically on an actual or simulated actuation signal in MODES 1, 2, and 3. In MODE 4, the required pump is already operating and the autostart function is not required.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

This SR is modified by three Notes. Note 1 indicates that the SR may be deferred until suitable test conditions are established. This deferral is required because there is insufficient steam pressure to perform the test. Note 2 states that one or more AFW trains may be considered OPERABLE during alignment and operation for steam generator level control, if it is capable of being manually (i.e., remotely or locally, as appropriate) realigned to the AFW mode of operation, provided it is not otherwise inoperable. This exception allows the system to be out of its normal standby alignment and temporarily incapable of automatic initiation without declaring the train(s) inoperable. Since AFW may be used during startup, shutdown, hot standby operations, and hot shutdown operations for steam generator level control, and these manual operations are an accepted function of the AFW System. OPERABILITY (i.e., the intended safety function) continues to be maintained. Note 3 states that the SR is only required to be met in MODES 1, 2, and 3. It is not required to be met in MODE 4, since the AFW train is only required for the purposes of removing decay heat when the SG is relied upon for heat removal. The operation of the AFW train is by manual means and automatic startup of the AFW train is not required.

REFERENCES

1. UFSAR, Section 10.4.7.2.
2. ASME Code for Operation and Maintenance of Nuclear Power Plants.

BASES

APPLICABLE SAFETY ANALYSES (continued)

(Ref. 3) entry conditions to MODE 5 during normal and post accident operations. The time required for this evolution is a function of the number of component cooling water and RHR System trains that are operating. One ERCW system train is sufficient to remove decay heat during subsequent operations in MODES 5 and 6. This assumes a maximum ERCW system temperature of 87°F occurring simultaneously with maximum heat loads on the system.

The ERCW system satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Two ERCW system trains are required to be OPERABLE to provide the required redundancy to ensure that the system functions to remove post accident heat loads, assuming that the worst case single active failure occurs coincident with the loss of offsite power.

An ERCW system train is considered OPERABLE during MODES 1, 2, 3, and 4 when the required ERCW pumps are operable and the associated piping, valves, heat exchanger, and instrumentation and controls require to perform the safety related function as described in UFSAR Section 9.2.2.

APPLICABILITY

In MODES 1, 2, 3, and 4, the ERCW system is a normally operating system that is required to support the OPERABILITY of the equipment serviced by the ERCW system and required to be OPERABLE in these MODES.

In MODES 5 and 6, the OPERABILITY requirements of the ERCW system are determined by the systems it supports.

ACTIONS

A.1 and A.2

Condition A is modified by two Notes that limit the conditions and parameters that allow entry into Condition A. The first Note limits the applicability of Condition A to the time period when the opposite unit is either defueled or in MODE 6 following defueled with refueling water cavity level ≥ 23 ft. above the top of the reactor vessel flange. The second Note requires a temperature limitation on UHS Temperature. In

BASES

ACTIONS (continued)

order to credit the temperature limit, the effected ERCW train must be aligned in accordance with UFSAR 9.2.2.2. This will allow the plant configuration to be aligned (i.e., cross-ties exist and isolation of loads to facilitate maintenance activities) to minimize the heat load on the ERCW system to ensure the ERCW system continues to meet its design function.

The 7 day Completion Time is acceptable based on the following:

- The low probability of a DBA occurring during that time;
- The heat load on the ERCW System is substantially lower than assumed for the DBA with the opposite unit defueled or subsequent to defueled; and
- The redundant capabilities afforded by the OPERABLE train.

If one ERCW system train is inoperable for planned maintenance, action must be taken to restore OPERABLE status within 7 days. In this Condition, the remaining OPERABLE ERCW system train is adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure in the OPERABLE ERCW system train could result in loss of ERCW system function.

Required Action A.1 is modified by two Notes. The first Note indicates that the applicable Conditions and Required Actions of LCO 3.8.1, “AC Sources – Operating,” should be entered if an inoperable ERCW system train results in an inoperable emergency diesel generator. The second Note indicates that the applicable Conditions and Required Actions of LCO 3.4.6, “RCS Loops – MODE 4,” should be entered if an inoperable ERCW system train results in an inoperable residual heat removal loop. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components.

Required Action A.2 ensures the credited temperature limit for Ultimate Heat Sink is maintained.

B.1

If one ERCW system train is inoperable for reasons other than Condition A, action must be taken to restore OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE ERCW system train is adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure in the OPERABLE ERCW system train could result in loss of ERCW system function. Required Action B.1 is modified by two Notes.

BASES

ACTIONS (continued)

The first Note indicates that the applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources - Operating," should be entered if an inoperable ERCW system train results in an inoperable emergency diesel generator. The second Note indicates that the applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops - MODE 4," should be entered if an inoperable ERCW system train results in an inoperable residual heat removal loop. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components. The 72 hour Completion Time is based on the redundant capabilities afforded by the OPERABLE train, and the low probability of a DBA occurring during this time period.

C.1 and C.2

If the ERCW system train 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 6 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 3.7.8.1

This SR is modified by a Note indicating that the isolation of the ERCW system components or systems may render those components inoperable, but does not affect the OPERABILITY of the ERCW system.

Verifying the correct alignment for manual, power operated, and automatic valves in the ERCW system flow path provides assurance that the proper flow paths exist for ERCW 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 being locked, sealed, or secured. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.8.2

This SR verifies proper automatic operation of the ERCW system valves on an actual or simulated actuation signal. The Safety Injection signal is the automatic actuation signal. The ERCW system is a normally operating system that cannot be fully actuated as part of normal testing. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.7.8.3

This SR verifies proper automatic operation of the ERCW system pumps on an actual or simulated (i.e., Safety Injection) actuation signal. The ERCW system is a normally operating system that cannot be fully actuated as part of normal testing during normal operation.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. UFSAR, Section 9.2.2.
2. UFSAR, Section 6.2.
3. UFSAR, Section 5.5.7.

ATTACHMENT 2

SEQUOYAH NUCLEAR PLANT, UNIT 2,
TECHNICAL SPECIFICATION BASES
CHANGED PAGES

TS Bases Affected Pages
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AMENDMENT LISTING

Amendments	Date and Revision
Amendment 308 Issued by NRC	10/11/07 (R308)
Bases Revision	12/12/07 (BR-30)
Amendment 309 Issued by NRC	03/24/08 (R309)
Amendment 310 Issued by NRC	04/02/08 (R310)
Amendment 311 Issued by NRC	04/04/08 (R311)
Amendment 312 Issued by NRC	08/29/08 (R312)
Bases Revision	08/29/08 (BR-31)
Bases Revision	08/28/08 (BR-32)
Amendment 313 Issued by NRC	10/28/08
Amendment 314 Issued by NRC	12/04/08
Amendment 315 Issued by NRC	04/13/09
Amendment 316 Issued by NRC	06/12/09
Bases Revision	06/12/09 (BR-33)
Amendment 317 Issued by NRC	08/14/09
Amendment 318 Issued by NRC	10/19/09
Bases Revision	10/19/09 (BR-34)
Amendment 319 Issued by NRC	01/28/10
Amendment 320 Issued by NRC	02/02/10
Bases Revision	03/25/10 (BR-35)
Bases Revision	05/27/10 (BR-36)
Amendment 321 Issued by NRC	12/21/10
Bases Revision	03/24/12 (BR-38)
Amendment 322 Issued by NRC	07/29/11
Amendment 323 Issued by NRC	07/10/12
Bases Revision	10/05/12 (BR-40)
Amendment 324 Issued by NRC	09/26/12
Bases Revision	10/10/12 (BR-39)
Amendment 325 Issued by NRC	10/31/12
Bases Revision	12/21/12 (BR-41)
Bases Revision	03/05/13 (BR-42)
Bases Revision	01/31/14 (BR-43)
Bases Revision	03/04/14 (BR-44)
Amendment 326 Issued by NRC	09/29/14
Amendment 327 Issued by NRC	09/30/15 (ITS)
Bases Revision	10/23/15 (BR-45)
Bases Revision	10/23/15 (BR-46)
Amendment 328 Issued by NRC	11/30/15
Bases Revision	04/27/16 (BR-47)
Bases Revision	05/24/16 (BR-48)
Bases Revision	07/01/16 (BR-49)
Amendment 329 Issued by NRC	09/29/16
Bases Revision	09/29/16 (BR-51)
Amendment 330 Issued by NRC	10/03/16 (License Only)
Bases Revision	10/07/16 (BR-50)
Bases Revision	11/23/16 (BR-52)
Amendment 331 Issued by NRC	03/27/17

BASES

BACKGROUND (continued)

confidence level that the minimum DNBR for the limiting rod is greater than or equal to the DNBR limit. This DNBR uncertainty derived from the SCD analysis, combined with the applicable DNB critical heat flux correlation limit, establishes the statistical DNBR design limit which must be met in plant safety analysis using values of input parameters without adjustment for uncertainty.

Operation above the maximum local linear heat generation rate for fuel melting could result in excessive fuel pellet temperature and cause melting of the fuel at its centerline. 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. The melting point of uranium dioxide varies slightly with burnup. As uranium is depleted and fission products produced, the net effect is a decrease in the melting point. Fuel centerline temperature is not a directly measurable parameter during operation. The maximum local fuel pin centerline temperature is maintained by limiting the local linear heat generation rate in the fuel. The local linear heat generation rate in the fuel is limited so that the maximum fuel centerline temperature will not exceed the acceptance criteria in the safety analysis.

The curves provided in Figure 2.1.1-1 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature for which the minimum DNBR is no less than the safety analysis DNBR limit, or the average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid.

These lines are bounding for all fuel types. The curves provided in Figure 2.1.1-1 are based upon enthalpy rise hot channel factors that result in acceptable DNBR performance of each fuel type. Acceptable DNBR performance is assured by operation within the DNB-based Limiting Safety Limit System Settings (Reactor Trip System trip limits). The plant trip setpoints are verified to be less than the limits defined by the safety limit lines provided in Figure 2.1.1-1 converted from power to delta-temperature and adjusted for uncertainty.

The limiting heat flux conditions for DNB are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion assuming the axial power imbalance or Delta-I (ΔI) is within the limits of the f_1 (ΔI) function of the Overtemperature Delta Temperature trip. When the axial power imbalance exceeds the tolerance (or deadband) of the $f_1(\Delta I)$ trip reset function, the Overtemperature Delta Temperature trip setpoint is reduced by the values

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.3.2

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

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

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

b. If the 300 ppm Surveillance limit is not met, it is possible that the EOL limit on MTC could be reached before the planned EOL. Because the MTC changes slowly with core depletion, the Frequency of 14 effective full power days is sufficient to avoid exceeding the EOL limit on MTC.

c. If the measured MTC at 60 ppm is more positive than the 60 ppm Surveillance limit, the EOL limit on MTC will not be exceeded because of the gradual manner in which MTC changes with core burnup. The 60 ppm Surveillance is only performed if the 300 ppm Surveillance limit was not met (see note b). If the 60 ppm Surveillance limit is met, no further Surveillance of EOL MTC is required for the remainder of the fuel cycle. If the 60 ppm Surveillance limit is not met, then Surveillance of EOL MTC is required for the remainder of the fuel cycle as described in note b.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 11.

2. UFSAR, Chapter 15.

3. BAW 10169P-A, "B&W Safety Analysis Methodology for Recirculating Steam Generator Plants," October 1989.

4. UFSAR, Section 15.2.1.

BASES

BACKGROUND (continued)

The COLR provides peaking factor limits that ensure that the design basis value of the DNB is met for normal operation, operational transients, and any transient condition arising from events of moderate frequency. The DNB design basis precludes DNB and is met by limiting the minimum local DNB heat flux ratio to the design limit value using an NRC approved critical heat flux correlation. All DNB limited transient events are assumed to begin with an $F_{\Delta H}(X,Y)$ value that satisfies the LCO requirements.

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

APPLICABLE SAFETY ANALYSES

Limits on $F_{\Delta H}(X,Y)$ preclude core power distributions that exceed the following fuel design limits:

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

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

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

d. Fuel design limits required by GDC 26 (Ref. 2) for the condition when control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn.

For transients that may be DNB limited, the Reactor Coolant System flow and $F_{\Delta H}(X,Y)$ are the core parameters of most importance. The limits on $F_{\Delta H}(X,Y)$ ensure that the DNB design basis is met for normal operation, operational transients, and any transients arising from events of moderate frequency. The DNB design basis is met by limiting the minimum local DNB heat flux ratio to the design limit value using an NRC approved critical heat flux correlation. This value provides a high degree of assurance that the hottest fuel rod in the core does not experience a DNB.

The allowable $F_{\Delta H}(X,Y)$, $F_{\Delta H}$ min margin and $f_1(\Delta I)$ min margin, increase with decreasing power level. This functionality in $F_{\Delta H}(X,Y)$ is

BASES

APPLICABLE SAFETY ANALYSES (continued)

included in the analyses that provide the Reactor Core Safety Limits (SLs) of SL 2.1.1. Therefore, any DNB events in which the calculation of the core limits is modeled implicitly use this variable value of $F_{\Delta H}(X,Y)$ in the analyses. Likewise, all transients that may be DNB limited are assumed to begin with $F_{\Delta H}$ min margin and f_1 (ΔI) min margin.

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

The fuel is protected in part by Technical Specifications, which ensure that the initial conditions assumed in the safety and accident analyses remain valid. The following LCOs ensure this: LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," LCO 3.1.6, "Control Bank Insertion Limits," LCO 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor $F_{\Delta H}(X,Y)$," and LCO 3.2.1, "Heat Flux Hot Channel Factor ($F_Q(X,Y,Z)$)."

$F_{\Delta H}(X,Y)$ and $F_Q(X,Y,Z)$ are indirectly measured periodically using the movable incore detector system. Measurements are generally taken with the core at, or near, steady state conditions. Core monitoring and control under transient conditions (Condition 1 events) are accomplished by operating the core within the limits of the LCOs on AFD, QPTR, and Bank Insertion Limits.

$F_{\Delta H}(X,Y)$ satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO states that $F_{\Delta H}(X,Y)$ shall be less than the limits provided in the COLR. This LCO relationship must be satisfied even if the core is operating at limiting conditions. This requires adjustment to the measured $F_{\Delta H}(X,Y)$ to account for limiting conditions and the differences between design and measured conditions. The adjustments are accounted for by comparing $F\Delta HR^M(X,Y)$ to the limits BHDES(X,Y) and BRDES(X,Y). Therefore, if the $F_{\Delta H}$ min margin is ≥ 0 and $f_1(\Delta I)$ min margin ≥ 0 the LCO is satisfied.

APPLICABILITY

The $F_{\Delta H}(X,Y)$ limits must be maintained in MODE 1 to preclude core power distributions from exceeding the fuel design limits for DNBR and PCT. Applicability in other modes is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the coolant to require a limit on the distribution of core power. Specifically, the design bases events that are sensitive to

BASES

APPLICABLITY (continued)

$F_{\Delta H}(X,Y)$ in other modes (MODES 2 through 5) have significant margin to DNB, and therefore, there is no need to restrict $F_{\Delta H}(X,Y)$ in these modes.

ACTIONS

The % $F_{\Delta H}$ margin is based on the relationship between $F\Delta HR^M(X,Y)$ and the limit, BHDES (X,Y), as follows:

$$\% \, F_{\Delta H} \, Margin = \left(1 - \frac{F\Delta \, HR^M \, (X,Y)}{BHDES(X,Y)}\right) x \, 100\%$$

If the reactor core is "operating as designed", then $F\Delta HR^M(X,Y)$ is less than BHDES (X,Y) and calculation of $\%F_{\Delta H}$ margin is not required. If the $\%F_{\Delta H}$ margin is less than zero, then $F\Delta HR^M(X,Y)$ is greater than BHDES (X, Y) and the $F_{\Delta H}(X,Y)$ limits may not be adequate to prevent exceeding the initial DNB conditions assumed for transients such as a LOFA. BHDES (X,Y) represents the maximum allowable design radial peaking factors which ensures that the initial condition DNB will be preserved for operation within the LCO limits, and includes allowances for calculational and measurement uncertainties. The $F_{\Delta H}$ min margin is the minimum for all core locations examined.

Condition A is modified by a Note that requires that Required Actions A.3 and A.5 must be completed whenever Condition A is entered. If $F_{\Delta H}$ min margin < 0 is restored to within limits prior to completion of the THERMAL POWER reduction in Required Action A.1, compliance with Required Actions A.3 and A.5 must be met.

However, if power is reduced below 50% RTP, Required Action A.5 requires that another determination of $F_{\Delta H}$ min margin must be verified prior to exceeding 50% RTP, prior to exceeding 75% RTP, and within 24 hours after reaching or exceeding 95% RTP.

A.1 and A.2

If the value of $F_{\Delta H}$ min margin is not restored to within its specified limit either by adjusting a misaligned rod or by reducing THERMAL POWER, the alternative option is to reduce allowable THERMAL POWER from RTP by at least RRH% (where RRH = Thermal power reduction required to compensate for each 1% that $F_{\Delta H}(X,Y)$ exceeds its limit) multiplied by the $F_{\Delta H}$ min margin in accordance with Required Action A.1 and reduce the Power Range Neutron Flux - High trip setpoints, as specified in TS Table 3.3.1-1 by ≥ RRH% multiplied times the $F_{\Delta H}$ min margin in accordance with Required Action A.2. Reducing allowable RTP by at least RRH% multiplied by the $F_{\Delta H}$ min margin increases the DNB

BASES

ACTIONS (continued)

margin and does not likely cause the DNBR limit to be violated in steady state operation. The reduction in trip setpoints ensures that continuing operation remains at an acceptable low power level with adequate DNBR margin. The allowed Completion Time of 2 hours for Required Action A.1 provides an acceptable time to reach the required power level from full power operation without allowing the plant to remain in an unacceptable condition for an extended period of time.

The allowed Completion Time of 72 hours to reset the trip setpoints per Required Action A.2 recognizes that, once power is reduced, the safety analysis assumptions are satisfied and there is no urgent need to reduce the trip setpoints. This is a sensitive operation that may inadvertently trip the Reactor Protection System.

A.3

Once the allowable power level has been reduced by at least RRH% multiplied by the $F_{\Delta H}$ min margin per Required Action A.1, an incore flux map (SR 3.2.2.1) must be obtained and the $F_{\Delta H}$ min margin is verified ≥ 0 at the lower power level. The unit is provided 22 additional hours to perform this task over and above the 2 hours allowed by Action A.1. The Completion Time of 24 hours is acceptable because of the increase in the DNB margin, which is obtained at lower power levels, and the low probability of having a DNB limiting event within this 24 hour period. Additionally, operating experience has indicated that this Completion Time is sufficient to obtain the incore flux map, perform the required calculations, and evaluate $F_{\Delta H}$ min margin.

A.4

If the value of $F\Delta HR^M(X,Y)$ is not restored to within its specified limit, Overtemperature ΔT K1 (OTΔT K1) term is required to be reduced by at least TRH multiplied by the FΔH min margin. The value of TRH is provided in the COLR. Completing Required Action A.4 ensures protection against the consequences of transients since this adjustment limits the peak transient power level which can be achieved during an anticipated operational occurrence. Also, completing Required Action A.4 within the allowed Completion Time of 48 hours is sufficient considering the small likelihood of a limiting transient in this time period.

BASES

APPLICABLE SAFETY ANALYSES (continued)

A Condition 2 event significantly affected by AFD is the Uncontrolled RCCA Bank Withdrawal at Power Event (Ref. 2). Condition 2 accidents, simulated to begin from within the AFD limits are used to confirm the adequacy of the Overpower ΔT and Overtemperature ΔT trip setpoints.

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

LCO

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

Signals are available to the operator from the Nuclear Instrumentation System (NIS) excore neutron detectors (Refs. 1 and 3). Separate signals are taken from the top and bottom detectors. The AFD is defined as the difference in normalized flux signals between the top and bottom excore detectors in each detector well. For convenience, this flux difference is converted to provide flux difference units expressed as a percentage and labeled as %Δ flux or %ΔI.

The AFD limits are provided in the COLR. The AFD limits resulting from analysis of core power distributions relative to the initial condition peaking limits comprise a power-dependent envelope of acceptable AFD values. During steady-state operation, the core normally is controlled to a target AFD within a narrow (approximately ± 5% AFD) band. However, the limiting AFD values may be somewhat greater than the extremes of the normal operating band.

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

APPLICABILITY

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

ACTIONS

A.1

As an alternative to restoring the AFD to within its specified limits, Required Action A.1 requires a THERMAL POWER reduction to < 50% RTP. This places the core in a condition for which the value of the AFD is not important in the applicable safety analyses. A Completion

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 QUADRANT POWER TILT RATIO (QPTR)

BASES

BACKGROUND

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

The power density at any point in the core must be limited so that the fuel design criteria are maintained. Together, LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.2.4, and LCO 3.1.6, "Control Bank Insertion Limits," provide limits on process variables that characterize and control the three dimensional power distribution of the reactor core. Control of these variables ensures that the core operates within the fuel design criteria and that the power distribution remains within the bounds used in the safety analyses.

APPLICABLE SAFETY ANALYSES

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

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

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

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

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

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

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

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.2.2

Verifying the correct alignment for manual, power operated, and automatic valves in the ECCS flow paths provides assurance that the proper flow paths will exist for ECCS 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. A valve that receives an actuation signal is allowed to be in a nonaccident position provided the valve will automatically reposition within the proper stroke time. 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 Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.5.2.3

With the exception of the operating centrifugal charging pump, the ECCS pumps are normally in a standby, nonoperating mode. As such, flow path piping has the potential to develop voids and pockets of entrained gases. Maintaining the piping from the ECCS pumps to the RCS full of water ensures that the system will perform properly, injecting its full capacity into the RCS upon demand. ECCS piping is verified full of water by venting and/or ultrasonic testing (UT) pump casings and accessible high point vents. This will also prevent water hammer, pump cavitation, and pumping of noncondensible gas (e.g., air, nitrogen, or hydrogen) into the reactor vessel following an SI signal or during shutdown cooling.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.5.2.4

Periodic surveillance testing of ECCS pumps to detect gross degradation caused by impeller structural damage or other hydraulic component problems is required by the ASME Code. This type of testing may be accomplished by measuring the pump developed head at only one point of the pump characteristic curve. This verifies both that the measured performance is within an acceptable tolerance of the original pump baseline performance and that the performance at the test flow is greater than or equal to the performance assumed in the plant safety analysis. SRs are specified in the Inservice Testing Program of the ASME Code.

BASES

ACTIONS (continued)

In the event the isolation valve leakage results in exceeding the overall containment leakage rate, Note 4 directs entry into the applicable Conditions and Required Actions of LCO 3.6.1.

Note 5 limits the number of open containment purge lines to no more than one set of supply valves and one set of exhaust valves.

A.1 and A.2

Condition A is applicable to penetration flow paths with two containment isolation valves, and penetration flow paths with only one containment isolation valve and a closed system. The closed system must meet the requirements of Reference 3.

In the event one containment isolation valve in one or more penetration flow paths is inoperable, except for inoperable containment vacuum relief isolation valve(s), or shield building bypass or containment purge isolation valve leakage not within limit, 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 manual valve, a blind flange, and a check valve with flow through the valve secured. For a penetration flow path 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 Completion Time specified for each Category of containment isolation valve identified in Table B 3.6.3-1, Containment Isolation Valve Completion Times. The Completion Time is justified in Reference 4.

For affected penetration flow paths that cannot be restored to OPERABLE status within the specified 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 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 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

BASES

ACTIONS (continued)

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.

Required Action A.2 is modified by two Notes. Note 1 applies to isolation devices 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. Note 2 applies to isolation devices that are locked, sealed, or otherwise secured in position and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since the function of locking, sealing, or securing components is to ensure that these devices are not inadvertently repositioned. 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, except for inoperable containment vacuum relief isolation valve(s), or shield building bypass or containment purge isolation valve leakage not within limit, 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 automatic valve, a closed manual valve, and a blind flange. 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 control 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.

BASES

ACTIONS (continued)

C.1

In the event one containment isolation valve in two or more penetration flow paths is inoperable except for inoperable containment vacuum relief isolation valve(s), or shield building bypass or containment purge isolation valve leakage not within limit, all but one of the affected penetration flow path(s) 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 manual valve, a blind flange, and a check valve with flow through the valve secured. For a penetration flow path isolated in accordance with Required Action C.1, the device used to isolate the penetration should be the closest available one to containment. Required Action C.1 must be completed within 4 hours. For subsequent containment isolation valve inoperabilities, the Required Action and Completion Time continue to apply to each additional containment isolation valve inoperability, with the Completion Time based on each subsequent entry into the Condition consistent with Note 2 to the ACTIONS Table (e.g., for each entry into the Condition). Each containment isolation valve(s) that is (are) declared inoperable for subsequent Condition C entries shall meet the Required Action and Completion Time. For the penetration flow paths isolated in accordance with Required Action C.1, the affected penetration(s) 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 that the penetrations requiring isolation following an accident are isolated. The 4 hour Completion Time 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.

D.1 and D.2

In the event two or more pairs of containment purge lines are open, all but one penetration flow paths 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 manual valve, and a blind flange. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1. The 1 hour Completion Time 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

LCO (continued)

Each motor-driven auxiliary feedwater pump (one Train A and one Train B) supplies flow paths to two steam generators. Each flow path contains an automatic air-operated level control valve (LCV). The LCVs have the same train designation as the associated pump and are provided trained air. The turbine driven auxiliary feedwater pump supplies flow paths to all four steam generators. Each of these flow paths contains an automatic opening (non-modulating) air-operated LCV, two of which are designated as Train A, receive A-train air, and provide flow to the same steam generators that are supplied by the B-train motor-driven auxiliary feedwater pump. The remaining two LCVs are designated as Train B, receive B-train air, and provide flow to the same steam generators that are supplied by the A-train motor-driven pump. This design provides the required redundancy to ensure that at least two steam generators receive the necessary flow assuming any single failure. It can be seen from the description provided above that the loss of a single train of air (A or B) will not prevent the auxiliary feedwater system from performing its intended safety function and is no more severe than the loss of a single auxiliary feedwater pump. Therefore, the loss of a single train of auxiliary air only affects the capability of a single motor-driven auxiliary feedwater pump because the turbine-driven pump is still capable of providing flow to two steam generators that are separate from the other motor-driven pump.

Two redundant steam sources are required to be operable to ensure that at least one source is available for the steam-driven auxiliary feedwater (AFW) pump operation following a feedwater or main steam line break. This requirement ensures that the plant remains within its design basis (i.e., AFW to two intact steam generators) given the event of a loss of the No.1 steam generator because of a main steam line or feedwater line break and a single failure of the B-train motor driven AFW pump. The two redundant sources must be aligned such that No.1 steam generator source is open and operable and the No.4 steam generator source is closed and operable.

For instances where one train of emergency raw cooling water (ERCW) is declared inoperable in accordance with technical specifications, the AFW turbine-driven pump is considered operable since it is supplied by both trains of ERCW. Similarly, the AFW turbine-driven pump is considered operable when one train of the AFW loss of power start function is declared inoperable in accordance with Technical Specifications because both 6.9 kilovolt shutdown board logic trains supply this function. This position is consistent with American National Standards Institute/ANS 58.9 requirements (i.e., postulation of the failure of the opposite train is not required while relying on the TS limiting condition for operation).

BASES

LCO (continued)

The LCO is modified by a Note indicating that one AFW train, which includes a motor driven pump, is required to be OPERABLE in MODE 4. This is because of the reduced heat removal requirements and short period of time in MODE 4 during which the AFW is required and the insufficient steam available in MODE 4 to power the turbine driven AFW pump.

APPLICABILITY

In MODES 1, 2, and 3, the AFW System is required to be OPERABLE in the event it is called upon to function when MFW is lost. In addition, the AFW System is required to supply enough makeup water to replace the steam generator secondary inventory, lost as the unit cools to MODE 4 conditions.

In MODE 4 the AFW System may be used for heat removal via the steam generators.

In MODE 5 or 6, the steam generators are not normally used for heat removal, and the AFW System is not required.

ACTIONS

A Note prohibits the application of LCO 3.0.4.b to an inoperable AFW train. There is an increased risk associated with entering a MODE or other specified condition in the Applicability with an AFW train inoperable and the provisions of LCO 3.0.4.b, which allow entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, should not be applied in this circumstance.

A.1

If the turbine driven AFW train is inoperable due to one inoperable steam supply, or if a turbine driven pump is inoperable for any reason while in MODE 3 immediately following refueling, action must be taken to restore the inoperable equipment to an OPERABLE status within 7 days. The 7 day Completion Time is reasonable, based on the following reasons:

a. For the inoperability of the turbine driven AFW pump due to one inoperable steam supply, the 7 day Completion Time is reasonable since there is a redundant steam supply line for the turbine driven pump and the turbine driven train is still capable of performing its specified function for most postulated events.

b. For the inoperability of a turbine driven AFW pump while in MODE 3 immediately subsequent to a refueling, the 7 day Completion Time is reasonable due to the minimal decay heat levels in this situation.

BASES

ACTIONS (continued)

c. For both the inoperability of the turbine driven pump due to one inoperable steam supply and an inoperable turbine driven AFW pump while in MODE 3 immediately following a refueling outage, the 7 day Completion Time is reasonable due to the availability of redundant OPERABLE motor driven AFW pumps, and due to the low probability of an event requiring the use of the turbine driven AFW pump.

Condition A is modified by a Note which limits the applicability of the Condition for an inoperable turbine driven AFW pump in MODE 3 to when the unit has not entered MODE 2 following refueling. Condition A allows one AFW train to be inoperable for 7 days vice the 72 hour Completion Time in Condition B. This longer Completion Time is based on the reduced decay heat following refueling and prior to the reactor being critical.

B.1

With one of the required AFW trains (pump or flow path) inoperable in MODE 1, 2, or 3 for reasons other than Condition A, action must be taken to restore OPERABLE status within 72 hours. This Condition includes the loss of two steam supply lines to the turbine driven AFW pump. The 72 hour Completion Time is reasonable, based on the redundant capabilities afforded by the AFW System, the time needed for repairs, and the low probability of a DBA occurring during this time period.

C.1 and C.2

With one of the required motor driven AFW trains (pump or flow path) inoperable and the turbine driven AFW train inoperable due to one inoperable steam supply, action must be taken to restore the affected equipment to OPERABLE status within 48 hours. Assuming no single active failures when in this condition, the accident (a feedline break (FLB) or main steam line break (MSLB)) could result in the loss of the remaining steam supply to the turbine driven AFW pump due to the faulted steam generator (SG).

BASES

ACTIONS (continued)

The 48 hour Completion Time is reasonable based on the fact that the remaining motor driven AFW train is capable of providing 100% of the AFW flow requirements, and the low probability of an event occurring that would challenge the AFW system.

D.1 and D.2

When Required Action A.1, B.1, C.1, or C.2 cannot be completed within the required Completion Time, or if two AFW trains are inoperable in MODE 1, 2, or 3 for reasons other than Condition C, 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 6 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.

In MODE 4 with two AFW trains inoperable, operation is allowed to continue because only one motor driven pump AFW train is required in accordance with the Note that modifies the LCO. Although not required, the unit may continue to cool down and initiate RHR.

E.1

If all three AFW trains are inoperable in MODE 1, 2, or 3, the unit is in a seriously degraded condition with no safety related means for conducting a cooldown, and only limited means for conducting a cooldown with nonsafety related equipment. 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 one AFW train to OPERABLE status.

Required Action E.1 is modified by a Note indicating that all required MODE changes are suspended until one AFW train is restored to OPERABLE status. In this case, LCO 3.0.3 is not applicable because it could force the unit into a less safe condition.

BASES

ACTIONS (continued)

F.1

In MODE 4, either the reactor coolant pumps or the RHR loops can be used to provide forced circulation. This is addressed in LCO 3.4.6, "RCS Loops - MODE 4." With the required AFW train inoperable, action must be taken to immediately restore the inoperable train to OPERABLE status. The immediate Completion Time is consistent with LCO 3.4.6.

SURVEILLANCE REQUIREMENTS

SR 3.7.5.1

Verifying the correct alignment for manual, power operated, and automatic valves in the AFW System water and steam supply flow paths provides assurance that the proper flow paths will exist for AFW 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 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 being mispositioned are in the correct position.

The SR is modified by a Note that states one or more AFW trains may be considered OPERABLE during alignment and operation for steam generator level control, if it is capable of being manually (i.e., remotely or locally, as appropriate) realigned to the AFW mode of operation, provided it is not otherwise inoperable. This exception allows the system to be out of its normal standby alignment and temporarily incapable of automatic initiation without declaring the train(s) inoperable. Since AFW may be used during startup, shutdown, hot standby operations, and hot shutdown operations for steam generator level control, and these manual operations are an accepted function of the AFW System, OPERABILITY (i.e., the intended safety function) continues to be maintained.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.7.5.2

Verifying that each AFW pump's developed head at the flow test point is greater than or equal to the required developed head ensures that AFW pump performance has not degraded during the cycle. Flow and differential head are normal tests of centrifugal pump performance required by the ASME Code (Ref 2). Because it is undesirable to introduce cold AFW into the steam generators while they are operating,

BASES

SURVEILLANCE REQUIREMENTS (continued)

this testing is performed 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 detect incipient failures by indicating abnormal performance. Performance of inservice testing discussed in the ASME Code (Ref. 2) (only required at 3 month intervals) satisfies this requirement.

This SR is modified by a Note indicating that the SR should be deferred until suitable test conditions are established. This deferral is required because there is insufficient steam pressure to perform the test.

SR 3.7.5.3

This SR verifies that AFW can be delivered to the appropriate steam generator in the event of any accident or transient that generates an ESFAS, or in the event the CSTs become depleted, by demonstrating that each automatic valve in the flow path actuates to its correct position on an actual or simulated actuation signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

The SR is modified by two Notes. Note 1 states that one or more AFW trains may be considered OPERABLE during alignment and operation for steam generator level control, if it is capable of being manually (i.e., remotely or locally, as appropriate) realigned to the AFW mode of operation, provided it is not otherwise inoperable. This exception allows the system to be out of its normal standby alignment and temporarily incapable of automatic initiation without declaring the train(s) inoperable. Since AFW may be used during startup, shutdown, hot standby operations, and hot shutdown operations for steam generator level control, and these manual operations are an accepted function of the AFW System, OPERABILITY (i.e., the intended safety function) continues to be maintained. Note 2 states that the SR is only required to be met in MODES 1, 2, and 3. It is not required to be met in MODE 4, since the AFW train is only required for the purposes of removing decay heat when the SG is relied upon for heat removal. The operation of the AFW train is by manual means and automatic startup of the AFW train is not required.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.5.4

This SR verifies that the AFW pumps will start in the event of any accident or transient that generates an ESFAS by demonstrating that each AFW pump starts automatically on an actual or simulated actuation signal in MODES 1, 2, and 3. In MODE 4, the required pump is already operating and the autostart function is not required.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

This SR is modified by three Notes. Note 1 indicates that the SR may be deferred until suitable test conditions are established. This deferral is required because there is insufficient steam pressure to perform the test. Note 2 states that one or more AFW trains may be considered OPERABLE during alignment and operation for steam generator level control, if it is capable of being manually (i.e., remotely or locally, as appropriate) realigned to the AFW mode of operation, provided it is not otherwise inoperable. This exception allows the system to be out of its normal standby alignment and temporarily incapable of automatic initiation without declaring the train(s) inoperable. Since AFW may be used during startup, shutdown, hot standby operations, and hot shutdown operations for steam generator level control, and these manual operations are an accepted function of the AFW System. OPERABILITY (i.e., the intended safety function) continues to be maintained. Note 3 states that the SR is only required to be met in MODES 1, 2, and 3. It is not required to be met in MODE 4, since the AFW train is only required for the purposes of removing decay heat when the SG is relied upon for heat removal. The operation of the AFW train is by manual means and automatic startup of the AFW train is not required.

REFERENCES

1. UFSAR, Section 10.4.7.2.

2. ASME Code for Operation and Maintenance of Nuclear Power Plants.

BASES

APPLICABLE SAFETY ANALYSES (continued)

(Ref. 3) entry conditions to MODE 5 during normal and post accident operations. The time required for this evolution is a function of the number of component cooling water and RHR System trains that are operating. One ERCW system train is sufficient to remove decay heat during subsequent operations in MODES 5 and 6. This assumes a maximum ERCW system temperature of 87°F occurring simultaneously with maximum heat loads on the system.

The ERCW system satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Two ERCW system trains are required to be OPERABLE to provide the required redundancy to ensure that the system functions to remove post accident heat loads, assuming that the worst case single active failure occurs coincident with the loss of offsite power.

An ERCW system train is considered OPERABLE during MODES 1, 2, 3, and 4 when the required ERCW pumps are operable and the associated piping, valves, heat exchanger, and instrumentation and controls required to perform the safety related function as described in UFSAR Section 9.2.2.

APPLICABILITY

In MODES 1, 2, 3, and 4, the ERCW system is a normally operating system that is required to support the OPERABILITY of the equipment serviced by the ERCW system and required to be OPERABLE in these MODES.

In MODES 5 and 6, the OPERABILITY requirements of the ERCW system are determined by the systems it supports.

ACTIONS

A.1 and A.2

Condition A is modified by two Notes that limit the conditions and parameters that allow entry into Condition A. The first Note limits the applicability of Condition A to the time period when the opposite unit is either defueled or in MODE 6 following defueled with refueling water cavity level ≥ 23 ft. above the top of the reactor vessel flange. The second Note requires a temperature limitation on UHS Temperature. In

BASES

ACTIONS (continued)

order to credit the temperature limit, the effected ERCW train must be aligned in accordance with UFSAR 9.2.2.2. This will allow the plant configuration to be aligned (i.e., cross-ties exist and isolation of loads to facilitate maintenance activities) to minimize the heat load on the ERCW system to ensure the ERCW system continues to meet its design function.

The 7 day Completion Time is acceptable based on the following:

- The low probability of a DBA occurring during that time;
- The heat load on the ERCW System is substantially lower than assumed for the DBA with the opposite unit defueled or subsequent to defueled; and
- The redundant capabilities afforded by the OPERABLE train.

If one ERCW system train is inoperable for planned maintenance, action must be taken to restore OPERABLE status within 7 days. In this Condition, the remaining OPERABLE ERCW system train is adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure in the OPERABLE ERCW system train could result in loss of ERCW system function.

Required Action A.1 is modified by two Notes. The first Note indicates that the applicable Conditions and Required Actions of LCO 3.8.1, “AC Sources – Operating,” should be entered if an inoperable ERCW system train results in an inoperable emergency diesel generator. The second Note indicates that the applicable Conditions and Required Actions of LCO 3.4.6, “RCS Loops – MODE 4,” should be entered if an inoperable ERCW system train results in an inoperable residual heat removal loop. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components.

Required Action A.2 ensures the credited temperature limit for Ultimate Heat Sink is maintained.

B.1

If one ERCW system train is inoperable for reasons other than Condition A, action must be taken to restore OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE ERCW system train is adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure in the OPERABLE ERCW system train could result in loss of ERCW system function. Required Action B.1 is modified by two Notes.

BASES

ACTIONS (continued)

The first Note indicates that the applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources - Operating," should be entered if an inoperable ERCW system train results in an inoperable emergency diesel generator. The second Note indicates that the applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops - MODE 4," should be entered if an inoperable ERCW system train results in an inoperable residual heat removal loop. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components. The 72 hour Completion Time is based on the redundant capabilities afforded by the OPERABLE train, and the low probability of a DBA occurring during this time period.

C.1 and C.2

If the ERCW system train 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 6 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 3.7.8.1

This SR is modified by a Note indicating that the isolation of the ERCW system components or systems may render those components inoperable, but does not affect the OPERABILITY of the ERCW system.

Verifying the correct alignment for manual, power operated, and automatic valves in the ERCW system flow path provides assurance that the proper flow paths exist for ERCW 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 being locked, sealed, or secured. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.8.2

This SR verifies proper automatic operation of the ERCW system valves on an actual or simulated actuation signal. The Safety Injection signal is the automatic actuation signal. The ERCW system is a normally operating system that cannot be fully actuated as part of normal testing. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.7.8.3

This SR verifies proper automatic operation of the ERCW system pumps on an actual or simulated (i.e., Safety Injection) actuation signal. The ERCW system is a normally operating system that cannot be fully actuated as part of normal testing during normal operation.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. UFSAR, Section 9.2.2.
2. UFSAR, Section 6.2.
3. UFSAR, Section 5.5.7.