(continued)

LIST OF TABLES

LIST OF FIGURES

BASES

BACKGROUND (continued)

is the total reactivity change divided by the total temperature change. The test is repeated by reversing the direction of the temperature change, and the final ITC is the average of the two calculated ITCs. The Moderator Temperature Coefficient (MTC) at beginning-of-life (BOL) is determined from the measured ITC. This test satisfies the requirement of SR 3.1.3.1. Performance of this test could violate LCO 3.4.2, "RCS Minimum Temperature for Criticality."

c. The Control Rod Worth Test is used to measure the reactivity worth of selected control banks. This test is performed at HZP and has four alternative methods of performance. The first method, the Boron Exchange Method, varies the reactor coolant boron concentration and moves the selected control bank in response to the changing boron concentration. The reactivity changes are measured with a reactivity computer. This sequence is repeated for the remaining control banks. The second method, the Rod Swap Method, measures the worth of a predetermined reference bank using the Boron Exchange Method above. The reference bank is then nearly fully inserted into the core. The selected bank is then inserted into the core as the reference bank is withdrawn. The HZP critical conditions are then determined with the selected bank fully inserted into the core. The worth of the selected bank is inferred, based on the position of the reference bank with respect to the selected bank. This sequence is repeated as necessary for the remaining control banks. The third method, the Boron Endpoint Method, moves the selected control bank over its entire length of travel and then varies the reactor coolant boron concentration to achieve HZP criticality again. The difference in boron concentration is the worth of the selected control bank. This sequence is repeated for the remaining control banks. The fourth method is by dynamic insertion (Dynamic Rod Worth Measurement (DRWM)). In this method, the core is critical in a near all rods out condition. All rods are withdrawn, and each bank is then inserted into and withdrawn from the core, measuring the worth dynamically. Performance of this test could violate LCO 3.1.4, LCO 3.1.5, or LCO 3.1.6.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The fuel is protected by LCOs that preserve the initial conditions of the core assumed during the safety analyses. The methods for development of the LCOs that are excepted by this LCO are described in the Westinghouse Reload Safety Evaluation Methodology Report (Ref. 4). The above mentioned PHYSICS TESTS, and other tests that may be required to calibrate nuclear instrumentation or to diagnose operational problems, may require the operating control or process variables to deviate from their LCO limitations.

The FSAR defines requirements for initial testing of the facility, including PHYSICS TESTS. Tables 14.2.1-1 and 14.2.1-2 summarize the zero, low power, and power tests. Reload fuel cycle PHYSICS TESTS are performed in accordance with Technical Specification requirements, fuel vendor guidelines, and established industry practices. Although these PHYSICS TESTS are generally accomplished within the limits for all LCOs, conditions may occur when one or more LCOs must be suspended to make completion of PHYSICS TESTS possible or practical. This is acceptable as long as the fuel design criteria are not violated. When one or more of the requirements specified in LCO 3.1.3, LCO 3.1.4, LCO 3.1.5, LCO 3.1.6, and LCO 3.4.2 are suspended for PHYSICS TESTS, the fuel design criteria are preserved as long as the power level is limited to ≤ 5% RTP, the reactor coolant temperature is kept ≥ 541°F, and SDM is ≥ the limit specified in the COLR.

The PHYSICS TESTS include measurement of core nuclear parameters or the exercise of control components that affect process variables. Among the process variables involved are AFD and QPTR, which represent initial conditions of the unit safety analyses. Also involved are the movable control components (control and shutdown rods), which are required to shut down the reactor. The limits for these variables are specified for each fuel cycle in the COLR. PHYSICS TESTS meet the criteria for inclusion in the Technical Specifications, since the components and process variable LCOs suspended during PHYSICS TESTS meet Criteria 1, 2, and 3 of the NRC Policy Statement.

Reference 5 allows special test exceptions (STEs) to be included as part of the LCO that they affect. It was decided, however, to retain this STE as a separate LCO because it was less cumbersome and provided additional clarity.

(continued)

BASES

SURVEILLANCE REQUIREMENTS	SR 3.1.8.3 (continued) The Frequency of 24 hours is based on the generally slow change in required boron concentration and on the low probability of an accident occurring without the required SDM.
REFERENCES	1. 10 CFR 50, Appendix B, Section XI. 2. 10 CFR 50.59. 3. Regulatory Guide 1.68, Revision 2, August 1978. 4. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology Report," July 1985. 5. WCAP-11618, including Addendum 1, April 1989. 6. WCAP-13360-P-A, "Westinghouse Dynamic Rod Worth Measurement Technique," January 1996.

BASES

ACTIONS

A.1 (continued)

the applicable safety analyses. A Completion Time of 30 minutes is reasonable, based on operating experience, to reach 50% RTP without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.3.1

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

This Surveillance verifies that the AFD, as indicated by the NIS excore channel, is within its specified limits and is consistent with the status of the AFD monitor alarm. With the AFD monitor alarm inoperable, the AFD is monitored every hour to detect operation outside its limit. The Frequency of 1 hour is based on operating experience regarding the amount of time required to vary the AFD, and the fact that the AFD is closely monitored. With the AFD monitor alarm OPERABLE, the Surveillance Frequency of 7 days is adequate considering that the AFD is monitored by a computer and any deviation from requirements is alarmed.

REFERENCES

1. WCAP-8403 (nonproprietary), "Power Distribution Control and Load Following Procedures," Westinghouse Electric Corporation, September 1974.

2. R. W. Miller et al., "Relaxation of Constant Axial Offset Control: F_Q Surveillance Technical Specification," WCAP-10216(NP), June 1983.

BASES

BACKGROUND

Signal Process Control and Protection System (continued)

the other channels providing the protection function actuation. Again, a single failure will neither cause nor prevent the protection function actuation. These requirements are described in IEEE-279-1971 (Ref. 4). The actual number of channels required for each unit parameter is specified in Reference 1.

Two logic channels are required to ensure no single random failure of a logic channel will disable the RTS. The logic channels are designed such that testing required while the reactor is at power may be accomplished without causing trip. Provisions to allow removing logic channels from service during maintenance are unnecessary because of the logic system's designed reliability.

Trip Setpoints and Allowable Values

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

The Trip Setpoints used in the bistables are based on the analytical limits stated in Reference 1. The selection of these Trip Setpoints is such that adequate protection is provided when all sensor and processing time delays are taken into account. To allow for calibration tolerances, instrumentation uncertainties, instrument drift, and severe environment errors for those RTS channels that must function in harsh environments as defined by 10 CFR 50.49 (Ref. 5), the Trip Setpoints and Allowable Values specified in Table 3.3.1-1 in the accompanying LCO are conservatively adjusted with respect to the analytical limits. A detailed description of the methodology used to calculate the Trip Setpoints, including their explicit uncertainties, is provided in the "RTS/ESFAS Setpoint Methodology Study" (Ref. 6). The actual nominal Trip Setpoint entered into the bistable is more conservative than that specified by the Allowable Value to account for changes in random measurement errors detectable by a COT. One example of such a change in measurement error is drift during the surveillance interval.

(continued)

BASES

BACKGROUND

Trip Setpoints and Allowable Values (continued)

If the measured setpoint does not exceed the Allowable Value, the bistable is considered OPERABLE.

Setpoints in accordance with the Allowable Value ensure that SLs are not violated during AOOs (and that the consequences of DBAs will be acceptable, providing the unit is operated from within the LCOs at the onset of the AOO or DBA and the equipment functions as designed). For the purpose of demonstrating compliance with 10 CFR 50.36 to the extent that the Technical Specifications are required to specify Limiting Safety System Settings (LSSS), the LSSS for VEGP are comprised of both the Nominal Trip Setpoints and the Allowable Values specified in Table 3.3.1-1. The Nominal Trip Setpoint is the expected value to be achieved during calibrations. The Nominal Trip Setpoint considers all factors which may affect channel performance by statistically combining rack drift, rack measurement and test equipment effects, rack calibration accuracy, rack comparator setting accuracy, rack temperature effects, sensor measurement and test equipment effects, sensor calibration accuracy, primary element accuracy, and process measurement accuracy. The Nominal Trip Setpoint is the value that will always ensure that safety analysis limits are met (with margin) given all of the above effects. The Allowable Value has been established by considering the values assumed for rack effects only. The Allowable Value serves as an operability limit for the purpose of the quarterly CHANNEL OPERATIONAL TESTS.

Each channel of the process control equipment can be tested on line to verify that the signal or setpoint accuracy is within the specified allowance requirements of Reference 2. Once a designated channel is taken out of service for testing, a simulated signal is injected in place of the field instrument signal. The process equipment for the channel in test is then tested, verified, and calibrated. SRs for the channels are specified in the SRs section.

The Trip Setpoints and Allowable Values listed in Table 3.3.1-1 are based on the methodology described in Reference 6, which incorporates all of the known uncertainties applicable for each channel. The magnitudes of these uncertainties are factored into the determination of each Trip Setpoint. All field sensors and signal

(continued)

BASES

BACKGROUND

Trip Setpoints and Allowable Values (continued)

processing equipment for these channels are assumed to operate within the allowances of these uncertainty magnitudes.

Solid State Protection System

The SSPS equipment is used for the decision logic processing of outputs from the signal processing equipment bistables. To meet the redundancy requirements, two trains of SSPS, each performing the same functions, are provided. If one train is taken out of service for maintenance or test purposes, the second train will provide reactor trip and/or ESF actuation for the unit. If both trains are taken out of service or placed in test, a reactor trip will result. Each train is packaged in its own cabinet for physical and electrical separation to satisfy separation and independence

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BASES

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(continued)

BASES

BACKGROUND

Reactor Trip Switchgear (continued)

trip mechanism is sufficient by itself, thus providing a diverse trip mechanism.

The decision logic matrix Functions are described in the functional diagrams included in Reference 1. In addition to the reactor trip or ESF, these diagrams also describe the various "permissive interlocks" that are associated with unit conditions. Each train has a built in testing device that can automatically test the decision logic matrix Functions and the actuation devices while the unit is at power. When any one train is taken out of service for testing, the other train is capable of providing unit monitoring and protection until the testing has been completed. The testing device is semiautomatic to minimize testing time.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

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

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

The LCO requires all instrumentation performing an RTS Function, listed in Table 3.3.1-1 in the accompanying LCO, to be OPERABLE. Failure of any instrument renders the affected channel(s) inoperable and reduces the reliability of the affected Functions. The Nominal Trip Setpoint column is modified by a Note that requires the as-left condition for a channel to be within the calibration tolerance for that channel. In addition, the as-left condition may be more conservative than the specified Nominal Trip Setpoint.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, AND APPLICABILITY (continued)

The conservative direction is established by the direction of the inequality applied to the Allowable Value. It is consistent with the setpoint methodology for the as-left trip setpoint to be outside the calibration tolerance but in the conservative direction with respect to the Nominal Trip Setpoint. For example, the Power Range Neutron Flux High trip setpoint may be set to a value less than 109% during initial startup following a refueling outage until a sufficiently high reactor power is achieved so that the power range channels may be calibrated. In addition, certain Required Actions may require that the Power Range Neutron Flux High trip setpoints and/or the Overpower Delta-T setpoints be reduced based on plant conditions.

The LCO generally requires OPERABILITY of four or three channels in each instrumentation Function, two channels of Manual Reactor Trip in each logic Function, and two trains in each Automatic Trip Logic Function. Four OPERABLE

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BASES

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

6. Overtemperature ΔT (continued)

This results in a two-out-of-four trip logic. Section 7.2.2.3 of Reference 1 discusses control and protection system interactions for this function. Note that this Function also provides a signal to generate a turbine runback prior to reaching the Trip Setpoint. A turbine runback will reduce turbine power and reactor power. A reduction in power will normally alleviate the Overtemperature ΔT condition and may prevent a reactor trip.

Delta-T_0, as used in the overtemperature and overpower ΔT trips, represents the 100% RTP value as measured for each loop. This normalizes each loop's ΔT trips to the actual operating conditions existing at the time of measurement, thus forcing the trip to reflect the equivalent full power conditions as assumed in the accident analyses. These differences in RCS loop ΔT can be due to several factors, e.g., differences in RCS loop flows and slightly asymmetric power distributions between quadrants. While RCS loop flows are not expected to change with cycle life, radial power redistribution between quadrants may occur, resulting in small changes in loop specific ΔT values. Therefore, loop specific ΔT_0 values are measured as needed to ensure they represent actual core conditions.

The parameter K_1 is the principal setpoint gain, since it defines the function offset. The parameters K_2 and K_3 define the temperature gain and pressure gain, respectively. The values for T' and P'are key reference parameters corresponding directly to plant safety analyses initial conditions assumptions for the Overtemperature ΔT function. For the purposes of performing a CHANNEL CALIBRATION, the values for K_1, K_2, K_3, T', and P' are utilized in the safety analyses without explicit tolerances, but should be considered as nominal values for instrument settings. That is, while an exact setting is not expected, a setting as close as reasonably possible is desired. Note that for T', the value for the hottest RCS loop will be set

(continued)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

6. Overtemperature ΔT (continued)

as close as possible to 588.4°F. The value of T' for the remaining RCS loops will be set appropriately less than 588.4°F based on the actual loop specific indicated T_{avg}. The engineering scaling calculations use each of the referenced parameters as an exact gain or reference value. Tolerances are not applied to the individual gain or reference parameters. Tolerances are applied to each calibration module and the overall string calibration. In order to ensure that the Overtemperature ΔT setpoint is consistent with the assumptions of the safety analyses, it is necessary to verify during the CHANNEL OPERATIONAL TEST that the Overtemperature ΔT setpoint is within the appropriate calibration tolerances for the defined calibration conditions (Ref. 9).

The LCO requires all four channels of the Overtemperature ΔT trip Function to be OPERABLE. Note that the Overtemperature ΔT Function receives input from channels shared with other RTS Functions. Failures that affect multiple Functions require entry into the Conditions applicable to all affected Functions.

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

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BASES

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

7. Overpower ΔT (continued)

Delta-T_0, as used in the overtemperature and overpower ΔT trips, represents the 100% RTP value as measured for each loop. This normalizes each loop's ΔT trips to the actual operating conditions existing at the time of measurement, thus forcing the trip to reflect the equivalent full power conditions as assumed in the accident analyses. These differences in RCS loop ΔT can be due to several factors, e.g., difference in RCS loop flows and slightly asymmetric power distributions between quadrants. While RCS loop flows are not expected to change with cycle life, radial power redistribution between quadrants may occur, resulting in small changes in loop specific ΔT values. Therefore, loop specific ΔT_0 values are measured as needed to ensure they represent actual core conditions.

The value for T" is a key reference parameter corresponding directly to plant safety analyses initial conditions assumptions for the Overpower ΔT function. For the purposes of performing a CHANNEL CALIBRATION, the values for K_4, K_5, K_6, and T" are utilized in the safety analyses without explicit tolerances, but should be considered as nominal values for instrument settings. That is, while an exact setting is not expected, a setting as close as reasonably possible is desired. Note that for T", the value for the hottest RCS loop will be set as close as possible to 588.4°F. The value of T" for the remaining RCS loops will be set appropriately less than 588.4°F based on the actual loop specific indicated T_{avg}. The engineering scaling calculations use each of the referenced parameters as an exact gain or reference value. Tolerances are not applied to the individual gain or reference parameters. Tolerances are applied to each calibration module and the overall string calibration. In order to ensure that the Overpower ΔT setpoint is consistent with the assumptions of the safety analyses, it is necessary to verify during the CHANNEL OPERATIONAL TEST that the Overpower ΔT setpoint is within the appropriate calibration tolerances for defined calibration conditions (Ref. 9). Note that for the parameter K_5,

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

7. Overpower ΔT (continued)

in the case of decreasing temperature, the gain setting must be ≥ 0 to prevent generating setpoint margin on decreasing temperature rates. Similarly, the setting for K_5 is required to be equal to 0 for conditions where T ≤ T".

The LCO requires four channels of the Overpower ΔT trip Function to be OPERABLE. Note that the Overpower ΔT trip Function receives input from channels shared with other RTS Functions. Failures that affect multiple Functions require entry into the Conditions applicable to all affected Functions.

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

8. Pressurizer Pressure

The same sensors (PI-0455A, B, & C, PI-0456, PI-0456A, PI-0457, PI-0457A, PI-0458, PI-0458A) provide input to the Pressurizer Pressure -- High and -- Low trips and the Overtemperature ΔT trip. Since the Pressurizer Pressure channels are also used to provide input to the Pressurizer Pressure Control System, the actuation logic must be able to withstand an input failure to

(continued)

BASES

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BASES

REFERENCES (continued)

2. FSAR, Chapter 6.

3. FSAR, Chapter 15.

4. IEEE-279-1971.

5. 10 CFR 50.49.

6. WCAP-11269, Westinghouse Setpoint Methodology for Protection Systems; as supplemented by:

   - Amendments 34 (Unit 1) and 14 (Unit 2), RTS Steam Generator Water Level — Low Low, ESFAS Turbine Trip and Feedwater Isolation SG Water Level — High High, and ESFAS AFW SG Water Level — Low Low.

   - Amendments 48 and 49 (Unit 1) and Amendments 27 and 28 (Unit 2), deletion of RTS Power Range Neutron Flux High Negative Rate Trip.

   - Amendments 60 (Unit 1) and 39 (Unit 2), RTS Overtemperature ΔT setpoint revision.

   - Amendments 57 (Unit 1) and 36 (Unit 2), RTS Overtemperature and Overpower ΔT time constants and Overtemperature ΔT setpoint.

   - Amendments 43 and 44 (Unit 1) and 23 and 24 (Unit 2), revised Overtemperature and Overpower ΔT trip setpoints and allowable values.

7. WCAP-10271-P-A, Supplement 2, Rev. 1, June 1990.

8. FSAR, Chapter 16.

9. Westinghouse Letter GP-16696, November 5, 1997.

BASES

BACKGROUND

Sequencer Output Relays (continued)

sequencer and are part of the control circuitry of these ESF loads. There are two independent trains of sequencers and each is powered by the respective train of 120-Vac ESF electrical power supply. The power supply for the output relays is the sequencer power supply. The applicable output relays are tested in the slave relay testing procedures, and in particular, in conjunction with the specific slave relay also required to actuate to energize the applicable ESF load.

APPLICABLE SAFETY ANALYSES, LCO, AND APPLICABILITY

Each of the analyzed accidents can be detected by one or more ESFAS Functions. One of the ESFAS Functions is the primary actuation signal for that accident. An ESFAS Function may be the primary actuation signal for more than one type of accident. An ESFAS Function may also be a secondary, or backup, actuation signal for one or more other accidents. For example, Pressurizer Pressure — Low is a primary actuation signal for small loss of coolant accidents (LOCAs) and a backup actuation signal for steam line breaks (SLBs) outside containment. Functions such as manual initiation, not specifically credited in the accident safety analysis, are qualitatively credited in the safety analysis and the NRC staff approved licensing basis for the unit. These Functions may provide protection for conditions that do not require dynamic transient analysis to demonstrate Function performance. These Functions may also serve as backups to Functions that were credited in the accident analysis (Ref. 3).

The LCO requires all instrumentation performing an ESFAS Function to be OPERABLE. Failure of any instrument renders the affected channel(s) inoperable and reduces the reliability of the affected Functions. The Nominal Trip Setpoint column is modified by a Note that requires the as-left conditions for a channel to be within the calibration tolerance for that channel. In addition, the as-left condition may be more conservative than the specified Nominal Trip Setpoint. The conservative direction is established by the direction of the inequality applied to the Allowable Value. It is consistent with the setpoint methodology for the as-left trip setpoint to be outside the calibration tolerance but in the conservative direction with respect to the Nominal Trip Setpoint.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, AND APPLICABILITY (continued)	The LCO generally requires OPERABILITY of four or three channels in each instrumentation function and two channels in each logic and manual initiation function. The two-out-of-three and the two-out-of-four configurations allow one channel to be tripped during maintenance or testing without causing an ESFAS initiation. If an instrument channel is equipped with installed bypass capability, such that no jumpers or lifted leads are

(continued)

BASES

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(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.2.8 (continued)

verification of these devices every 18 months. The 18 month Frequency is consistent with the typical refueling cycle and is based on unit operating experience, which shows that random failures of instrumentation components causing serious response time degradation, but not channel failure, are infrequent occurrences.

This SR is modified by a Note that clarifies that the turbine driven AFW pump is tested within 24 hours after reaching 900 psig in the SGs.

SR 3.3.2.9

SR 3.3.2.9 is the performance of a TADOT as described in SR 3.3.2.6 for the P-4 Reactor Trip Interlock, and the Frequency is once per 18 months. This Frequency is based on operating experience. The SR is modified by a note that excludes verification of setpoints during the TADOT. The function tested has no associated setpoint.

REFERENCES

1. FSAR, Chapter 6.
2. FSAR, Chapter 7.
3. FSAR, Chapter 15.
4. IEEE-279-1971.
5. 10 CFR 50.49.
6. WCAP-11269, Westinghouse Setpoint Methodology for Protection Systems; as supplemented by:
   - Amendments 38 (Unit 1) and 18 (Unit 2), ESFAS Safety Injection Pressurizer — Low allowable value revision.
   - Amendments 34 (Unit 1) and 14 (Unit 2), RTS Steam Generator Water Level — Low Low, ESFAS Turbine Trip and Feedwater Isolation SG Water Level — High High, and ESFAS AFW SG Water Level — Low Low.

(continued)

BASES

REFERENCES (continued)

- Amendments 43 and 44 (Unit 1) and 23 and 24 (Unit 2), revised ESFAS Interlocks Pressurizer P-11 trip setpoint and allowable value.

7. WCAP-10271-P-A, Supplement 2, Rev. 1, June 1990.

8. FSAR, Chapter 16.

9. Westinghouse Letter GP-16696, November 5, 1997.

BASES

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BASES (continued)

TABLE B 3.3.4-1

REMOTE SHUTDOWN SYSTEM MONITORING INSTRUMENTATION

INSTRUMENT FUNCTION	READOUT[1] LOCATION	CHANNELS AVAILABLE
1. Source Range Neutron Flux	A	1 (NI-31E)
2. Extended Range Neutron Flux	B	1 (NI-13135 C&D)
3. RCS Cold Leg Temperature	A, B	1/Loop (Loop 1 TI-0413D, Panel A) (Loop 2 TI-0423D, Panel B) (Loop 3 TI-0433D, Panel B) (Loop 4 TI-0443D, Panel A)
4. RCS Hot Leg Temperature	A	2 (Loop 1 TI-0413C Loop 4 TI-0443C)
5. Core Exit Thermocouples	B	2 (Loop 2 Core Quadrant 1TI-10055) (Loop 3 Core Quadrant 1TI-10056) (Loop 1 Core Quadrant 2TI-10055) (Loop 4 Core Quadrant 2TI-10056)
6. RCS Wide Range Pressure	A, B	2 (PI-405A, Panel A) (PI-403A, Panel B)
7. Steam Generator Level Wide Range	A, B	1/Loop (Loop 1 LI-501B, Panel A) (Loop 2 LI-502B, Panel B) (Loop 3 LI-503B, Panel B) (Loop 4 LI-504B, Panel A)
8. Pressurizer Level	A, B	2 (LI-459C, Panel A) (LI-460C, Panel B)
9. RWST Level	L	1 (LI-0990C)
10. BAST Level	L	1 (PI-10115)[2]
11. CST Level	L	2 (Tank 1 LI-5100) (Tank 2 LI-5115)

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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.10 Pressurizer Safety Valves

BASES

BACKGROUND

The pressurizer safety valves provide, in conjunction with the Reactor Protection System, overpressure protection for the RCS. The pressurizer safety valves are of the pop type. The valves are spring loaded and self actuated by direct fluid pressure with backpressure compensation. The safety valves are designed to prevent the system pressure from exceeding the system Safety Limit (SL), 2735 psig, which is 110% of the design pressure.

Because the safety valves are self actuating, they are considered independent components. The relief capacity for each valve, 420,000 lb/hr at a pressurizer pressure of 2560 psig, is based on postulated overpressure transient conditions resulting from a complete loss of steam flow to the turbine with the reactor operating at 102 percent of engineered safeguards design power. The relief rate is stated at a pressure of 2560 psig which is equivalent to the former set pressure of 2485 psig plus 3% for set pressure tolerance and valve accumulation. This event results in the maximum surge rate into the pressurizer, which specifies the minimum relief capacity for the safety valves. The decrease in set pressure to 2460 psig and increase in tolerance does not significantly affect the relief capacity of the safety valves.

The discharge flow from the pressurizer safety valves is directed to the pressurizer relief tank. This discharge flow is indicated by an increase in temperature downstream of the pressurizer safety valves or increase in the pressurizer relief tank temperature or level.

Overpressure protection is required in MODES 1, 2, 3, 4, 5, and MODE 6 with the reactor vessel head on; however, in MODES 4, 5, and 6 with the reactor vessel head on, overpressure protection is provided by operating procedures and by meeting the requirements of LCO 3.4.12, "Cold Temperature Overpressure Protection System (COPS)."

The upper and lower pressure limits are based on the ± 2% tolerance requirement assumed in the safety analyses. The lift setting is for the ambient conditions associated with

(continued)

BASES

BACKGROUND (continued)

MODES 1, 2, and 3. This requires either that the valves be set hot or that a correlation between hot and cold settings be established.

The pressurizer safety valves are part of the primary success path and mitigate the effects of postulated accidents. OPERABILITY of the safety valves ensures that the RCS pressure will be limited to 110% of design pressure.

The consequences of exceeding the American Society of Mechanical Engineers (ASME) pressure limit (Ref. 1) could include damage to RCS components, increased leakage, or a requirement to perform additional stress analyses prior to resumption of reactor operation.

APPLICABLE SAFETY ANALYSES

All accident and safety analyses in the FSAR (Ref. 2) that require safety valve actuation assume operation of three pressurizer safety valves to limit increases in RCS pressure. The overpressure protection analysis (Ref. 3) is also based on operation of three safety valves. Accidents that could result in overpressurization if not properly terminated include:

a. Uncontrolled rod withdrawal from full power;

b. Loss of reactor coolant flow;

c. Loss of external electrical load;

d. Loss of normal feedwater;

e. Loss of all AC power to station auxiliaries;

f. Locked rotor; and

g. Feedwater line break.

Detailed analyses of the above transients are contained in Reference 2. Safety valve actuation is required in events c, e, and f (above) to limit the pressure increase. Compliance with this LCO is consistent with the design bases and accident analyses assumptions.

(continued)

BASES (continued)

LCO

Pressurizer safety valves satisfy Criterion 3 of the NRC Policy Statement.

The three pressurizer safety valves are set to open at an RCS pressure of 2460 psig, and within the specified tolerance, to avoid exceeding the maximum design pressure SL, and to maintain accident analyses assumptions. The upper and lower pressure tolerance limits are based on the ± 2% tolerance requirements assumed in the safety analyses.

The limit protected by this Specification is the reactor coolant pressure boundary (RCPB) SL of 110% of design pressure.

APPLICABILITY

In MODES 1, 2, and 3, OPERABILITY of three valves is required because the combined capacity is required to keep reactor coolant pressure below 110% of its design value during certain accidents. MODE 3 is conservatively included, although the listed accidents may not require the safety valves for protection.

The LCO is not applicable in MODE 4, MODE 5, or MODE 6 (with the reactor vessel head on) because the cold overpressure protection system is in service. Overpressure protection is not required in MODE 6 with reactor vessel head removed.

The Note allows entry into MODE 3 with the lift settings outside the LCO limits. This permits testing and examination of the safety valves at high pressure and temperature near their normal operating range, but only after the valves have had a preliminary cold setting. The cold setting gives assurance that the valves are OPERABLE near their design condition. Only one valve at a time will be removed from service for testing. The 54 hour exception is based on 18 hour outage time for each of the three valves. The 18 hour period is derived from operating experience that hot testing can be performed in this timeframe.

(continued)

BASES (continued)

ACTIONS

A.1

With one pressurizer safety valve inoperable, restoration must take place within 15 minutes. The Completion Time of 15 minutes reflects the importance of maintaining the RCS overpressure protection system. An inoperable safety valve coincident with an RCS overpressure event could challenge the integrity of the pressure boundary.

B.1 and B.2

If the Required Action of A.1 cannot be met within the required Completion Time or if two or more pressurizer safety valves are inoperable, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In MODE 4, overpressure protection is provided by the cold overpressure protection system. The change from MODE 1, 2, or 3 to MODE 4 reduces the RCS energy (core power and pressure), lowers the potential for large pressurizer insurges, and thereby removes the need for overpressure protection by three pressurizer safety valves.

SURVEILLANCE REQUIREMENTS

SR 3.4.10.1

SRs are specified in the Inservice Testing Program. Pressurizer safety valves are to be tested in accordance with the requirements of Section XI of the ASME Code (Ref. 4), which provides the activities and Frequencies necessary to satisfy the SRs. No additional requirements are specified. The lift settings shall be ≥ 2410 psig and ≤ 2510 psig. The lift setting pressures shall correspond to ambient conditions of the valves at normal operating temperature and pressure.

(continued)

BASES

SURVEILLANCE REQUIREMENTS	SR 3.4.10.1 (continued) The pressurizer safety valve setpoint tolerance is ± 2% for OPERABILITY; however, the valves shall be reset to ± 1% during the surveillance to allow for drift.
REFERENCES	1. ASME, Boiler and Pressure Vessel Code, Section III. 2. FSAR, Chapter 15. 3. WCAP-7769, Rev. 1, June 1972. 4. ASME, Boiler and Pressure Vessel Code, Section XI.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.12.3

Each required RHR suction relief valve shall be demonstrated OPERABLE by verifying its RHR suction isolation valves are open and by testing it in accordance with the Inservice Testing Program. This Surveillance is only required to be performed if the RHR suction relief valve is being used to meet this LCO. For Train A, the RHR suction relief valve is PSV-8708A and the suction isolation valves are HV-8701A and B. For Train B, the RHR suction relief valve is PSV-8708B and the suction isolation valves are HV-8702A and B.

The RHR suction valves are verified to be opened every 12 hours. The Frequency is considered adequate in view of other administrative controls such as valve status indications available to the operator in the control room that verify the RHR suction isolation valves remain open.

The ASME Code, Section XI (Ref. 8), test per Inservice Testing Program verifies OPERABILITY by proving proper relief valve mechanical motion and by measuring and, if required, adjusting the lift setpoint.

SR 3.4.12.4

The RCS vent of $\geq$ 2.14 square inches (based on an equivalent length of 10 feet of pipe) is proven OPERABLE by verifying its open condition either:

a. Once every 12 hours for a valve that cannot be locked.

b. Once every 31 days for a valve that is locked, sealed, or secured in position. A removed pressurizer safety valve fits this category.

The passive vent arrangement must only be open to be OPERABLE. This Surveillance is required to be performed if the vent is being used to satisfy the pressure relief requirements of the LCO 3.4.12b.

SR 3.4.12.5

The PORV block valve must be verified open every 72 hours to provide the flow path for each required PORV to perform its function when actuated. The valve must be remotely verified open in the main control room. This Surveillance is performed if the PORV satisfies the LCO.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.2.2 (continued)

mispositioned are in the correct position. The 31 day Frequency is appropriate because the valves are operated under administrative control, and an improper valve position would only affect a single train. This Frequency has been shown to be acceptable through operating experience.

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. 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 31 day Frequency takes into consideration the gradual nature of gas accumulation in the ECCS piping and the procedural controls governing system operation.

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 Section XI of 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 that the measured performance is within an acceptable tolerance of the original pump baseline performance. SRs are specified in the Inservice Testing Program, which encompasses Section XI of the ASME Code. Section XI of the ASME Code provides the activities and Frequencies necessary to satisfy the requirements.

In addition to the acceptance criteria of the Inservice Testing Program, performance of this SR also verifies that pump performance is greater than or equal to the performance assumed in the safety analysis.

(continued)

BASES

APPLICABILITY (continued)

the containment air locks are based on a fuel handling accident inside containment. The requirements for the containment air locks during MODE 6 are addressed in LCO 3.9.4, "Containment Penetrations."

ACTIONS

The ACTIONS are modified by a Note that allows entry and exit to perform repairs on the affected air lock component. If the outer door is inoperable, then it may be easily accessed to repair. If the inner door is the one that is inoperable, however, then a short time exists when the containment boundary is not intact (during access through the outer door). The ability to open the OPERABLE door, even if it means the containment boundary is temporarily not intact, is acceptable due to the low probability of an event that could pressurize the containment during the short time in which the OPERABLE door is expected to be open. After each entry and exit, the OPERABLE door must be immediately closed.

A second Note has been added to provide clarification that, for this LCO, separate Condition entry is allowed for each air lock.

In the event the air lock leakage results in exceeding the overall containment leakage rate, Note 3 directs entry into the applicable Conditions and Required Actions of LCO 3.6.1, "Containment."

A.1, A.2, and A.3

With one air lock door in one or more containment air locks inoperable, the OPERABLE door must be verified closed (Required Action A.1) in each affected containment air lock. This ensures that a leak tight containment barrier is maintained by the use of an OPERABLE air lock door. This action must be completed within 1 hour. This specified time period is consistent with the ACTIONS of LCO 3.6.1, which requires containment be restored to OPERABLE status within 1 hour.

(continued)

BASES

ACTIONS

A.1, A.2, and A.3 (continued)

Note that for the purpose of Required Actions A.1, A.2, and A.3, the bulkhead associated with an air lock door is considered to be part of the door. For example, an air lock door may be declared inoperable if the associated door shaft seal(s) are replaced or the equalizing valve becomes inoperable, etc. It is appropriate to treat the associated bulkhead as part of the door because a leak path through the bulkhead is no different than a leak path past the door seals. The remaining OPERABLE door/bulkhead provides the necessary barrier between the containment atmosphere and the environs.

In addition, the affected air lock penetration must be isolated by locking closed the OPERABLE air lock door within

(continued)

BASES

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(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.6.2

Operating each pair of containment cooling fan units for ≥ 15 minutes ensures that all fan units are OPERABLE and that all associated controls are functioning properly. It also ensures that blockage or fan or motor failure can be detected for corrective action. The 31 day Frequency was developed considering the known reliability of the fan units and controls, the two train redundancy available, and the low probability of significant degradation of the containment cooling train occurring between surveillances. It has also been shown to be acceptable through operating experience.

SR 3.6.6.3

Verifying that the NSCW flow rate to each pair of units (FI-1818A & B and FI-1819A & B) is ≥ 1359 gpm provides assurance that the design flow rate assumed in the safety analyses will be achieved (Ref. 4). The Frequency was developed considering the known reliability of the Cooling Water System, the two train redundancy available, and the low probability of a significant degradation of flow occurring between surveillances.

SR 3.6.6.4

Verifying each containment spray pump's developed head at the flow test point is greater than or equal to the required developed head ensures that spray pump performance has not degraded during the cycle. Flow and differential pressure are normal tests of centrifugal pump performance required by Section XI of the ASME Code (Ref. 6). Since the containment spray pumps cannot be tested with flow through the spray headers, they are tested on recirculation flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice testing confirms component OPERABILITY, trend performance, and detect incipient failures by abnormal performance. The Frequency of the SR is in accordance with the Inservice Testing Program.

In addition to the acceptance criteria of the Inservice Testing Program, performance of this SR also verifies that

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.6.4 (continued)

pump performance is greater than or equal to the performance assumed in the safety analysis.

SR 3.6.6.5 and SR 3.6.6.6

These SRs require verification that each automatic containment spray valve actuates to its correct position and that each containment spray pump starts upon receipt of an actual or simulated actuation of a containment High-3 pressure signal. This surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 18 month Frequency is based on the need to perform these Surveillances under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillances were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillances when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

The surveillance of containment sump isolation valves is also required by SR 3.5.2.5. A single surveillance may be used to satisfy both requirements.

SR 3.6.6.7

This SR requires verification that each containment cooling train actuates upon receipt of an actual or simulated safety injection signal and operates at low speed. The 18 month Frequency is based on engineering judgment and has been shown to be acceptable through operating experience. See SR 3.6.6.5 and SR 3.6.6.6, above, for further discussion of the basis for the 18 month Frequency.

SR 3.6.6.8

With the containment spray inlet valves closed and the spray header drained of any solution, low pressure air or smoke can be blown through test connections. This SR ensures that

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.6.8 (continued)

each spray nozzle is unobstructed and provides assurance that spray coverage of the containment during an accident is not degraded. Due to the passive design of the nozzle, a test at 10 year intervals is considered adequate to detect obstruction of the nozzles.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 38, GDC 39, GDC 40, GDC 41, GDC 42, and GDC 43.
2. 10 CFR 50, Appendix K.
3. FSAR, Chapter 16.
4. FSAR, Section 6.2.
5. Not used.
6. ASME, Boiler and Pressure Vessel Code, Section XI.

B 3.7 PLANT SYSTEMS

B 3.7.5 Auxiliary Feedwater (AFW) System

BASES

BACKGROUND

The AFW System automatically supplies feedwater to the steam generators to remove decay heat from the Reactor Coolant System upon the loss of normal feedwater supply. The AFW pumps take suction through separate and independent suction lines from the condensate storage tank (CST) (LCO 3.7.6) and pump to the steam generator secondary side via separate and independent connections to the main feedwater (MFW) piping outside containment. The steam generators function as a heat sink for core decay heat. The heat load is dissipated by releasing steam to the atmosphere from the steam generators via the main steam safety valves (MSSVs) (LCO 3.7.1) or atmospheric relief valves (LCO 3.7.4). If the main condenser is available, steam may be released via the steam dump valves and recirculated to the CST.

The AFW System consists of two motor driven AFW pumps and one steam turbine driven pump configured into three trains. The pumps are equipped with independent recirculation lines to prevent pump operation against a closed system. Each motor driven AFW pump is powered from an independent Class 1E power supply and feeds two steam generators, although each pump has the capability to be realigned by local manual valve alignment to feed other steam generators. The steam turbine driven AFW pump receives steam from two main steam lines upstream of the main steam isolation valves. Each of the steam feedlines will supply 100% of the requirements of the turbine driven AFW pump. The turbine driven AFW pump supplies a common header capable of feeding all steam generators with DC powered control valves actuated to the appropriate steam generator by the Engineered Safety Feature Actuation System (ESFAS). Thus, the requirement for diversity in motive power sources for the AFW System is met.

The AFW System is capable of supplying feedwater to the steam generators during normal unit startup, shutdown, hot standby, transient, and accident conditions.

(continued)

BASES

BACKGROUND (continued)

The AFW System is designed to supply sufficient water to the steam generator(s) to remove decay heat with steam generator pressure at the setpoint of the MSSVs. Subsequently, the AFW System supplies sufficient water to cool the unit to RHR entry conditions, with steam released through the ARVs.

The AFW System actuates automatically on steam generator water level — low — low by the ESFAS (LCO 3.3.2). The system also actuates on loss of offsite power, safety injection, or trip of all MFW pumps.

The AFW System is discussed in the FSAR, Subsection 10.4.9 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The AFW System mitigates the consequences of any event with loss of normal feedwater.

The design basis of the AFW System is to supply water to the steam generator to remove decay heat and other residual heat by delivering at least the minimum required flow rate to the steam generators at pressures corresponding to the lowest steam generator safety valve set pressure plus 3%.

In addition, the AFW System must supply enough makeup water to replace steam generator secondary inventory lost as the unit cools to MODE 4 conditions. Sufficient AFW flow must also be available to account for flow losses such as pump recirculation and line breaks. The OPERABILITY of the AFW system in MODE 4 is not assumed in the safety analysis.

The limiting Design Basis Accidents (DBAs) and transients for the AFW System are as follows:

a. Feedwater Line Break (FWLB); and

b. Loss of MFW.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.5.2 (continued)

testing each pump once every 3 months, as required by Ref. 2.

In addition to the acceptance criteria of the Inservice Testing Program, performance of this SR also verifies that pump performance is greater than or equal to the performance assumed in the safety analysis.

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, 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 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The 18 month Frequency is acceptable based on operating experience and the design reliability of the equipment.

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. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

(continue

BASES

ACTIONS (continued)

D.1 and D.2

If the Required Actions of Conditions A, B, or C are not completed within their associated Completion Times or if the UHS is inoperable for reasons other than described in Conditions A, B, or 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 in 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 3.7.9.1

This SR verifies that adequate long term (30 day) cooling can be maintained. The specified level also ensures that sufficient NPSH is available to operate the NSCW System pumps. The 24 hour Frequency is based on operating experience related to trending of the parameter variations during the applicable MODES. This SR verifies that the UHS water level is ≥ 80.25 feet (plant elevation of 217 feet-3 inches or 73% of instrument span on LI-1606 and LI-1607).

SR 3.7.9.2

This SR verifies that the NSCW System is available to cool the CCW System to at least its maximum design temperature with the maximum accident or normal design heat loads for 30 days following a Design Basis Accident. The 24 hour Frequency is based on operating experience related to trending of the parameter variations during the applicable MODES. This SR verifies that the water temperature of the UHS is ≤ 90°F (TJI-1690 and TJI-1691).

SR 3.7.9.3

Operating each NSCW cooling tower fan for ≥ 15 minutes ensures that all fans are OPERABLE and that all associated

(continued)

B 3.7 PLANT SYSTEMS

B 3.7.17 Fuel Storage Pool Boron Concentration

BASES

BACKGROUND

Fuel assemblies are stored in high density racks. The Unit 1 spent fuel storage racks contain storage locations for 1476 fuel assemblies, and the Unit 2 spent fuel storage racks contain storage locations for 2098 fuel assemblies. The Unit 1 racks use boral as a neutron absorber in a flux trap design. The Unit 2 racks contain Boraflex, however, no credit is taken for Boraflex. Westinghouse 17x17 fuel assemblies with initial enrichments of up to and including 5.0 weight percent U-235 can be stored in any location in the Unit 1 or Unit 2 fuel storage pool provided the fuel burnup-enrichment combinations are within the limits that are specified in Figures 3.7.18-1 (Unit 1) or 3.7.18-2 (Unit 2) of the Technical Specifications. Fuel assemblies that do not meet the burnup-enrichment combination of Figures 3.7.18-1 or 3.7.18-2 may be stored in the storage pools of Units 1 or 2 in accordance with checkerboard storage configurations described in Figures 4.3.1-2 through 4.3.1-9. The acceptable fuel assembly storage configurations are based on the Westinghouse Spent Fuel Rack Criticality Methodology, described in WCAP-14416-NP-A, Rev. 1, (Reference 4). This methodology includes computer code benchmarking, spent fuel rack criticality calculations methodology, reactivity equivalencing methodology, accident methodology, and soluble boron credit methodology.

The Westinghouse Spent Fuel Rack Criticality Methodology ensures that the multiplication factor, K_{eff}, of the fuel and spent fuel storage racks is less than or equal to 0.95 as recommended by ANSI 57.2-1983 (Reference 3) and NRC guidance (References 1, 2 and 6). The codes, methods, and techniques contained in the methodology are used to satisfy this criterion on K_{eff}.

The methodology of the NITAWL-II, XSDRNPM-S, and KENO-Va codes is used to establish the bias and bias uncertainty. PHOENIX-P, a nuclear design code used primarily for core reactor physics calculations is used to simulate spent fuel storage rack geometries.

(continued)

BACKGROUND (continued)

Reference 4 describes how credit for fuel storage pool soluble boron is used under normal storage configuration conditions. The storage configuration is defined using K_{eff} calculations to ensure that the K_{eff} will be less than 1.0 with no soluble boron under normal storage conditions including tolerances and uncertainties. Soluble boron credit is then used to maintain K_{eff} less than or equal to 0.95. The Unit 1 pool requires 600 ppm and the Unit 2 pool requires 500 ppm to maintain K_{eff} less than or equal to 0.95 for all allowed combinations of storage configurations, enrichments, and burnups. The analyses assumed 19.9% of the boron atoms have atomic weight 10 (B-10). The effects of B-10 depletion on the boron concentration for maintaining $K_{eff} \leq 0.95$ are negligible. The treatment of reactivity equivalencing uncertainties, as well as the calculation of postulated accidents crediting soluble boron is described in WCAP-14416-NP-A, Rev. 1.

This methodology was used to evaluate the storage of fuel with initial enrichments up to and including 5.0 weight percent U-235 in the Vogtle fuel storage pools. The resulting enrichment, and burnup limits for the Unit 1 and Unit 2 pools, respectively, are shown in Figures 3.7.18-1 and 3.7.18-2. Checkerboard storage configurations are defined to allow storage of fuel that is not within the acceptable burnup domain of Figures 3.7.18-1 and 3.7.18-2. These storage requirements are shown in Figures 4.3.1-2 through 4.3.1-9. A boron concentration of 2000 ppm assures that no credible dilution event will result in a K_{eff} of > 0.95.

APPLICABLE SAFETY ANALYSES

Most fuel storage pool accident conditions will not result in an increase in K_{eff}. Examples of such accidents are the drop of a fuel assembly on top of a rack, and the drop of a fuel assembly between rack modules, or between rack modules and the pool wall.

From a criticality standpoint, a dropped assembly accident occurs when a fuel assembly in its most reactive condition is dropped onto the storage racks. The rack structure from a criticality standpoint is not excessively deformed. Previous accident analysis with unborated water showed that the dropped assembly which comes to rest horizontally on top of the rack has sufficient water separating it from the

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APPLICABLE SAFETY ANALYSES (continued)

active fuel height of stored assemblies to preclude neuronic interaction. For the borated water condition, the interaction is even less since the water contains boron, an additional thermal neutron absorber.

However, three accidents can be postulated for each storage configuration which could increase reactivity beyond the analyzed condition. The first postulated accident would be a change in pool temperature to outside the range of temperatures assumed in the criticality analyses (50°F to 185°F). The second accident would be dropping a fuel assembly into an already loaded cell. The third would be the misloading of a fuel assembly into a cell for which the restrictions on location, enrichment, or burnup are not satisfied.

An increase in the temperature of the water passing through the stored fuel assemblies causes a decrease in water density which results in an addition of negative reactivity for flux trap design racks such as the Unit 1 racks. However, since Boraflex is not considered to be present for the Unit 2 racks and the fuel storage pool water has a high concentration of boron, a density decrease causes a positive reactivity addition. The reactivity effects of a temperature range from 32°F to 240°F were evaluated. The increase in reactivity due to the increase in temperature is bounded by the misload accident, for the Unit 2 racks. The increase in reactivity due to the decrease in temperature below 50°F is bounded by the misplacement of a fuel assembly between the rack and pool walls for the Unit 1 racks.

For the accident of dropping a fuel assembly into an already loaded cell, the upward axial leakage of that cell will be reduced, however, the overall effect on the rack reactivity will be insignificant. This is because the total axial leakage in both the upward and downward directions for the entire fuel array is worth about 0.003 Δk. Thus, minimizing the upward-only leakage of just a single cell will not cause any significant increase in reactivity. Furthermore, the neutronic coupling between the dropped assembly and the already loaded assembly will be low due to several inches of assembly nozzle structure which would separate the active fuel regions. Therefore, this accident would be bounded by the misload accident.

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APPLICABLE SAFETY ANALYSES (continued)

The fuel assembly misloading accident involves placement of a fuel assembly in a location for which it does not meet the requirements for enrichment or burnup, including the placement of an assembly in a location that is required to be left empty. The result of the misloading is to add positive reactivity, increasing K_{eff} toward 0.95. A fourth accident was evaluated for the Unit 1 fuel storage racks containing boral. The fourth accident was the misplacement of a fuel assembly between the rack and pool wall. This was the limiting accident for the Unit 1 racks. The

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BASES

APPLICABLE SAFETY ANALYSES (continued)

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

maximum required additional boron to compensate for this event is 1250 ppm for Unit 2, and 800 ppm for Unit 1 which is well below the limit of 2000 ppm.

The concentration of dissolved boron in the fuel storage pool satisfies Criterion 2 of the NRC Policy Statment.

LCO

The fuel storage pool boron concentration is required to be $\geq$ 2000 ppm. The specified concentration of dissolved boron in the fuel storage pool preserves the assumptions used in the analyses of the potential criticality accident scenarios as described in reference 5. The amount of soluble boron required to offset each of the above postulated accidents was evaluated for all of the proposed storage configurations. That evaluation established the amount of soluble boron necessary to ensure that K_{eff} will be maintained less than or equal to 0.95 should pool temperature exceed the assumed range or a fuel assembly misload occur. The amount of soluble boron necessary to mitigate these events was determined to be 1250 ppm for Unit 2 and 800 ppm for Unit 1. The specified minimum boron concentration of 2000 ppm assures that the concentration will remain above these values. In addition, the boron concentration is consistent with the boron dilution evaluation that demonstrated that any credible dilution event could be terminated prior to reaching the boron concentration for a K_{eff} of > 0.95. These values are 600 ppm for Unit 1 and 500 ppm for Unit 2.

APPLICABILITY

This LCO applies whenever fuel assemblies are stored in the spent fuel storage pool.

ACTIONS

A.1, A.2.1, and A.2.2

The Required Actions are modified by a Note indicating that LCO 3.0.3 does not apply.

When the concentration of boron in the fuel storage pool is less than required, immediate action must be taken to preclude the occurrence of an accident or to mitigate the consequences of an accident in progress. This is most

(continued)

BASES

ACTIONS (continued)

efficiently achieved by immediately suspending the movement of fuel assemblies. Immediate action to restore the concentration of boron is also required simultaneously with suspending movement of fuel assemblies. This does not preclude movement of a fuel assembly to a safe position.

If the LCO is not met while moving irradiated fuel assemblies in MODE 5 or 6, LCO 3.0.3 would not be applicable. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, inability to suspend movement of fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.7.17.1

This SR verifies that the concentration of boron in the fuel storage pool is within the required limit. As long as this SR is met, the analyzed accidents are fully addressed. The 7 day Frequency is appropriate because no major replenishment of pool water is expected to take place over such a short period of time. The gate between the Unit 1 and Unit 2 fuel storage pool is normally open. When the gate is open the pools are considered to be connected for the purpose of conducting the surveillance.

REFERENCES

1. USNRC Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition. NUREG-0800, June 1987.

2. USNRC Spent Fuel Storage Facility Design Bases (for Comment) Proposed Revision 2, 1981. Regulatory Guide 1.13.

3. ANS, "Design Requirements for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations," ANSI/ANS-57.2-1983.

4. WCAP-14416 NP-A, Rev. 1, "Westinghouse Spent Fuel Rack Criticality Analysis Methodology," November 1996.

5. Vogtle FSAR, Section 4.3.2.

6. Nuclear Regulatory Commission, Letter to All Power Reactor Licensees from B. K. Grimes, "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," April 14, 1978.

B 3.7 PLANT SYSTEMS

B 3.7.18 Fuel Assembly Storage in the Fuel Storage Pool

BASES

BACKGROUND

The Unit 1 spent fuel storage racks contain storage locations for 1476 fuel assemblies, and the Unit 2 spent fuel storage racks contain storage locations for 2098 fuel assemblies.

Westinghouse 17X17 fuel assemblies with an enrichment of up to and including 5.0 weight percent U-235 can be stored in the acceptable storage configurations that are specified in Figures 3.7.18-1 (Unit 1), 3.7.18-2 (Unit 2), and 4.3.1-2 through 4.3.1-9. The acceptable fuel assembly storage locations are based on the Westinghouse Spent Fuel Rack Criticality Methodology, described in WCAP-14416-NP-A, Rev. 1 (reference 1). Additional background discussion can be found in B 3.7.17.

Westinghouse 17x17 fuel assemblies with nominal enrichments no greater than 3.50 w/o^{235}U may be stored in all storage cell locations of the Unit 1 pool. Fuel assemblies with initial nominal enrichment greater than 3.50 w/o^{235}U must satisty a minimum burnup requirement as shown in Figure 3.7.18-1. Fuel assemblies having a K_∞ of 1.431 at cold reactor core conditions may also be stored in all cells of the Unit 1 fuel storage racks.

Westinghouse 17x17 fuel assemblies with nominal enrichments no greater than 5.0 w/o^{235}U may be stored in a 3-out-of-4 checkerboard arrangement with empty cells in the Unit 1 pool. There are no minimum burnup requirements for this configuration.

Westinghouse 17x17 fuel assemblies with nominal enrichments no greater that 5.0 w/o^{235}U may be stored in a 2-out-of-4 checkerboard arrangement with empty cells in the Unit 2 pool. There are no minimum burnup requirements for this configuration.

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BASES

BACKGROUND (continued)

Westinghouse 17x17 fuel assemblies with nominal enrichments no greater than 1.77 w/o^{235}U may be stored in all storage cell locations of the Unit 2 pool. Fuel assemblies with initial nominal enrichment greater than 1.77 w/o^{235}U must satisfy a minimum burnup requirement as shown in Figure 3.7.18-2.

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BASES

BACKGROUND
(continued)

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BASES

BACKGROUND (continued)

Westinghouse 17x17 fuel assemblies with nominal enrichments no greater than 2.40 w/o^{235}U may be stored in a 3-out-of-4 checkerboard arrangement with empty cells in the Unit 2 pool. Fuel assemblies with initial nominal enrichment greater than 2.40 w/o^{235}U must satisfy a minimum burnup requirement as shown in Figure 4.3.1-2.

Westinghouse 17x17 fuel assemblies may be stored in the Unit 2 pool in a 3x3 array. The center assembly must have an initial enrichment no greater than 3.20 w/o^{235}U. Alternatively, the center of the 3x3 array may be loaded with any assembly which meets a maximum infinite multiplication factor (K_∞) value of 1.410 at 68°F. One method of achieving this value of K_∞ is by the use of IFBAs. The surrounding fuel assemblies must have an initial nominal enrichment no greater than 1.48 w/o^{235}U or satisfy a minimum burnup requirement for higher initial enrichments as shown in Figure 4.3.1-3.

APPLICABLE SAFETY ANALYSIS

Most fuel storage pool accident conditions will not result in an increase in K_{eff}. Examples of such accidents are the drop of a fuel assembly on top of a rack and the drop of a fuel assembly between rack modules or between rack modules and the pool wall. However, accidents can be postulated for each storage configuration which could increase reactivity beyond the analyzed condition. A discussion of these accidents is contained in B 3.7.17.

The configuration of fuel assemblies in the fuel storage pool satisfies Criterion 2 of the NRC Policy Statement.

LCO

The restrictions on the placement of fuel assemblies within the fuel storage pool ensure the K_{eff} of the fuel storage pool will always remain < 0.95, assuming the pool to be flooded with borated water.

The combination of initial enrichment and burnup are specified in Figures 3.7.18-1 and 3.7.18-2 for all cell storage in the Unit 1 and Unit 2 pools, respectively. Other acceptable enrichment burnup and checkerboard combinations are described in Figures 4.3.1-2 through 4.3.1-9.

(continued)

BASES (continued)

APPLICABILITY

This LCO applies whenever any fuel assembly is stored in the fuel storage pool.

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply.

When the configuration of fuel assemblies stored in the fuel storage pool is not in accordance with the acceptable combination of initial enrichment, burnup, and storage configurations, the immediate action is to initiate action to make the necessary fuel assembly movement(s) to bring the configuration into compliance with Figures 3.7.18-1 (Unit 1), 3.7.18-2 (Unit 2), or Specification 4.3.1.1 (Unit 1) or 4.3.1.2 (Unit 2).

If unable to move irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not be applicable. If unable to move irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the action is independent of reactor operation. Therefore inability to move fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.7.18.1

This SR verifies by administrative means that the initial enrichment and burnup of the fuel assembly is within the acceptable burnup domain of Figures 3.7.18-1 (Unit 1) or 3.7.18-2 (Unit 2). For fuel assemblies in the unacceptable range of Figures 3.7.18-1 and 3.7.18-2, performance of this SR will also ensure compliance with Specification 4.3.1.1 (Unit 1) or 4.3.1.2 (Unit 2).

Fuel assembly movement will be in accordance with preapproved plans that are consistent with the specified fuel enrichment, burnup, and storage configurations. These plans are administratively verified prior to fuel movement. Each assembly is verified by visual inspection to be in accordance with the preapproved plan prior to storage in the fuel storage pool. Storage commences following unlatching of the fuel assembly in the fuel storage pool.

(continued)

BASES (continued)

REFERENCES

1. WCAP-14416-NP-A, Revision 1, "Westinghouse Spent Fuel Rack Criticality Analysis Methodology," November 1996.

BASES

LCO (continued)

train. For the DGs, separation and independence are complete.

For the offsite AC sources, separation and independence are to the extent practical. A circuit may be connected to more than one ESF bus while the bus is being transferred to the other circuit.

APPLICABILITY

The AC sources and sequencers are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure that:

a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of AOOs or abnormal transients; and

b. Adequate core cooling is provided and containment OPERABILITY and other vital functions are maintained in the event of a postulated DBA.

The AC power requirements for MODES 5 and 6 are covered in LCO 3.8.2, "AC Sources — Shutdown."

ACTIONS

A.1

To ensure a highly reliable power source remains with one offsite circuit inoperable, it is necessary to verify the OPERABILITY of the remaining required offsite circuit on a more frequent basis. Since the Required Action only specifies "perform," a failure of SR 3.8.1.1 acceptance criteria does not result in a Required Action not met. However, if a second required circuit fails SR 3.8.1.1, the second offsite circuit is inoperable, and Condition D, for two offsite circuits inoperable, is entered.

A.2

Required Action A.2, which only applies if the train cannot be powered from an offsite source, is intended to provide assurance that an event coincident with a single failure of the associated DG will not result in a complete loss of safety function of critical redundant required features.

(continued)

BASES

ACTIONS

A.2 (continued)

These features are powered from the redundant AC electrical power train. This includes motor driven auxiliary feedwater pumps. Single train systems, such as turbine driven auxiliary feedwater pumps, may not be included.

The Completion Time for Required Action A.2 is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." In this Required Action, the Completion Time only begins on discovery that both:

a. The train has no offsite power supplying its loads; and

b. A required feature on the other train is inoperable.

If at any time during the existence of Condition A (one offsite circuit inoperable) a redundant required feature subsequently becomes inoperable, this Completion Time begins to be tracked.

Discovering no offsite power to one train of the onsite Class 1E Electrical Power Distribution System coincident with one or more inoperable required support or supported features, or both, that are associated with the other train that has offsite power, results in starting the Completion Times for the Required Action. Twenty-four hours is acceptable because it minimizes risk while allowing time for restoration before subjecting the unit to transients associated with shutdown.

The remaining OPERABLE offsite circuit and DGs are adequate to supply electrical power to Train A and Train B of the onsite Class 1E Distribution System. The 24 hour Completion Time takes into account the component OPERABILITY of the redundant counterpart to the inoperable required feature. Additionally, the 24 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

(continued)

BASES

ACTIONS (continued)

A.3

According to Regulatory Guide 1.93 (Ref. 6), operation may continue in Condition A for a period that should not exceed 72 hours. With one required offsite circuit inoperable, the reliability of the offsite system is degraded, and the potential for a loss of offsite power is increased, with attendant potential for a challenge to the unit safety systems. In this Condition, however, the remaining OPERABLE offsite circuit and DGs are adequate to supply electrical power to the onsite Class 1E Distribution System.

The 72 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

The second Completion Time for Required Action A.3 establishes a limit on the maximum time allowed for any combination of required AC power sources to be inoperable during any single contiguous occurrence of failing to meet the LCO. If Condition A is entered while, for instance, a DG is inoperable and that DG is subsequently returned OPERABLE, the LCO may already have been not met for up to 11 days. This could lead to a total of 14 days, since initial failure to meet the LCO, to restore the offsite circuit. At this time, a DG could again become inoperable, the circuit restored OPERABLE, and an additional 72 hours, or 14 days depending on SAT availability, allowed prior to complete restoration of the LCO. The 14 day Completion Time provides a limit on the time allowed in a specified condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which Conditions A and B are entered concurrently. The "AND" connector between the 72 hour and 14 day Completion Times means that both Completion Times apply simultaneously, and the more restrictive Completion Time must be met.

Tracking the 14 day Completion Time is a requirement for beginning the Completion Time "clock" that is in addition to the normal Completion Time requirements. With respect to the 14 day Completion Time, the "time zero" is specified as

(continued)

ACTIONS

A.3 (continued)

commencing at the time LCO 3.8.1 was initially not met, instead of at the time Condition A was entered. This results in the requirement when in this Condition to track the time elapsed from both the Condition A "time zero" and the "time zero" when LCO 3.8.1 was initially not met.

B.1

To ensure a highly reliable power source remains with an inoperable DG, it is necessary to verify the availability of the offsite circuits on a more frequent basis. Since the Required Action only specifies "perform," a failure of SR 3.8.1.1 acceptance criteria does not result in a Required Action being not met. However, if a circuit fails to pass SR 3.8.1.1, it is inoperable. Upon offsite circuit inoperability, additional Conditions and Required Actions must then be entered.

B.2

The 13.8/4.16 kV Standby Auxiliary Transformer (SAT) is a qualified offsite circuit that may be connected to the onsite Class 1E distribution system independently of the RATs and may be utilized to meet the LCO 3.8.1 requirements for an offsite circuit. Its availability permits an extension of the allowable out-of-service time for a DG to 14 days from the discovery of failure to meet LCO 3.8.1. The SAT is available when it is:

- Operable in accordance with plant procedures;
- Not already being applied to any of the four 4.16 kV ESF buses for Units 1 and 2 in accordance with Specification 3.8.1 as either an offsite source or to meet the requirements of an LCO 3.8.1 Condition; and,
- Not providing power to the other unit when that unit is in MODE 5 or 6 or defueled.

(continued)

BASES

ACTIONS

B.2 (continued)

Furthermore, the SAT can be applied to only one of the four 4.16 kV ESF buses at any given time for Units 1 and 2 to meet the requirements of an LCO 3.8.1 Condition.

When one or more of these criteria are not satisfied, the SAT is not available. These criteria are structured to ensure that the SAT is available as an alternate offsite source to support the extended DG Completion Time of 14 days. Therefore, when a DG is inoperable, it is necessary to verify the availability of the SAT within one hour and once per 12 hours thereafter. If Required Action B.2 is not met or the status of the SAT changes after Required Action B.2 is initially met, Condition C must be entered concurrently.

B.3

Required Action B.3 is intended to provide assurance that a loss of offsite power, during the period that a DG is inoperable, does not result in a complete loss of safety function of critical systems. These features are designed with redundant safety related trains. This includes motor driven auxiliary feedwater pumps. Single train systems, such as turbine driven auxiliary feedwater pumps, are not included. Redundant required feature failures consist of inoperable features associated with a train, redundant to the train that has an inoperable DG.

The Completion Time for Required Action B.3 is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." In this Required Action, the Completion Time only begins on discovery that both:

a. An inoperable DG exists; and

b. A required feature on the other train (Train A or Train B) is inoperable.

(continued)

BASES

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(continued)

BASES

ACTIONS

B.3 (continued)

If at any time during the existence of this Condition (one DG inoperable) a required feature subsequently becomes inoperable, this Completion Time would begin to be tracked.

Discovering one required DG inoperable coincident with one or more inoperable required support or supported features, or both, that are associated with the OPERABLE DG, results in starting the Completion Time for the Required Action. Four hours from the discovery of these events existing concurrently is acceptable because it minimizes risk while allowing time for restoration before subjecting the unit to transients associated with shutdown.

In this Condition, the remaining OPERABLE DG and offsite circuits are adequate to supply electrical power to the onsite Class 1E Distribution System. Thus, on a component basis, single failure protection for the required feature's function may have been lost; however, function has not been lost. The 4 hour Completion Time takes into account the OPERABILITY of the redundant counterpart to the inoperable required feature. Additionally, the 4 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

B.4.1 and B.4.2

Required Action B.4.1 provides an allowance to avoid unnecessary testing of the OPERABLE DG. If it can be determined that the cause of the inoperable DG does not exist on the OPERABLE DG, SR 3.8.1.2 does not have to be performed. If the cause of inoperability exists on the other DG, the other DG would be declared inoperable upon discovery and Condition F of LCO 3.8.1 would be entered. Once the failure is repaired, the common cause failure no longer exists, and Required Action B.4.1 is satisfied. If the cause of the initial inoperable DG cannot be confirmed not to exist on the remaining DG, performance of SR 3.8.1.2 suffices to provide assurance of continued OPERABILITY of that DG.

(continued)

BASES

ACTIONS

B.4.1 and B.4.2 (continued)

In the event the inoperable DG is restored to OPERABLE status prior to completing either B.4.1 or B.4.2, the applicable plant procedures will continue to require the evaluation of the common cause possibility. This continued evaluation, however, is no longer under the 24 hour constraint imposed while in Condition B.

According to Generic Letter 84-15 (Ref. 7), 24 hours is reasonable to confirm that the OPERABLE DG is not affected by the same problem as the inoperable DG.

B.5.1 and B.5.2

Required Action B.5.1 provides assurance that an enhanced black-start combustion turbine generator (CTG) is functional when a DG is out of service for greater than 72 hours. Required Action B.5.1 is modified by a Note that states that it is only applicable provided that the two enhanced black-start CTGs and black-start diesel generator have a combined reliability of ≥ 95% based on a minimum of 20 tests per enhanced black-start CTG and quarterly testing thereafter. This quarterly testing will subject each enhanced black-start CTG to a start and load-run test. The black-start diesel generator will also be tested quarterly, but separately from the enhanced black-start CTGs. Required Action B.5.1 may be met by starting either of the enhanced black-start CTGs and the black-start diesel generator and verifying that they achieve steady state voltage and frequency. The black-start diesel generator may be started separately.

If a DG is to be removed from service voluntarily for greater than 72 hours, it may be advantageous to test an enhanced black-start CTG prior to taking the DG out of service. In such cases where advanced notice of removing a DG from service is available, Required Action B.5.1 may be performed up to 72 hours prior to entry into Condition B. In other cases, Required Action B.5.1 must be performed within 72 hours after entry into Condition B.

(continued)

BASES

ACTIONS

B.5.1 and B.5.2 (continued)

If the combined reliability of the enhanced black-start CTGs has not been demonstrated or maintained ≥ 95%, the option of starting and running any one of the six CTGs while in Condition B is available in the form of Required Action B.5.2. In the event of preplanned maintenance that would exceed 72 hours, any one of the six CTGs must be started prior to entry into Condition B and allowed to run for the duration of Condition B. Otherwise, any one of the six CTGs must be started within 72 hours (and allowed to run) after entry into Condition B if the DG is to be out of service for more than 72 hours. Note that Required Action B.5.1 requires that one of the two enhanced black-start CTGs be started, but any one of the six CTGs could be started to satisfy Required Action B.5.2. Since a CTG is started and running while the DG is inoperable, it is not necessary that the CTG have enhanced black-start capability.

B.6

The availability of the SAT provides an additional AC source which permits operation to continue for a period not to exceed 14 days from discovery of failure to meet the LCO.

In Condition B, the remaining OPERABLE DG and offsite circuits are adequate to supply electrical power to the onsite Class 1E Distribution System. The 14 day Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

In addition, the Configuration Risk Management Program (CRMP) is used to assess changes in core damage frequency resulting from applicable plant configurations. The CRMP uses the equipment out of service risk monitor, a computer based tool that may be used to aid in the risk assessment of on-line maintenance and to evaluate the change in risk from a component failure. The equipment out of service risk monitor uses the plant probabilistic risk assessment model to evaluate the risk of removing equipment from service

(continued)

BASES

ACTIONS

B.6 (continued)

based on current plant configuration and equipment condition. The CRMP is used when a DG is intentionally taken out of service for a planned activity excluding short duration activities (e.g., performing an air roll on the EDG prior to a routine surveillance). In addition, the CRMP is used for unplanned maintenance or repairs of a DG.

Planned activities involving an extended DG AOT will be synchronized with other maintenance activities as much as possible in order to maximize equipment reliability while minimizing the time equipment is unavailable. In addition, Required Action B.3 requires that features supported by the inoperable DG be declared inoperable within 4 hours of discovery when redundant features are discovered to be inoperable. The combination of planned maintenance centered around the extended DG AOT, Required Action B.3, and use of the CRMP provides an appropriate level of assurance that risk significant activities with an unacceptable risk achievement worth will be minimized during an extended DG AOT.

The Completion Time for Required Action B.6 also establishes a limit on the maximum time allowed for any combination of required AC power sources to be inoperable during any single contiguous occurrence of failing to meet the LCO. If Condition B is entered while, for instance, an offsite circuit is inoperable, the LCO may already have been not met for up to 72 hours. If the offsite circuit is restored within the required 72 hours, this could lead to a total of 17 days, since initial failure to meet the LCO, to restore compliance with the LCO (i.e., restore the DG). However, the 14 day Completion Time provides a

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BASES

ACTIONS

B.6 (continued)

limit on time allowed in a specified condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which Conditions A and B (and consequently Condition E) are entered concurrently.

Tracking the 14 day Completion Time is a requirement for beginning the Completion Time "clock" that is in addition to the normal Completion Time requirements. With respect to the Completion Time, the "time zero" is specified as commencing at the time LCO 3.8.1 was initially not met, instead of at the time Condition B was entered. This results in the requirement when in this Condition to track the time elapsed from both the Condition B "time zero" and the "time zero" when LCO 3.8.1 was initially not met.

C.1

If the availability of the SAT cannot be verified, or if no CTG meets the requirements of either Required Action B.5.1 or B.5.2, the DG must be restored to OPERABLE status within 72 hours. The 72 hour Completion Time begins upon entry into Condition C. However, the total time to restore an inoperable DG cannot exceed 14 days (per the Completion Time of Required Action B.6).

The Completion Time of 72 hours (in the absence of the SAT) is consistent with Regulatory Guide 1.93 (Ref.6). The 72 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and low probability of a DBA occurring this period.

D.1 and D.2

Required Action D.1, which applies when two offsite circuits are inoperable, is intended to provide assurance that an event with a coincident single failure will not result in a complete loss of redundant required safety functions. The Completion Time for this failure of redundant required features is reduced to 12 hours from that allowed for one train without offsite power (Required Action A.2). The rationale for the reduction to 12 hours is that Regulatory Guide 1.93 (Ref. 6) allows a Completion Time of 24 hours for

(continued)

BASES

ACTIONS

D.1 and D.2 (continued)

two required offsite circuits inoperable, based upon the assumption that two complete safety trains are OPERABLE. When a concurrent redundant required feature failure exists, this assumption is not the case, and a shorter Completion Time of 12 hours is appropriate. These features are powered from redundant AC safety trains. This includes motor driven auxiliary feedwater pumps. Single train features, such as turbine driven auxiliary pumps, are not included in the list.

The Completion Time for Required Action D.1 is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for

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BASES

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(continued)

ACTIONS

D.1 and D.2 (continued)

beginning the allowed outage time "clock." In this Required Action the Completion Time only begins on discovery that both:

a. All required offsite circuits are inoperable; and

b. A required feature is inoperable.

If at any time during the existence of Condition D (two offsite circuits inoperable) a required feature becomes inoperable, this Completion Time begins to be tracked.

According to Regulatory Guide 1.93 (Ref. 6), operation may continue in Condition D for a period that should not exceed 24 hours. This level of degradation means that the offsite electrical power system does not have the capability to effect a safe shutdown and to mitigate the effects of an accident; however, the onsite AC sources have not been degraded. This level of degradation generally corresponds to a total loss of the immediately accessible offsite power sources.

Because of the normally high availability of the offsite sources, this level of degradation may appear to be more severe than other combinations of two AC sources inoperable that involve one or more DGs inoperable. However, two factors tend to decrease the severity of this level of degradation:

a. The configuration of the redundant AC electrical power system that remains available is not susceptible to a single bus or switching failure; and

b. The time required to detect and restore an unavailable offsite power source is generally much less than that required to detect and restore an unavailable onsite AC source.

With both of the required offsite circuits inoperable, sufficient onsite AC sources are available to maintain the unit in a safe shutdown condition in the event of a DBA or transient. In fact, a simultaneous loss of offsite AC sources, a LOCA, and a worst case single failure were

(continued)

BASES

ACTIONS

D.1 and D.2 (continued)

postulated as a part of the design basis in the safety analysis. Thus, the 24 hour Completion Time provides a period of time to effect restoration of one of the offsite circuits commensurate with the importance of maintaining an AC electrical power system capable of meeting its design criteria.

According to Reference 6, with the available offsite AC sources, two less than required by the LCO, operation may continue for 24 hours. If two offsite sources are restored within 24 hours, unrestricted operation may continue. If only one offsite source is restored within 24 hours, power operation continues in accordance with Condition A.

E.1 and E.2

Pursuant to LCO 3.0.6, the Distribution System ACTIONS would not be entered even if all AC sources to it were inoperable, resulting in de-energization. Therefore, the Required Actions of Condition E are modified by a Note to indicate that when Condition E is entered with no AC source to one or more trains, the Conditions and Required Actions for LCO 3.8.9, "Distribution Systems — Operating," must be immediately entered. This allows Condition E to provide requirements for the loss of one offsite circuit and one DG, without regard to whether a train is de-energized. LCO 3.8.9 provides the appropriate restrictions for a de-energized train.

According to Regulatory Guide 1.93 (Ref. 6), operation may continue in Condition E for a period that should not exceed 12 hours.

In Condition E, individual redundancy is lost in both the offsite electrical power system and the onsite AC electrical power system. Since power system redundancy is provided by two diverse sources of power, however, the reliability of the power systems in this Condition may appear higher than that in Condition D (loss of both required offsite circuits). This difference in reliability is offset by the susceptibility of this power system configuration to a

(continued)

BASES

ACTIONS

E.1 and E.2 (continued)

single bus or switching failure. The 12 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

F.1

With Train A and Train B DGs inoperable, there are no remaining standby AC sources. Thus, with an assumed loss of offsite electrical power, insufficient standby AC sources are available to power the minimum required ESF functions. Since the offsite electrical power system is the only source of AC power for this level of degradation, the risk associated with continued operation for a very short time could be less than that associated with an immediate controlled shutdown (the immediate shutdown could cause grid instability, which could result in a total loss of AC power). Since any inadvertent generator trip could also result in a total loss of offsite AC power, however, the time allowed for continued operation is severely restricted. The intent here is to avoid the risk associated with an immediate controlled shutdown and to minimize the risk associated with this level of degradation.

According to Reference 6, with both DGs inoperable, operation may continue for a period that should not exceed 2 hours.

G.1

The sequencer(s) is an essential support system to both the offsite circuit and the DG associated with a given ESF bus. Furthermore, the sequencer is on the primary success path for most major AC electrically powered safety systems powered from the associated ESF bus. The sequencers are required to provide the system response to both an SI signal and a loss of or degraded ESF bus voltage signal. Therefore, loss of an ESF bus sequencer affects every major ESF system in the train. The 12 hour Completion Time provides a period of time to correct the problem commensurate with the importance of maintaining sequencer

(continued)

BASES

ACTIONS

G.1 (continued)

OPERABILITY. This time period also ensures that the probability of an accident (requiring sequencer OPERABILITY) occurring during periods when the sequencer is inoperable is minimal.

H.1 and H.2

If the inoperable AC electric power sources or an automatic load sequencer cannot be restored to OPERABLE status within the required Completion Time, or Required Actions B.1, B.3, B.4.1, B.4.2, or B.6 cannot be met within the required Completion Times, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging plant systems.

I.1

Condition I corresponds to a level of degradation in which all redundancy in the AC electrical power supplies has been lost. At this severely degraded level, any further losses in the AC electrical power system will cause a loss of unction. Therefore, no additional time is justified for continued operation. The unit is required by LCO 3.0.3 to commence a controlled shutdown.

SURVEILLANCE REQUIREMENTS

The AC sources are designed to permit inspection and testing of all important areas and features, especially those that have a standby function, in accordance with 10 CFR 50, Appendix A, GDC 18 (Ref. 8). Periodic component tests are supplemented by extensive functional tests during refueling outages (under simulated accident conditions). The SRs for demonstrating the OPERABILITY of the DGs are in accordance with the recommendations of Regulatory Guide 1.9 (Ref. 3), Regulatory Guide 1.108 (Ref. 9), and Regulatory Guide 1.137 (Ref. 10), as addressed in the FSAR.

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SURVEILLANCE REQUIREMENTS

SR 3.8.1.11 (continued)

2. Post Corrective maintenance testing that requires performance of this Surveillance in order to restore the component to OPERABLE, provided the maintenance was required, or performed in conjunction with maintenance required to maintain OPERABILITY or reliability.

SR 3.8.1.12

This Surveillance demonstrates that DG noncritical protective functions (e.g., high jacket water temperature) are bypassed on a loss of voltage signal concurrent with an ESF actuation test signal. The noncritical trips are bypassed during DBAs and provide an alarm on an abnormal engine condition. This alarm provides the operator with sufficient time to react appropriately. The DG availability to mitigate the DBA is more critical than protecting the engine against minor problems that are not immediately detrimental to emergency operation of the DG.

The 18 month Frequency is based on engineering judgment, taking into consideration unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths. Operating experience has shown that these components usually pass the SR when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

The SR is modified by a Note. The reason for the Note is that performing the Surveillance would remove a required DG from service. Credit may be taken for unplanned events that satisfy this SR. Examples of unplanned events may include:

1. Unexpected operational events which cause the equipment to perform the function specified by this Surveillance, for which adequate documentation of the required performance is available; and

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BASES

BACKGROUND (continued)

Each DG building contains two ventilation supply fans and associated dampers. The ventilation supply fans are required to limit the DG building air temperature to ≤ 120° F to support the operation of the associated DG. The fans in each DG building and associated dampers start and actuate on different signals. Fans 1/2-1566-B7-001 (train A) and 1/2-1566-B7-002 (train B) start automatically and the necessary intake and discharge dampers actuate to the correct position on a train associated DG running signal and fans 1/2-1566-B7-003 and 1/2-1566-B7-004 start automatically and the necessary intake and discharge dampers actuate to the correct position on high DG building temperature signal coincident with a DG running signal.

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the FSAR, Chapter 6 (Ref. 4), and in the FSAR, Chapter 15 (Ref. 5), assume Engineered Safety Feature (ESF) systems are OPERABLE. The DGs are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that fuel, Reactor Coolant System and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for Section 3.2, Power Distribution Limits; Section 3.4, Reactor Coolant System (RCS); and Section 3.6, Containment Systems.

Since diesel fuel oil, lube oil, air start, and ventilation subsystems support the operation of the standby AC power sources, they satisfy Criterion 3 of the NRC Policy Statement.

LCO

Stored diesel fuel oil is required to have sufficient supply for 7 days of full load operation. In MODES 1, 2, 3, and 4, both storage tanks are required to provide for ≥ 7 days of operation supplying the maximum post loss of coolant accident load demand. However, in MODES 5 and 6, the highest DG loading identified for either train is significantly less than the maximum post loss of coolant accident loading for MODES 1 through 4, and the capacity of one storage tank is sufficient to provide for ≥ 7 days of DG operation. It is also required to meet specific standards for quality. Additionally, sufficient lubricating

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BASES

LCO (continued)

oil supply must be available to ensure the capability to operate at full load for 7 days. This requirement, in conjunction with an ability to obtain replacement supplies within 7 days, supports the availability of DGs required to shut down the reactor and to maintain it in a safe condition for an anticipated operational occurrence (AOO) or a postulated DBA with loss of offsite power. DG day tank fuel requirements, as well as

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BASES

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BASES

ACTIONS

A.1 (continued)

These circumstances may be caused by events, such as full load operation required after an inadvertent start while at minimum required level, or feed and bleed operations, which may be necessitated by increasing particulate levels or any number of other oil quality degradations. This restriction allows sufficient time for obtaining the requisite replacement volume and performing the analyses required prior to addition of fuel oil to the tank. A period of 48 hours is considered sufficient to complete restoration of the required level prior to declaring the DG inoperable. This period is acceptable based on the remaining capacity (> 4 days), the fact that procedures will be initiated to obtain replenishment, and the low probability of an event during this brief period.

Note that the above discussion is applicable to MODES 1, 2, 3, and 4. In MODES 5 and 6, the highest load demand identified for the DGs is sufficiently small that a single storage tank will provide for ≥ 7 days of DG operation. However, if the stored fuel oil in the required storage tank is found to be < 68,000 gallons and > 52,000 gallons during MODES 5 and 6, Condition A and Required Action A.1 continue to apply.

B.1

With lube oil inventory < 336 gal, sufficient lubricating oil to support 7 days of continuous DG operation at full load conditions may not be available. However, the Condition is restricted to lube oil volume reductions that maintain at least a 6 day supply. These values are based on Reference 10. This restriction allows sufficient time to obtain the requisite replacement volume. A period of 48 hours is considered sufficient to complete restoration of the required volume prior to declaring the DG inoperable. This period is acceptable based on the remaining capacity (> 6 days), the low rate of usage, the fact that procedures will be initiated to obtain replenishment, and the low probability of an event during this brief period.

(continued)

BASES

ACTIONS (continued)

C.1

This Condition is entered as a result of a failure to meet the acceptance criterion of the particulate component for stored fuel oil of SR 3.8.3.3. Normally, trending of particulate levels allows sufficient time to correct high particulate levels prior to reaching the limit of acceptability. Poor sample technique (e.g., bottom sampling), contaminated sampling equipment, and errors in laboratory analysis can produce failures that do not follow

(continued)

BASES

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(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.3.1 (continued)

Note that in MODES 1, 2, 3, and 4, both storage tanks are required to provide for ≥ 7 days operation at full load. In MODES 5 and 6 only one storage tank is required to provide for ≥ 7 days DG operation.

(continued)

BASES

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(continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.4 DC Sources — Operating

BASES

BACKGROUND

There are four safety features 125 VDC systems (identified A, B, C, and D) per unit. Each system has a 59-cell lead calcium battery, switchgear (electrically operated drawout circuit breakers), two redundant battery chargers, and 125 VDC distribution panels (molded case circuit breakers). Systems A, B, and C each have a 125 VDC motor control center for motor operated valves. There is no capability to connect the DC systems between themselves, between Unit 1 and Unit 2 systems, or between the safety features systems and the nonsafety features systems. Table B 3.8.4-1 shows the DC sources and train associations. The 125 VDC systems A and C form the train A safety features DC system and their associated battery chargers receive power from two Class 1E train A motor control centers. The 125 VDC systems B and D form the train B safety features DC system and their battery chargers receive power from two Class 1E train B motor control centers.

The 125 VDC systems A, B, C, and D supply DC power to channels 1, 2, 3, and 4, respectively, and are designated as Class 1E equipment in accordance with the applicable sections of Institute of Electrical and Electronic Engineers (IEEE) Standard 308 (Ref. 1). As required by 10 CFR 50, Appendix A, GDC 17 (Ref. 2), the DC electrical power system is designed so that no single failure in any 125 VDC system will result in conditions that will prevent the safe shutdown of the reactor plant. The plant design and circuit layout from these DC systems provide physical separation of equipment, cabling, and instrumentation essential to plant safety. Each 125 VDC battery is separately housed in a ventilated room apart from its chargers and distribution equipment. All the components of the 125 VDC Class 1E systems are housed in Category 1 structures.

During normal operation the 125 VDC load is powered from the battery chargers with the batteries floating on the system. In case of loss of normal power to the battery chargers, the DC load is automatically powered from the batteries.

(continued)

BASES

BACKGROUND (continued)

Batteries are sized in accordance with IEEE 485 (Ref. 3) to have sufficient capacity to supply the required loads for a loss of coolant/loss of offsite power (LOCA/LOSP) duration of 2 3/4 hours and a station blackout (SBO) duration of 4 hours. For LOSP/LOCA, they are sized at a minimum temperature of 70°F; their initial capacity was increased by 10% for load growth and 25% for aging. The required final (end of duty cycle and end of life) battery cell voltages for each load group have been analyzed to demonstrate that adequate voltage is provided to the loads. The battery voltage specifications are discussed in detail for each load group in FSAR, Chapter 8 (Ref. 4).

Each 125 VDC battery is provided with two battery chargers, each of which is sized to supply the continuous (long term) demand on its associated DC system while providing sufficient power to replace 110% of the equivalent ampere-hours removed from the battery during a design basis battery discharge cycle within a 12 hour period after charger input power is restored. Normally, both battery chargers are on line with load sharing circuitry to ensure that the DC load is properly shared between the two chargers. Only one charger is required OPERABLE to support the associated DC power system. The sizing of each battery charger meets the requirements of IEEE 308 (Ref. 1) and Regulatory Guide 1.32 (Ref. 5).

The DC power distribution system is described in more detail in Bases for LCO 3.8.9, "Distribution System — Operating," and LCO 3.8.10, "Distribution Systems — Shutdown."

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the FSAR, Chapter 6 (Ref. 6), and in the FSAR, Chapter 15 (Ref. 7), assume that Engineered Safety Feature (ESF) systems are OPERABLE. The DC electrical power system provides normal and emergency DC electrical power for the DGs, emergency auxiliaries, and control and switching during all MODES of operation.

The OPERABILITY of the DC sources is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the unit. This includes maintaining the DC sources OPERABLE during accident conditions in the event of:

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

a. An assumed loss of all offsite AC power or all onsite AC power; and

b. A worst case single failure.

The DC sources satisfy Criterion 3 of the NRC Policy Statement.

LCO

The DC electrical power sources, each source consisting of one battery, battery charger, and the corresponding control equipment and interconnecting cabling supplying power to the associated bus within the train are required to be OPERABLE to ensure the availability of the required power to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence (AOO) or a postulated DBA. Loss of any train DC electrical power source does not prevent the minimum safety function from being performed (Ref. 4).

An OPERABLE DC electrical power source requires the battery and one charger per battery to be operating and connected to the associated DC bus.

APPLICABILITY

The DC electrical power sources are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure safe unit operation and to ensure that:

a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of AOOs or abnormal transients; and

b. Adequate core cooling is provided, and containment integrity and other vital functions are maintained in the event of a postulated DBA.

The DC electrical power requirements for MODES 5 and 6 are addressed in the Bases for LCO 3.8.5, "DC Sources — Shutdown."

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.4.6 (continued)

irrespective of the status of the unit during these demand occurrences. The minimum required amperes and duration ensures that these requirements can be satisfied.

The Surveillance Frequency is acceptable, given the unit conditions required to perform the test and the other administrative controls existing to ensure adequate charger performance during these 18 month intervals. In addition, this Frequency is intended to be consistent with expected fuel cycle lengths.

For a battery charger with charger output aligned to the associated 1E 125 VDC bus, this Surveillance is required to be performed during MODES 5 and 6 since it would require the DC electrical power subsystem to be inoperable during performance of the test.

This SR is modified by a Note. The reason for the Note is that performing the Surveillance on a battery charger with charger output aligned to the associated 1E 125 VDC bus would perturb the electrical distribution system and challenge safety systems. This note is not intended to restrict performance of the SR on a redundant battery charger that is not in service supplying the associated 125 VDC bus. Credit may be taken for unplanned events that satisfy this SR. Examples of unplanned events may include:

1. Unexpected operational events which cause the equipment to perform the function specified by this Surveillance, for which adequate documentation of the required performance is available; and

2. Post Corrective maintenance testing that requires performance of this Surveillance in order to restore the component to OPERABLE, provided the maintenance was required, or performed in conjunction with maintenance required to maintain OPERABILITY or reliability.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.4.7

A battery service test is a special test of battery capability, as found, to satisfy the design requirements (battery duty cycle) of the DC electrical power system. The discharge rate and test length should correspond to the design duty cycle requirements as specified in Reference 4.

The Surveillance Frequency of 18 months is consistent with the recommendations of Regulatory Guide 1.32 (Ref. 5) and Regulatory Guide 1.129 (Ref. 10), which state that the battery service test should be performed during refueling operations or at some other outage, with intervals between tests, not to exceed 18 months.

This SR is modified by two Notes. Note 1 allows the performance of a modified performance discharge test in lieu of a service test.

A modified discharge test is a test of the battery capacity and its ability to provide a high rate, short duration load (usually the highest rate of the duty cycle). This will confirm the battery's ability to meet the critical period of the load duty cycle, in addition to determining its percentage of rated capacity. The modified discharge test will be performed in accordance with the guidance provided in IEEE-450 (Ref. 11) with the exception that the battery electrolyte temperature may be corrected after the test using a methodology approved by the battery manufacturer.

The reason for Note 2 is that performing the Surveillance would perturb the electrical distribution system and challenge safety systems. Credit may be taken for unplanned events that satisfy this SR. Examples of unplanned events may include:

1. Unexpected operational events which cause the equipment to perform the function specified by this Surveillance, for which adequate documentation of the required performance is available; and

2. Post Corrective maintenance testing that requires performance of this Surveillance in order to restore the component to OPERABLE, provided the maintenance was required, or performed in conjunction with maintenance required to maintain OPERABILITY or reliability.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.4.8

A battery performance discharge test is a test of constant current capacity of a battery, normally done in the as found condition, after having been in service, to detect any change in the capacity. The test is intended to determine overall battery degradation due to age and usage.

A battery modified performance discharge test is described in the Bases for SR 3.8.4.7. Either the battery performance discharge test or the modified performance discharge test is acceptable for satisfying SR 3.8.4.8; however, only the modified performance discharge test may be used to satisfy SR 3.8.4.8 while satisfying the requirements of SR 3.8.4.7 at the same ime.

The acceptance criteria for this Surveillance are consistent with IEEE-450 (Ref. 9) and IEEE-485 (Ref. 3). These references recommend that the battery be replaced if its capacity is below 80% of the manufacturer rating. A capacity of 80% shows that the battery rate of deterioration is increasing, even if there is ample capacity to meet the load requirements.

The Surveillance Frequency for this test as specified in Table 3.8.4-1 is 60 months when the battery is less than or equal to 85% of its expected life with no degradation and 12 months if the battery shows degradation and is less than or equal to 85% of its expected life. When the battery has exceeded 85% of its expected life with no degradation, the Frequency becomes 24 months. Degradation is indicated, according to IEEE-450 (Ref. 9), when the battery capacity drops by more than 10% relative to its capacity on the previous performance test or when it is ≥ 10% below the manufacturer rating. These Frequencies are similar to those recommended by IEEE-450 (Ref. 9) and require that testing be performed in a conservative manner relative to the battery life and degradation which in turn will ensure that battery capacity is adequately monitored and that the battery remains capable of performing its intended function.

This SR is modified by a Note. The reason for Note 1 is that performing the Surveillance would perturb the electrical distribution system and challenge safety systems.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.4.8 (continued)

Credit may be taken for unplanned events that satisfy this SR. Examples of unplanned events may include:

1. Unexpected operational events which cause the equipment to perform the function specified by this Surveillance, for which adequate documentation of the required performance is available; and

2. Post Corrective maintenance testing that requires performance of this Surveillance in order to restore the component to OPERABLE, provided the maintenance was required, or performed in conjunction with maintenance required to maintain OPERABILITY or reliability.

REFERENCES

1. IEEE-308-1978.
2. 10 CFR 50, Appendix A, GDC 17.
3. IEEE-485-1983, June 1983.
4. FSAR, Chapter 8.
5. Regulatory Guide 1.32, February 1977.
6. FSAR, Chapter 6.
7. FSAR, Chapter 15.
8. Regulatory Guide 1.93, December 1974.
9. IEEE-450-1975 and 1987.
10. Regulatory Guide 1.129, December 1974.
11. IEEE-450-1995.

Table B 3.8.4-1
DC Sources

TYPE	VOLTAGE	TRAIN A	TRAIN B
DC sources	125 V	System A Battery 1/2AD1B One charger 1/2AD1CA or 1/2AD1CB *Bus powered by system A 1/2AD1	System B Battery 1/2BD1B One charger 1/2BD1CA or 1/2BD1CB *Bus powered by system B 1/2BD1
	125 V	System C Battery 1/2CD1B One charger 1/2CD1CA or 1/2CD1CB *Bus powered by system C 1/2CD1	System D Battery 1/2DD1B One charger 1/2DD1CA or 1/2DD1CB *Bus powered by system D 1/2DD1

* Operability requirements for the buses are addressed in Specifications 3.8.9, Distribution Systems — Operating, or 3.8.10, Distribution Systems — Shutdown.

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B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.5 DC Sources — Shutdown

BASES

BACKGROUND

A description of the DC sources is provided in the Bases for LCO 3.8.4, "DC Sources — Operating."

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident and transient analyses in the FSAR, Chapter 6 (Ref. 1) and Chapter 15 (Ref. 2), assume that Engineered Safety Feature systems are OPERABLE. The DC electrical power system provides normal and emergency DC electrical power for the diesel generators, emergency auxiliaries, and control and switching during all MODES of operation.

The OPERABILITY of the DC subsystems is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY.

The OPERABILITY of the minimum DC electrical power sources during MODES 5 and 6 and during movement of irradiated fuel assemblies ensures that:

a. The unit can be maintained in the shutdown or refueling condition for extended periods;

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and

c. Adequate DC electrical power is provided to mitigate events postulated during shutdown, such as a fuel handling accident.

The DC sources satisfy Criterion 3 of the NRC Policy Statement.

LCO

The DC electrical power sources required to support the necessary portions of AC, DC, and AC vital bus electrical

(continued)

BASES

LCO (continued)

power distribution subsystems required by LCO 3.8.10, "Distribution Systems — Shutdown," shall be OPERABLE. At a minimum, at least one train of DC electrical power sources with each DC source within the train (Systems A and C OR Systems B and D) consisting of one battery, and one required battery charger per battery, and the corresponding control equipment and interconnecting cabling within the train, are required to be OPERABLE. The equipment associated with each train of DC Sources is shown in Table B 3.8.4-1.

In the case where the requirements of LCO 3.8.10 call for portions of a second train of the distribution subsystems to be OPERABLE (e.g., to support two trains of RHR, two trains of CREFS, or instrumentation such as source range indication, containment ventilation isolation actuation, and/or CREFS actuation), the associated required DC bus(es) are OPERABLE if energized to the proper voltage from either:

- an OPERABLE DC Source, in accordance with LCO 3.8.5, or
- the associated charger(s) using the corresponding control equipment and interconnecting cabling within the train, in accordance with LCO 3.8.10.

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BASES

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BASES

LCO (continued)

The above requirements ensure the availability of sufficient DC electrical power sources to operate the unit in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents).

APPLICABILITY

The DC electrical power sources required to be OPERABLE in MODES 5 and 6 provide assurance that:

a. Required features needed to mitigate a fuel handling accident are available;

b. Required features necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

c. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

The DC electrical power requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.4.

ACTIONS

A.1, A.2.1, A.2.2, A.2.3, and A.2.4

If two subsystems are required by LCO 3.8.10, the remaining subsystem with DC power available may be capable of supporting sufficient systems to allow continuation of CORE ALTERATIONS and fuel movement. By allowing the option to declare required features inoperable with the associated DC power source(s) inoperable, appropriate restrictions will be implemented in accordance with the affected required features LCO ACTIONS. In many instances, this option may involve undesired administrative efforts. Therefore, the allowance for sufficiently conservative actions is made (i.e., to suspend CORE ALTERATIONS, movement of irradiated fuel assemblies, and operations involving positive reactivity additions). The Required Action to suspend positive reactivity additions does not preclude actions to

(continued)

BASES

BACKGROUND (continued)

four channels. Each inverter is independently connected to its respective instrument distribution panel so that the loss of an inverter cannot affect more than one of the six distribution panels. Therefore no single failure in the instrumentation and control power supply system or its associated power supplies can cause a loss of power to more than one of the redundant loads.

Specific details on inverters and their operating characteristics are found in the FSAR, Chapter 8 (Ref. 1).

The inverters and associated channels, AC vital buses, and DC panels are shown below:

CHANNEL	INVERTER	AC VITAL BUS	DC PANEL
Channel I	1/2AD1I1	1/2AY1A	1/2AD1
Channel I	1/2AD1I11	1/2AY2A	1/2AD1
Channel II	1/2BD1I2	1/2BY1B	1/2BD1
Channel II	1/2BD1I12	1/2BY2B	1/2BD1
Channel III	1/2CD1I3	1/2CY1A	1/2CD1
Channel IV	1/2DD1I4	1/2DY1B	1/2DD1

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the FSAR, Chapter 6 (Ref. 2) and Chapter 15 (Ref. 3), assume Engineered Safety Feature systems are OPERABLE. The inverters are designed to provide the required capacity, capability, redundancy, and reliability to ensure the availability of necessary power to the RPS and ESFAS instrumentation and controls so that the fuel, Reactor Coolant System, and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for Section 3.2, Power Distribution Limits; Section 3.4, Reactor Coolant System (RCS); and Section 3.6, Containment Systems.

The OPERABILITY of the inverters is consistent with the initial assumptions of the accident analyses and is based on meeting the design basis of the unit. This includes maintaining required AC vital buses OPERABLE during accident conditions in the event of:

(continued)

BASES (continued)

LCO

The inverters ensure the availability of electrical power for the instrumentation for systems required to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence or a postulated DBA. Per LCO 3.8.10, "Distribution Systems — Shutdown," the necessary portions of the necessary AC vital bus electrical power distribution subsystems shall be OPERABLE to support equipment required to be OPERABLE. At a minimum, at least one train of AC vital bus electrical power subsystems energized from the associated inverters connected to the respective DC bus is required to be OPERABLE.

In the case where the requirements of LCO 3.8.10 call for portions of a second train of the distribution subsystems to be OPERABLE (e.g., to support two trains of RHR, two trains of CREFS, or instrumentation such as source range indication, containment ventilation isolation actuation, and/or CREFS actuation), the required AC vital bus electrical power distribution subsystems may be energized from the associated inverters with the inverters connected to the respective bus, in accordance with LCO 3.8.8, or the Class 1E regulated transformer, in accordance with LCO 3.8.10. This ensures the availability of sufficient inverter power sources to operate the unit in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents).

APPLICABILITY

The inverters required to be OPERABLE in MODES 5 and 6 provide assurance that:

a. Systems needed to mitigate a fuel handling accident are available;

b. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

c. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

Inverter requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.7.

(continued)

BASES (continued)

ACTIONS

A.1, A.2.1, A.2.2, A.2.3, and A.2.4

If two trains are required by LCO 3.8.10, "Distribution Systems — Shutdown," the remaining OPERABLE Inverters may be capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS, fuel movement, and operations with a potential for positive reactivity additions. By the allowance of the option to declare required features inoperable with the associated inverter(s) inoperable, appropriate restrictions will be implemented in accordance with the affected required features LCOs'

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BASES (continued)

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Table B 3.8.9-1 (page 1 of 1)
AC and DC Electrical Power Distribution Systems

TYPE	VOLTAGE	TRAIN A*	TRAIN B*
AC safety buses	4160 V	Switchgear ESF Bus 1/2AA02	Switchgear ESF Bus 1/2BA03
	480 V	Switchgear 1/2AB04 1/2AB05 1/2AB15	Switchgear 1/2BB06 1/2BB07 1/2BB16
	480 V	Motor Control Centers 1/2ABE, 1/2ABA, 1/2ABC, 1/2ABF, 1/2ABB, 1/2ABD	Motor Control Centers 1/2BBE, 1/2BBA, 1/2BBC, 1/2BBF, 1/2BBB, 1/2BBD
DC buses	125 V	Switchgear 1/2AD1 1/2CD1	Switchgear 1/2BD1 1/2DD1
	125 V	Motor Control Centers 1/2AD1M, 1/2CD1M	Motor Control Centers 1/2BD1M
	125 V	Distribution Panels 1/2AD11, 1/2AD12, 1/2CD11	Distribution Panels 1/2BD11, 1/2BD12, 12DD11
AC vital buses	120 V	Distribution Panels	Distribution Panels
		Channel I 1/2AY1A, 1/2AY2A	Channel II 1/2BY1B, 1/2BY2B
		Channel III 1/2CY1A	Channel IV 1/2DY1B
		Associated Regulating Transformers**	Associated Regulating Transformers**

* Each train of the AC and DC electrical power distribution systems is a subsystem.

** A regulating transformer is a component of the Electrical Power Distribution Systems only when it is in service providing power to a 120 VAC vital bus.

BASES

LCO

Various combinations of subsystems, equipment, and components are required OPERABLE by other LCOs, depending on the specific plant condition. Implicit in those requirements is the required OPERABILITY of necessary support required features. This LCO explicitly requires energization of the portions of the electrical distribution system necessary to support OPERABILITY of required systems, equipment, and components — all specifically addressed in each LCO.

The necessary portions of the AC electrical power distribution subsystems are considered OPERABLE if they are energized to their proper voltages.

The necessary portions of the DC electrical power subsystems are considered OPERABLE if the following criteria are satisfied:

- At least one train of the necessary portions of DC electrical subsystems is energized to the proper voltage by an OPERABLE train of DC sources in accordance with LCO 3.8.5, "DC Sources," and

- In the case where portions of a second train of the DC electrical subsystems are required OPERABLE (to support two trains of RHR, two trains of CREFS, or instrumentation such as source range indication, containment ventilation isolation actuation, and/or CREFS actuation), the required portions of DC electrical subsystems are OPERABLE when energized to the proper voltage from either:

    - an OPERABLE DC source in accordance with LCO 3.8.5, or

    - the associated charger using the corresponding control equipment and interconnecting cabling within the train. In some cases where there is an increased potential for the addition or removal of loads larger than breaker control power ("larger loads"), as provided in plant administrative controls, both the associated battery and associated charger are required to

(continued)

BASES

LCO (continued)

support the second train to ensure stability of the required 125 VDC bus. The addition or removal of larger loads on the DC bus or on a supported AC vital bus involves a higher potential of DC bus transients when there is only the charger tied to the bus.

The necessary portions of the AC vital bus subsystems are considered OPERABLE if the following criteria are satisfied:

- At least one train of the necessary portions of AC vital bus subsystems is energized to the proper voltage by OPERABLE inverters connected to the respective DC bus, in accordance with LCO 3.8.8, "Inverters — Shutdown," and
- In the case where portions of a second train of AC vital bus subsystems are required OPERABLE (to support two trains of RHR, two trains of CREFS, or instrumentation such as source range indication, containment ventilation isolation actuation, and/or CREFS actuation), the required portions of AC vital bus subsystems are OPERABLE when energized to the proper voltage from either:
  - an OPERABLE inverter in accordance with LCO 3.8.8, or
  - the associated Class 1E regulating transformer.

Maintaining these portions of the distribution system energized ensures the availability of sufficient power to operate the unit in a safe manner to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents).

APPLICABILITY

The AC and DC electrical power distribution subsystems required to be OPERABLE in MODES 5 and 6 provide assurance that:

a. Systems needed to mitigate a fuel handling accident are available;

(continued)

BASES

APPLICABILITY (continued)

b. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

c. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition and refueling condition.

The AC, DC, and AC vital bus electrical power distribution subsystems requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.9.

ACTIONS

A.1, A.2.1, A.2.2, A.2.3, A.2.4, and A.2.5

Although redundant required features may require redundant trains of electrical power distribution subsystems to be OPERABLE, one OPERABLE distribution subsystem train may be

(continued)

BASES

BACKGROUND (continued)

In MODE 6, the 24 inch main or shutdown purge and exhaust valves are used to exchange large volumes of containment air to support refueling operations or other maintenance activities. During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment any open 24 inch valves are capable of being automatically closed by the Containment Ventilation Isolation signal (LCO 3.3.6). The 14 inch mini-purge and exhaust valves, though typically not opened during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, if opened are also capable of being closed automatically by the Containment Ventilation Isolation signal.

The other containment penetrations that provide direct access from containment atmosphere to outside atmosphere must be isolated on at least one side. Isolation may be achieved by a closed automatic isolation valve, a manual isolation valve, blind flange, or equivalent. Equivalent isolation methods allowed under the provisions of 10 CFR 50.59 may include use of a material that can provide a temporary, atmospheric pressure, ventilation barrier for the other containment penetrations during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment (Ref. 1).

APPLICABLE SAFETY ANALYSES

During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, the most severe radiological consequences result from a fuel handling accident. The fuel handling accident is a postulated event that involves damage to irradiated fuel (Ref. 2). Fuel handling accidents, analyzed in Reference 3, include dropping a single irradiated fuel assembly onto another irradiated fuel assembly.

To support the plant configuration of both air lock doors open, it was assumed in FSAR calculations for dose analysis that the designated individual for closure of the air lock would have the air lock closed within 15 minutes of the fuel handling accident. The 15 minute duration was chosen as the limit for the response capability for the person who is designated for closing the air lock door. The NRC

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

acceptance of this specification was based on doses for a 2 hour release as well as a licensee commitment for a person designated to close the door quickly.

Also, the requirements of LCO 3.9.7, "Refueling Cavity Water Level," and the minimum decay time of 100 hours prior to CORE ALTERATIONS ensure that the release of fission product radioactivity, subsequent to a fuel handling accident, results in doses that are well within the guideline values specified in 10 CFR 100. Standard Review Plan, Section 15.7.4, Rev. 1 (Ref. 3), defines "well within" 10 CFR 100 to be 25% or less of the 10 CFR 100 values. The acceptance limits for offsite radiation exposure will be 25% of 10 CFR 100 values or the NRC staff approved licensing basis (e.g., a specified fraction of 10 CFR 100 limits). Containment penetrations satisfy Criterion 3 of the NRC Policy Statement.

LCO

This LCO limits the consequences of a fuel handling accident in containment by limiting the potential escape paths for fission product radioactivity released within containment. The LCO requires the equipment door, one door in the emergency air lock, and any penetration providing direct access from the containment atmosphere to the outside atmosphere to be closed except for the OPERABLE containment ventilation penetrations and the personnel air lock door with a designated individual assigned to close it. For the OPERABLE containment ventilation penetrations, this LCO ensures that these penetrations are isolable by the Containment Ventilation Isolation System. The OPERABILITY requirements for LCO 3.3.6, Containment Ventilation Isolation Instrumentation ensure that the automatic purge supply and exhaust valve closure times specified in the FSAR can be achieved and, therefore, meet the assumptions used in the safety analysis to ensure that releases through the valves are terminated, such that radiological doses are within the acceptance limit.

Item b of this LCO includes requirements for both the emergency air lock and the personnel air lock. The emergency air lock is required to be isolated by at least one air lock door at all times when Specification 3.9.4 is applicable.

(continued)

BASES

LCO (continued)

The personnel air lock is required by Item b of this LCO to be isolable by at least one air lock door. Both containment personnel air lock doors may be open during movement of irradiated fuel in the containment and during CORE ALTERATIONS provided one air lock door is isolable. The personnel air lock is isolable when the following criteria are satisfied:

1. one personnel air lock door is OPERABLE,
2. at least 23 feet of water shall be maintained over the top of the reactor vessel flange in accordance with Specification 3.9.7,
3. a designated individual is available to close the door.

OPERABILITY of a containment personnel air lock door requires that the door seal protectors are easily removed, that no cables or hoses are being run through the air lock, and that the air lock door is capable of being quickly closed. The requirement that the plant maintain 23 feet of water above the reactor vessel flange ensures there is sufficient time to close the personnel air lock following a loss of shutdown cooling before boiling occurs. This requirement for the personnel air lock may be satisfied by maintaining at least one air lock door closed.

APPLICABILITY

The containment penetration requirements are applicable during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment because this is when there is a potential for a fuel handling accident. In MODES 1, 2, 3, and 4, containment penetration requirements are addressed by LCO 3.6.1, "Containment." In MODES 5 and 6, when CORE ALTERATIONS or movement of irradiated fuel assemblies within containment are not being conducted, the potential for a fuel handling accident does not exist. Therefore, under these conditions no requirements are placed on containment penetration status.

(continued)

BASES (continued)

ACTIONS

A.1 and A.2

If the containment equipment hatch, air locks, or any containment penetration that provides direct access from the containment atmosphere to the outside atmosphere is not in the required status, the unit must be placed in a condition where the isolation function is not needed. This is accomplished by immediately suspending CORE ALTERATIONS and movement of irradiated fuel assemblies within containment. Performance of these actions shall not preclude completion of movement of a component to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.9.4.1

This Surveillance demonstrates that each of the containment penetrations required to be in its closed position is in that position. The Surveillance on the open containment ventilation isolation valves will demonstrate that the valves are not blocked from closing. Also the Surveillance will demonstrate that each valve operator has motive power, which will ensure that each valve is capable of being closed by an OPERABLE automatic containment ventilation isolation signal.

The Surveillance is performed every 7 days during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment. The Surveillance interval is selected to be commensurate with the normal duration of time to complete fuel handling operations. Including a surveillance before the start of refueling operations will provide two or three surveillance verifications during the applicable period for this LCO. As such, this Surveillance ensures that a postulated fuel handling accident that releases fission product radioactivity within the containment will not result in a release of fission product radioactivity to the environment.

SR 3.9.4.2

This Surveillance demonstrates that each containment ventilation isolation valve actuates to its isolation position on manual initiation or on an actual or simulated high radiation signal. The 18 month Frequency maintains

(continued)

BASES

SURVEILLANCE REQUIREMENTS

3.9.4.2 (continued)

consistency with other similar ESFAS instrumentation and valve testing requirements. In LCO 3.3.6, "Containment Ventilation Isolation Instrumentation," additional containment ventilation isolation system surveillance requirements include a 12 hour Channel Check, a 31 day Actuation Logic Test and Master Relay Test, a 92 day COT and Slave Relay Test, an 18 month TADOT, and Channel Calibration and a response time test every 18 months on a STAGGERED TEST BASIS. Also, SR 3.6.3.5 demonstrates that the isolation time of each valve is in accordance with the Inservice Testing Program requirements. These Surveillances performed during MODE 6 will ensure that the valves are capable of closing after a postulated fuel handling accident to limit a release of fission product radioactivity from the containment.

REFERENCES

1. GPU Nuclear Safety Evaluation SE-0002000-001, Rev. 0, May 20, 1988.

2. FSAR, Subsection 15.7.4.

3. NUREG-0800, Section 15.7.4, Rev. 1, July 1981.