ENCLOSURE

This enclosure contains the clean typed pages for those Technical Specification and Technical Requirements changes beyond the issue of the Final Draft Technical Specifications. In general, all of these changes have been seen in draft form by the NRC and have been accepted for future inclusion in the Technical Specifications or Technical Requirements. In addition, the following Technical Specifications or Technical Requirements have open issues that are being negotiated with the NRC for inclusion in the Technical Specifications or Technical Requirements.

Technical Specification 3.3.4 on Remote Shutdown (Change Package 95-018)

Technical Specification SR 3.3.2.10 and SR 3.7.5.2 Notes on testing of the Turbine Driven Auxiliary Feedwater Pump (Change Package 95-038)

Technical Specification 3.8.1 on clarification of diesel generator testing requirements (Change Package 95-046)

Technical Specification 3.8.4 and 3.8.6 on clarification of basis for connection resistance limits (Change Package 95-053)

Technical Specification 3.8.4 on clarification of requirements for performance of a modified performance discharge test on the batteries (Change Package 95-055)

Technical Requirement 3.7.3 on clarification of basis and exceptions to testing requirements for snubbers (Change Package 95-067)*

Technical Specification 3.6.3 on clarification of containment isolation valves which may be tested at the frequency stated in SR 3.6.3.3 (Change Package 95-069)*

Technical Specification 3.7.5 on clarification of Auxiliary Feedwater System Design criteria (Change Package 95-070)*

Technical Specification 3.4.12 on clarification of acceptable alternative means for preventing inadvertent injection to the RCS (Change Package 95-071)*

Technical Specification 3.4.14 on clarification of how to include Pressure Isolation Valve leakage in the leakage addressed in 3.4.13 (Change Package 95-072)*

*Change packages have been or will be provided for preliminary review by the NRC.

(continued)

3.0 LCO APPLICABILITY

LCO 3.0.6 (continued)	When a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.
LCO 3.0.7	Test Exception LCOs 3.1.9 and 3.1.10 allow specified Technical Specification (TS) requirements to be changed to permit performance of special tests and operations. Unless otherwise specified, all other TS requirements remain unchanged. Compliance with Test Exception LCOs is optional. When a Test Exception LCO is desired to be met but is not met, the ACTIONS of the Test Exception LCO shall be met. When a Test Exception LCO is not desired to be met, entry into a MODE or other specified condition in the Applicability shall be made in accordance with the other applicable Specifications.

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.2.4.1	----------NOTE---------- 1. With input from one power range neutron flux channel inoperable and THERMAL POWER < 75% RTP, the remaining three power range channels can be used for calculating QPTR. 2. SR 3.2.4.2 may be performed in lieu of this Surveillance if adequate power range neutron flux channel inputs are not OPERABLE. ---------- Verify QPTR is within limit by calculation.	7 days AND Once within 12 hours and every 12 hours thereafter with the QPTR alarm inoperable
SR 3.2.4.2	----------NOTE---------- Only required to be performed if input from one or more power range neutron flux channels are inoperable with THERMAL POWER ≥ 75% RTP. ---------- Verify QPTR is within limit using the movable incore detectors.	Once within 12 hours AND 12 hours thereafter

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
M. One channel inoperable.	----------NOTE---------- The inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels. ----------	
	M.1 Place channel in trip.	6 hours
	OR	
	M.2 Reduce THERMAL POWER to < P-7.	12 hours
N. One Reactor Coolant Flow--Low (single loop) channel inoperable.	----------NOTE---------- One channel may be bypassed for up to 4 hours for surveillance testing. ----------	
	N.1 Place channel in trip.	6 hours
	OR	
	N.2 Reduce THERMAL POWER to < P-8.	10 hours

(continued)

Table 3.3.1-1 (page 3 of 9)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
9. Pressurizer Water Level – High	1[f]	3	X	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10	≤ 92.7% span	≤ 92% span
10. Reactor Coolant Flow – Low						
a. Single Loop	1[g]	3 per loop	N	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.15	≥ 89.6% flow	≥ 90% flow
b. Two Loops	1[h]	3 per loop	X	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.15	≥ 89.6% flow	≥ 90% flow
11. Undervoltage RCPs	1[f]	1 per bus	M	SR 3.3.1.9 SR 3.3.1.10 SR 3.3.1.15	≥ 4734 V	≥ 4830 V
12. Underfrequency RCPs	1[f]	1 per bus	M	SR 3.3.1.9 SR 3.3.1.10 SR 3.3.1.15	≥ 56.9 Hz	≥ 57.5 Hz

(continued)

(f) Above the P-7 (Low Power Reactor Trips Block) interlock.

(g) Above the P-8 (Power Range Neutron Flux) interlock.

(h) Above the P-7 (Low Power Reactor Trips Block) interlock and below the P-8 (Power Range Neutron Flux) interlock.

Table 3.3.2-1 (page 4 of 7)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
6. Auxiliary Feedwater						
a. Automatic Actuation Logic and Actuation Relays	1,2,3	2 trains	G	SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.5	NA	NA
b. SG Water Level-Low Low	1,2,3	3 per SG	M	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.9 SR 3.3.2.10	≥ 16.4%	≥ 17.0%
Coincident with:						
1) Vessel ΔT equivalent to power ≤ 50% RTP	1,2	3	N	SR 3.3.2.4 SR 3.3.2.9	Vessel ΔT variable input ≤ 52.7% RTP	Vessel ΔT variable input ≤ 50% RTP
With a time delay (Ts) if one S/G is affected					≤ 1.01 Ts (Note 1, (Page 3.3-40)	≤ Ts (Note 1, (Page 3.3-40)
or						
A time delay (Tm) if two or more S/G's are affected					≤ 1.01 Tm (Note 1, (Page 3.3-40)	≤ Tm (Note 1, (Page 3.3-40)
OR						
2) Vessel ΔT equivalent to power > 50% RTP with no time delay (Ts and Tm = 0)	1,2	3	N	SR 3.3.2.4 SR 3.3.2.9	Vessel ΔT variable input ≤ 52.7% RTP	Vessel ΔT variable input ≤ 50% RTP

(continued)

Table 3.3.5-1 (page 1 of 1)
LOP DG Start Instrumentation

FUNCTION	REQUIRED CHANNELS PER BUS	SURVEILLANCE REQUIREMENTS	TRIP SETPOINT	ALLOWABLE VALUE
1. 6.9 kV Emergency Bus Undervoltage (Loss of Voltage)				
a. Bus Undervoltage	3	SR 3.3.5.1 SR 3.3.5.2	≥ 5994 V and ≤ 6006 V	≥ 5967.6 V
b. Time Delay	2	SR 3.3.5.3	≥ 0.73 sec and ≤ 0.77 sec	≥ 0.58 sec and ≤ 0.94 sec
2. 6.9 kV Emergency Bus Undervoltage (Degraded Voltage)				
a. Bus Undervoltage	3	SR 3.3.5.1 SR 3.3.5.2	≥ 6593.4 V and ≤ 6606.6 V	≥ 6570 V
b. Time Delay	2	SR 3.3.5.3	≥ 5.84 sec and ≤ 6.16 sec	≥ 5.7 sec and ≤ 6.3 sec
3. Diesel Generator Start	2	SR 3.3.5.1 SR 3.3.5.2	≥ 4733.4 V and ≤ 4926.6 V with an internal time delay of ≥ 0.46 sec and ≤ 0.54 sec	≥ 2295.6 V with an internal time delay of 0.56 sec at zero volts.
4. Load Shed	4	SR 3.3.5.1 SR 3.3.5.2	≥ 4733.4 V and ≤ 4926.6 V with an internal time delay of ≥ 2.79 sec and ≤ 3.21 sec	≥ 2295.6 V with an internal time delay of ≤ 3.3 sec at zero volts.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

LCO 3.4.1 RCS DNB parameters for pressurizer pressure, RCS average temperature, and RCS total flow rate shall be within the limits specified below:

a. Pressurizer pressure ≥ 2214 psig;

b. RCS average temperature ≤ 593.5°F; and

c. RCS total flow rate ≥ 397,000 gpm (process computer) or 398,000 gpm (control board indication).

APPLICABILITY: MODE 1.

----------NOTE----------
Pressurizer pressure limit does not apply during:

a. THERMAL POWER ramp > 5% RTP per minute; or

b. THERMAL POWER step > 10% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more RCS DNB parameters not within limits.	A.1 Restore RCS DNB parameter(s) to within limit.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.1.1	Verify pressurizer pressure is ≥ 2214 psig.	12 hours
SR 3.4.1.2	Verify RCS average temperature is ≤ 593.5°F.	12 hours
SR 3.4.1.3	Verify RCS total flow rate is ≥ 397,000 gpm (process computer) or 398,000 gpm (control board indication).	12 hours
SR 3.4.1.4	----------NOTE---------- Not required to be performed until 24 hours after ≥ 90% RTP. ---------- Verify by precision heat balance that RCS total flow rate is ≥ 397,000 gpm.	18 months

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.15.1	Verify annulus negative pressure is equal to or more negative than -5 inches water gauge with respect to the atmosphere.	12 hours
SR 3.6.15.2	Verify the door in each access opening is closed, except when the access opening is being used for normal transient entry and exit.	31 days
SR 3.6.15.3	Verify shield building structural integrity by performing a visual inspection of the exposed interior and exterior surfaces of the shield building.	During shutdown for SR 3.6.1.1 Type A tests
SR 3.6.15.4	Verify each Emergency Gas Treatment System train with final flow $\geq$ 3600 and $\leq$ 4400 cfm produces an annulus pressure equal to or more negative than -0.61 inch water gauge at elevation 783 with respect to the atmosphere and with an inleakage of $\leq$ 250 cfm.	18 months on a STAGGERED TEST BASIS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.10.3	Verify each CREVS train actuates on an actual or simulated actuation signal.	18 months
SR 3.7.10.4	Verify one CREVS train can maintain a positive pressure of ≥ 0.125 inches water gauge, relative to the outside atmosphere and adjacent areas during the pressurization mode of operation at a makeup flow rate of ≤ 711 cfm and a recirculation flow rate ≥ 2960 and ≤ 3618 cfm.	18 months on a STAGGERED TEST BASIS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.4.14	----------NOTE---------- This Surveillance is normally not performed in MODE 1, 2, 3, or 4. However, credit may be taken for unplanned events that satisfy this SR. ---------- Verify battery capacity is ≥ 80% of the manufacturer's rating when subjected to a performance discharge test or a modified performance discharge test.	60 months AND 12 months when battery shows degradation or has reached 85% of expected life with capacity < 100% of manufaturer's rating AND 24 months when battery has reached 85% of the expected life with capacity ≥ 100% of manufacturer's rating

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.5.1	----------NOTE---------- The following SRs are not required to be performed: SR 3.8.4.11, SR 3.8.4.12, SR 3.8.4.13, and SR 3.8.4.14. ---------- For DC sources required to be OPERABLE, the following SRs are applicable: SR 3.8.4.1 SR 3.8.4.6 SR 3.8.4.11 SR 3.8.4.2 SR 3.8.4.7 SR 3.8.4.12 SR 3.8.4.3 SR 3.8.4.8 SR 3.8.4.13 SR 3.8.4.4 SR 3.8.4.9 SR 3.8.4.14 SR 3.8.4.5 SR 3.8.4.10	In accordance with applicable SRs

5.2 Organization

5.2.2 Unit Staff (continued)

of overtime to be used, or during extended periods of shutdown for refueling, major maintenance, or major plant modification, on a temporary basis the following guidelines shall be followed:

1. An individual should not be permitted to work more than 16 hours straight, excluding shift turnover time;

2. An individual should not be permitted to work more than 16 hours in any 24 hour period, nor more than 24 hours in any 48 hour period, nor more than 72 hours in any 7 day period, all excluding shift turnover time;

3. A break of at least 8 hours should be allowed between work periods, including shift turnover time;
4. Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift.

Any deviation from the above guidelines shall be authorized in advance by the Plant Manager or his designee, in accordance with approved administrative procedures, or by higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation.

Controls shall be included in the procedures such that individual overtime shall be reviewed monthly by the Plant Manager or his designee to ensure that excessive hours have not been assigned. Routine deviation from the above guidelines is not authorized.

f. The Operations Manager shall hold or have held an SRO license on a similar unit. Either the Operations Manager or Operations Superintendent shall have a valid SRO license on this unit.

g. The Shift Technical Advisor (STA) shall provide advisory technical support to the Shift Operations Supervisor (SOS) in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. In addition, the STA shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on shift (Generic Letter 86-04 dated 02/13/86).

5.7 Procedures, Programs, and Manuals

5.7.2.12 Steam Generator (SG) Tube Surveillance Program (continued)

e. Inspection frequency - The above required inservice inspections of the SG tubes shall be performed at the following frequencies:

1. The first inservice inspection shall be performed after 6 effective full power months but within 24 calendar months of initial criticality. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections, not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months;

2. If the results of the inservice inspection of a SG conducted in accordance with Table 5.7.2.12-1 at 40-month intervals fall in Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 5.7.2.12.e.1; the interval may then be extended to a maximum of once per 40 months; and

3. Additional, unscheduled inservice inspections shall be performed on each SG in accordance with the first sample inspection specified in Table 5.7.2.12-1 during the shutdown subsequent to any of the following conditions:

a) Primary-to-secondary tube leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.13, or

(continued)

5.7 Procedures, Programs, and Manuals

TABLE 5.7.2.12-1

STEAM GENERATOR TUBE INSPECTION
SUPPLEMENTAL SAMPLING REQUIREMENTS

1st Sample Inspection			2nd Sample Inspection		3rd Sample Inspection	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S tubes per SG	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug defective tubes and inspect an additional 2S tubes in this SG.	C-1	None	N/A	N/A
			C-2	Plug defective tubes and inspect an additional 4S tubes in this SG.	C-1	N/A
					C-2	Plug defective tubes.
					C-3	Perform action for C-3 result of first sample.
			C-3	Perform action for C-3 result of first sample.	N/A	N/A
	C-3	Inspect all tubes in this SG, plug defective tubes and inspect 2S tubes in each other SG. Notification to NRC pursuant to 10CFR50.72	All other SGs C-1	None	N/A	N/A
			Some SGs C-2 but no other is C-3	Perform action for C-2 result of second sample.	N/A	N/A
			Additional SG is C-3	Inspect all tubes in each SG and plug defective tubes. Notification to NRC pursuant to 10CFR50.72.	N/A	N/A

S = 3 N/n % Where N is the number of SGs in the unit and n is the number of S.G.s inspected during an inspection.

(continued)

5.7 Procedures, Programs, and Manuals

TABLE 5.7.2.12-2

MINIMUM NUMBER OF STEAM GENERATORS TO BE INSPECTED DURING INSERVICE INSPECTION

Preservice Inspection	All
First Inservice Inspection	Two
Second and Subsequent Inservice Inspections	One[1]

1. The inservice inspection may be limited to one SG on a rotating schedule encompassing 3 N % of the tubes (where N is the number of SGs in the plant) if the results of the first or previous inspections indicate that all SGs are performing in a like manner. Note that under some circumstances, the operating conditions in one or more SGs may be found to be more severe than those in other SGs. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.

   One of the other two SGs not inspected during the first inservice inspections shall be inspected during the second inspection period and the remaining SG shall be inspected during the third inspection period. The fourth and subsequent inspections shall follow the instructions described above.

5.7.2.13 Secondary Water Chemistry Program

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation and low pressure turbine disc stress corrosion cracking. The program shall include:

a. Identification of a sampling schedule for the critical variables and control points for these variables;

b. Identification of the procedures used to measure the values of the critical variables;

c. Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in leakage;

(continued)

5.9 Reporting Requirements

5.9.8 PAMS Report

When a Report is required by Condition B or G of LCO 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.9.9 SG Tube Inspection Report

Following each inservice inspection of steam generator (SG) tubes, in accordance with the SG Tube Surveillance Program, the number of tubes plugged and tubes sleeved in each SG shall be reported to the NRC within 15 days.

The complete results of the SG tube inservice inspection shall be submitted to the NRC within 12 months following the completion of the inspection. The report shall include:

1. Number and extent of tubes inspected,

2. Location and percent of wall-thickness penetration for each indication of an imperfection, and

3. Identification of tubes plugged.

Results of SG tube inspections that fall into Category C-3 shall be reported to the NRC in accordance with 10 CFR 50.72. This report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

5.7 Procedures, Programs, and Manuals

5.7.2.7 Radioactive Effluent Controls Program

This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM;

b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to 10 times the concentration values in 10 CFR 20.1001-20.2402, Appendix B, Table 2, Column 2;

c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM;

d. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from each unit to unrestricted areas, conforming to 10 CFR 50, Appendix I;

e. Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days;

f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2% of the guidelines for the annual dose or dose commitment, conforming to 10 CFR 50, Appendix I;

(continued)

BASES

LCO 3.0.6 (continued)

remedial actions, or compensatory actions may be identified as a result of the support system inoperability and corresponding exception to entering supported system Conditions and Required Actions. The SFDP implements the requirements of LCO 3.0.6.

Cross train checks to identify a loss of safety function for those support systems that support multiple and redundant safety systems are required. The cross train check verifies that the supported systems of the redundant OPERABLE support system are OPERABLE, thereby ensuring safety function is retained. If this evaluation determines that a loss of safety function exists, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

LCO 3.0.7

There are certain special tests and operations required to be performed at various times over the life of the plant. These special tests and operations are necessary to demonstrate select plant performance characteristics, to perform special maintenance activities, and to perform special evolutions.

Test Exception LCOs 3.1.9 and 3.1.10 allow specified Technical Specification (TS) requirements to be changed to permit performances of these special tests and operations, which otherwise could not be performed if required to comply with the requirements of these TS. Unless otherwise specified, all the other TS requirements remain unchanged. This will ensure all appropriate requirements of the MODE or other specified condition not directly associated with or required to be changed to perform the special test or operation will remain in effect.

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

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

13. Steam Generator Water Level-Low Low (continued)

(T_M) for the affected protection set, through the Man-Machine Interface. Failure of the vessel ΔT channel input (failure of more than one T_H RTD or failure of both T_C RTDs) affects the TTD calculation for a protection set. This results in the requirement that the operator adjust the threshold power level for zero seconds time delay from 50% RTP to 0% RTP, through the Man Machine Interface.

The LCO requires three channels of SG Water Level – Low Low per SG to be OPERABLE. This function initiates a reactor trip and the ESFAS function auxiliary feedwater pump start. The reactor trip feature is required to be OPERABLE in MODES 1 and 2 and the auxiliary feedwater pump start feature is required to be OPERABLE in MODES 1, 2, and 3.

In MODE 3, OPERABILITY of loop ΔT input to TTD is not required because MODE 3 ΔT = 0 (by definition). The Eagle-21™ code does not allow anything less than 0. The value of ΔT is low-limited to 0.0 prior to use in the calculation of the single and multiple trip time delays.

For MODES 1, 2, and 3, channel check surveillance testing on RCS loop ΔT input to TTD is not required. There are no provisions made to verify the RCS loop ΔT for the SG Level TTD Function. The power level can only be verified by connecting the Eagle-21™ Man-Machine Interface terminal and viewing the Dynamic Information for this channel. The Eagle-21™ system uses a redundant sensor algorithm for the hot leg and cold leg inputs, and will alert the operator if a failure occurs with the sensor or input signal conditioning.

The coefficients (A, B, C, D, E, F, G, and H) shown in the equation of Note 3 represent conservative values for the calculation of the time delay (i.e., the values given are 99% of the values used for the safety analyses). For the Eagle-21™ System, these coefficients are displayed (via the Man-Machine Interface) as A, B, C and D for the single request time delay, and E, F, G and H for the multiple request time delay.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

13. Steam Generator Water Level-Low Low (continued)

In MODE 1 or 2, when the reactor is critical, the SG Water Level – Low Low trip must be OPERABLE. In MODES 1, 2, and 3 the normal source of water for the SGs is the Main Feedwater (MFW) System (not safety related). The AFW System is the safety related backup source of water to ensure that the SGs remain the heat sink for the reactor in these MODES. The ESFAS function of the SG Water Level-Low Low trip must be OPERABLE in MODES 1, 2, and 3. In MODES 3, 4, 5, and 6, the SG Water Level – Low Low trip Function does not have to be OPERABLE because the reactor is not operating or even critical. Decay heat removal is accomplished by the AFW or MFW System in MODE 3 and by the Residual Heat Removal (RHR) System in MODE 4, 5, or 6.

14. Turbine Trip

a. Turbine Trip-Low Fluid Oil Pressure

The Turbine Trip – Low Fluid Oil Pressure trip Function anticipates the loss of heat removal capabilities of the secondary system following a turbine trip. This trip Function acts to minimize the pressure/temperature transient on the reactor. Any turbine trip from a power level below the P-9 setpoint, approximately 50% power, will not actuate a reactor trip. Three pressure switches monitor the control oil pressure in the Turbine Electrohydraulic Control System. A low pressure condition sensed by two-out-of-three pressure switches will actuate a reactor trip. These pressure switches do not provide any input to the control system. The unit is designed to withstand a complete loss of load and not sustain core damage or challenge the RCS pressure limitations. Core protection is provided by the Pressurizer Pressure – High trip Function and RCS integrity is ensured by the pressurizer safety valves.

The LCO requires three channels of Turbine Trip – Low Fluid Oil Pressure to be OPERABLE in MODE 1 above P-9.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.3

SR 3.3.1.3 compares the incore system to the NIS channel output every 31 EFPD. If the absolute difference is ≥ 3%, the NIS channel is still OPERABLE, but must be readjusted. If the NIS channel cannot be properly readjusted, the channel is declared inoperable. This Surveillance is performed to verify the f(ΔI) input to the Overtemperature ΔT Function.

Two Notes modify SR 3.3.1.3. Note 1 indicates that the excore NIS channel shall be adjusted if the absolute difference between the incore and excore AFD is ≥ 3%. Note 2 clarifies that the Surveillance is required only if reactor power is ≥ 15% RTP and that 96 hours is allowed for performing the first Surveillance after reaching 15% RTP. This surveillance is typically performed at 50% RTP to ensure the results of the evaluation are more accurate and the adjustments more reliable. Ninety-six (96) hours are allowed to ensure Xenon stability and allow for instrumentation alignments.

The Frequency of every 31 EFPD is adequate. It is based on unit operating experience, considering instrument reliability and operating history data for instrument drift. Also, the slow changes in neutron flux during the fuel cycle can be detected during this interval.

SR 3.3.1.4

SR 3.3.1.4 is the performance of a TADOT every 31 days on a STAGGERED TEST BASIS. This test shall verify OPERABILITY by actuation of the end devices.

The RTB test shall include separate verification of the undervoltage and shunt trip mechanisms. Independent verification of RTB undervoltage and shunt trip Function is not required for the bypass breakers. No capability is provided for performing such a test at power. The bypass breaker test shall include a local shunt trip. A Note has been added to indicate that this test must be performed on the bypass breaker prior to placing it in service.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.11 (continued)

the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed on the 18 month Frequency.

SR 3.3.1.12

SR 3.3.1.12 is the performance of a COT of RTS interlocks every 18 months.

The Frequency is based on the known reliability of the interlocks and the multichannel redundancy available, and has been shown to be acceptable through operating experience.

SR 3.3.1.13

SR 3.3.1.13 is the performance of a TADOT of the Manual Reactor Trip, Reactor Trip from Manual SI, and the Reactor Trip from Automatic SI Input from ESFAS. This TADOT is performed every 18 months. The test shall independently verify the OPERABILITY of the undervoltage and shunt trip mechanisms for these Reactor Trip Functions for the Reactor Trip Breakers. The test shall also verify OPERABILITY of the Reactor Trip Bypass Breakers for these functions. Independent verification of the Reactor Trip Bypass Breakers undervoltage and shunt trip mechanisms is not required.

The Frequency is based on the known reliability of the Functions and the multichannel redundancy available, and has been shown to be acceptable through operating experience.

The SR is modified by a Note that excludes verification of setpoints from the TADOT. The Functions affected have no setpoints associated with them.

SR 3.3.1.14

SR 3.3.1.14 is the performance of a TADOT of Turbine Trip Functions. This TADOT is as described in SR 3.3.1.4, except that this test is performed prior to reactor startup. A Note states that this Surveillance is not required if it has been performed within the previous 31 days. Verification

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

b. Auxiliary Feedwater-Steam Generator Water Level-Low Low (continued)

The Steam Generator Water Level Channel Trip Time Delay (TTD) creates additional operational margin when the unit needs it most, during power escalation from low power, by allowing the operator time to recover level when the primary side load is sufficiently small to allow such action. The TTD is based on continuous monitoring of primary side power through the use of vessel ΔT. Two time delays are calculated, based on the number of steam generators indicating less than the Low-Low Level channel Trip Setpoint per Note 1 of Table 3.3.2-1. The magnitude of the delays decreases with increasing primary side power level, up to 50% RTP. Above 50% RTP there are no time delays for the Low-Low Level channel trips.

The algorithm for the TTD, T_s and T_m, determines the trip delay as a function of power level (P) and four constants (A...D for T_s, E...H for T_m). An allowance for the accuracy of the Eagle-21™ time base is included in the determination of the magnitude of the constants. The magnitude of the accuracy allowance is 1%, i.e., the constant values were multiplied by 0.99 to account for this potential error.

In the event of a failure of a Steam Generator Water Level channel, the channel is placed in the trip condition as input to the Solid State Protection System and does not affect the TTD setpoint calculations for the remaining OPERABLE channels. It is then necessary for the operator to force the use of the shorter TTD time delay by adjustment of the single steam generator time delay calculation (T_S) to match the multiple steam generator time delay calculation (T_m) for the affected protection set, through the Man Machine Interface. Failure of the vessel ΔT channel input (failure of more than one T_H RTD or failure of both T_C RTDs) affects the TTD calculation for a protection set. This results in the requirement that the operator adjust the threshold power level for zero seconds time delay from 50% RTP to 0% RTP, through the Man Machine Interface.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

c. Auxiliary Feedwater-Safety Injection

An SI signal starts the motor driven and turbine driven AFW pumps. The AFW initiation functions are the same as the requirements for their SI function. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead, Function 1, SI, is referenced for all initiating functions and requirements.

d. Auxiliary Feedwater-Loss of Offsite Power

A loss of offsite power to the RCP buses will be accompanied by a loss of reactor coolant pumping power and the subsequent need for some method of decay heat removal. The loss of offsite power is detected by a voltage drop on each 6.9 kV shutdown board. Loss of power to either 6.9 kV shutdown board will start the turbine driven AFW pump to ensure that enough water is available to serve as the heat sink for reactor decay heat and sensible heat removal following the reactor trip.

Functions 6.a through 6.d (except the loop ΔT input to the trip time delay) must be OPERABLE in MODES 1, 2, and 3 to ensure that the SGs remain the heat sink for the reactor. SG Water Level – Low Low in any operating SG will cause the motor driven AFW pumps to start. The system is aligned so that upon a start of the pump, water immediately begins to flow to the SGs. SG Water Level – Low Low in any two operating SGs will cause the turbine driven pumps to start. These Functions do not have to be OPERABLE in MODES 5 and 6 because there is not enough heat being generated in the reactor to require the SGs as a heat sink. In MODE 4, AFW actuation does not need to be OPERABLE because either AFW or residual heat removal (RHR) will already be in operation to remove decay heat or sufficient time is available to manually place either system in operation.

(continued)

B 3.3 INSTRUMENTATION

B 3.3.5 Loss of Power (LOP) Diesel Generator (DG) Start Instrumentation

BASES

BACKGROUND

The DGs provide a source of emergency power when offsite power is either unavailable or is insufficiently stable to allow safe unit operation. Undervoltage protection will generate an LOP start if a loss of voltage or degraded voltage condition occurs in the switchyard. There are four LOP start signals, one for each 6.9 kV shutdown board.

Three degraded voltage relays (one per phase) are provided on each 6.9 kV Shutdown Board for detecting a sustained undervoltage condition. The relays are combined in a two-out-of-three logic configuration to generate a supply breaker trip signal if the voltage is below 96% for 6 seconds (nominal). Additionally, three undervoltage relays (one per phase) are provided on each 6.9 kV Shutdown Board for the purpose of detecting a loss of voltage condition. These relays are combined in a two-out-of-three logic to generate a supply breaker trip signal if the voltage is below 87% for 0.75 seconds (nominal).

Once the supply breakers have been opened, either one of two induction disk type relays, which have a voltage setpoint of 70% of 6.9 kV (nominal, decreasing) and an internal time delay of 0.5 seconds (nominal) at zero volts, will start the diesel generators. Four additional induction disk type relays, in a logic configuration of one-of-two taken twice which have a voltage setpoint of 70% of 6.9 kV (nominal, decreasing) and an internal time delay of 3 seconds (nominal), at zero volts, will initiate load shedding of the 6.9 kV shutdown boards and selected loads on the 480 V shutdown boards and close the 480 V shutdown boards' current limiting reactor bypass breaker. The LOP start actuation is described in FSAR Section 8.3, "Onsite (Standby) Power System" (Ref. 1).

Trip Setpoints and Allowable Values

The Trip Setpoints used in the relays and timers are based on the analytical limits presented in TVA calculations, References 3, 5, and 6. The selection of these Trip Setpoints is such that adequate protection is provided when all sensor and time delays are taken into account.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

result in meeting the DNBR criterion. This is the acceptance limit for the RCS DNB parameters. Changes to the unit that could impact these parameters must be assessed for their impact on the DNBR criteria. The transients analyzed for include loss of coolant flow events and dropped or stuck rod events. A key assumption for the analysis of these events is that the core power distribution is within the limits of LCO 3.1.7, "Control Bank Insertion Limits"; LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)"; and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)."

The pressurizer pressure limit of 2214 psig and the RCS average temperature limit of 593.5°F correspond to analytical limits of 2189 psig and 594.7°F used in the safety analyses, with allowance for measurement uncertainty.

The RCS DNB parameters satisfy Criterion 2 of the NRC Policy Statement.

LCO

This LCO specifies limits on the monitored process variables – pressurizer pressure, RCS average temperature, and RCS total flow rate – to ensure the core operates within the limits assumed in the safety analyses. Operating within these limits will result in meeting the DNBR criterion in the event of a DNB limited transient.

RCS total flow rate contains a measurement error of 1.6% (process computer) or 1.9% (control board indication) based on performing a precision heat balance and using the result to calibrate the RCS flow rate indicators. Potential fouling of the feedwater venturi, which might not be detected, could bias the result from the precision heat balance in a nonconservative manner. Therefore, a penalty of 0.1% for undetected fouling of the feedwater venturi raises the nominal flow measurement allowance to 1.7% (process computer) or 2.0% (control board indication).

Any fouling that might bias the flow rate measurement greater than 0.1% can be detected by monitoring and trending various plant performance parameters. If detected, either the effect of the fouling shall be quantified and compensated for in the RCS flow rate measurement or the venturi shall be cleaned to eliminate the fouling.

(continued)

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 totally enclosed pop type, spring loaded, self actuated valves 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 totally enclosed and self actuating, they are considered independent components. The relief capacity for each valve, 420,000 lb/hr, is based on postulated overpressure transient conditions resulting from a complete loss of steam flow to the turbine. This event results in the maximum surge rate into the pressurizer, which specifies the minimum relief capacity for 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, and 5; however, in MODE 4, MODE 5, and MODE 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 Overpressure Mitigation System (COMS)."

The upper and lower pressure limits are based on a ± 3% tolerance. The lift setting is for the ambient conditions associated with MODES 1, 2, and 3. This requires either that the valves be set hot or that a correlation between hot and cold settings be established.

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

(continued)

BASES

BACKGROUND (continued)

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, d, e, f, and g (above) to limit the pressure increase. Compliance with this LCO is consistent with the design bases and accident analyses assumptions.

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

LCO

The three pressurizer safety valves are set to open at the RCS design pressure (2485 psig), and within the specified tolerance, to avoid exceeding the maximum design pressure SL, to maintain accident analyses assumptions, and to comply with ASME requirements. The upper and lower

(continued)

BASES

LCO (continued)

pressure tolerance limits are based on a ± 3% tolerance. The limit protected by this Specification is the reactor coolant pressure boundary (RCPB) SL of 110% of design pressure. Inoperability of one or more valves could result in exceeding the SL if a transient were to occur. The consequences of exceeding the ASME pressure limit could include damage to one or more RCS components, increased leakage, or additional stress analysis being required prior to resumption of reactor operation.

APPLICABILITY

In MODES 1, 2, and 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 when all RCS cold leg temperatures are ≤ 350°F or in MODE 5 because COMS is provided. Overpressure protection is not required in MODE 6 with reactor vessel head detensioned.

The Note allows entry into MODE 3 with the lift settings outside the LCO limits. This permits testing and examination of the safety valves at high pressure and temperature near their normal operating range, but only after the valves have had a preliminary cold setting. The cold setting gives assurance that the valves are OPERABLE near their design condition. Only one valve at a time will be removed from service for testing. The 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.

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

(continued)

BASES

ACTIONS

A.1 (continued)

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. With any RCS cold leg temperatures at or below 350°F, overpressure protection is provided by the COMS 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 pressurizer safety valve setpoint is ± 3% for OPERABILITY, however, the valves are reset to ± 1% during the surveillance to allow for drift.

REFERENCES

1. ASME Boiler and Pressure Vessel Code, Section III, NB 7000, 1971 Edition through Summer 1973.

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

must be performed at an increased frequency of 24 hours to provide information that is adequate to detect leakage.

Restoration of the required containment pocket sump level monitor to OPERABLE status within a Completion Time of 30 days is required to regain the function after the monitor's failure. This time is acceptable, considering the Frequency and adequacy of the RCS water inventory balance required by Required Action A.1.

Required Action A.1 is modified by a Note that indicates that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when the containment pocket sump level monitor is inoperable. This allowance is provided because other instrumentation is available to monitor RCS leakage.

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

With either the gaseous or the particulate containment atmosphere radioactivity monitoring instrumentation channels inoperable, alternative action is required. Either grab samples of the containment atmosphere must be taken and analyzed or water inventory balances, in accordance with SR 3.4.13.1, must be performed to provide alternate periodic information.

With a sample obtained and analyzed or water inventory balance performed every 24 hours, the reactor may be operated for up to 30 days to allow restoration of the required containment atmosphere radioactivity monitors.

The 24 hour interval provides periodic information that is adequate to detect leakage. The 30 day Completion Time recognizes at least one other form of leakage detection is available.

Required Action B.1 and Required Action B.2 are modified by a Note that indicates that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when the gaseous and particulate containment atmosphere radioactivity monitor channel is inoperable. This allowance

(continued)

BASES

ACTIONS

B.1.1, B.1.2, and B.2 (continued)

is provided because other instrumentation is available to monitor for RCS LEAKAGE.

C.1 and C.2

If a Required Action of Condition A or B cannot be met, 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 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

D.1

With all required monitors inoperable, no automatic means of monitoring leakage are available, and immediate plant shutdown in accordance with LCO 3.0.3 is required.

SURVEILLANCE REQUIREMENTS

SR 3.4.15.1

SR 3.4.15.1 requires the performance of a CHANNEL CHECK of the required containment atmosphere radioactivity monitor. The check gives reasonable confidence that the channel is operating properly. The Frequency of 12 hours is based on instrument reliability and is reasonable for detecting off normal conditions.

SR 3.4.15.2

SR 3.4.15.2 requires the performance of a COT on the required containment atmosphere radioactivity monitor. The test ensures that the monitor can perform its function in the desired manner. The test verifies the alarm setpoint is within the required accuracy. The Frequency of 92 days considers instrument reliability, and operating experience has shown that it is proper for detecting degradation.

(continued)

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.3 ECCS – Shutdown

BASES

BACKGROUND

The Background section for Bases 3.5.2, "ECCS – Operating," is applicable to these Bases, with the following modifications.

In MODE 4, the required ECCS train consists of two separate subsystems: the high head centrifugal charging subsystem for injection and recirculation and the low head residual heat removal (RHR) subsystem for recirculation.

The ECCS flow paths consist of 1) piping, valves, heat exchangers, and pumps such that water from the refueling water storage tank (RWST) can be injected via the charging pump into the Reactor Coolant System (RCS) following the accidents described in Bases 3.5.2 and 2) piping, valves, heat exchangers, and pumps such that water from the containment sump can be recirculated to the RCS from the RHR and charging subsystems.

APPLICABLE SAFETY ANALYSES

The Applicable Safety Analyses section of Bases 3.5.2 also applies to this Bases section with the following modifications.

Due to the stable conditions associated with operation in MODE 4 and the reduced probability of occurrence of a Design Basis Accident (DBA), the ECCS operational requirements are reduced. It is understood in these reductions that certain automatic safety injection (SI) actuation is not available. In this MODE, sufficient time exists for manual actuation of the required ECCS to mitigate the consequences of a DBA.

Only one train of ECCS is required for MODE 4. This requirement dictates that single failures are not considered during this MODE of operation. The ECCS trains satisfy Criterion 3 of the NRC Policy Statement.

LCO

In MODE 4, one of the two independent (and redundant) ECCS trains is required to be OPERABLE to ensure that sufficient ECCS flow is available to the core following a DBA.

(continued)

BASES

LCO (continued)

In MODE 4, an ECCS train consists of a centrifugal charging subsystem and an RHR subsystem. Each centrifugal charging subsystem includes the piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the RWST and transferring suction to discharge of the RHR subsystem. Each RHR subsystem includes the piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the containment sump and recirculating to the RCS.

During an event requiring ECCS actuation, a flow path is required to provide an abundant supply of water from the RWST to the RCS via a charging pump and its respective supply header. in the long term, the flow path may be switched to take its supply from the containment sump and provide recirculation flow to the RCS.

APPLICABILITY

In MODES 1, 2, and 3, the OPERABILITY requirements for ECCS are covered by LCO 3.5.2.

In MODE 4, one OPERABLE ECCS train is acceptable without single failure consideration, on the basis of the stable reactivity of the reactor and the limited core cooling requirements.

In MODES 5 and 6, plant conditions are such that the probability of an event requiring ECCS injection is extremely low. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops – MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops – MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation – High Water Level," and LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation – Low Water Level."

ACTIONS

A.1

With no ECCS RHR subsystem OPERABLE, the plant is not prepared to respond to a loss of coolant accident. The Completion Time of immediately to initiate actions that would restore at least one ECCS RHR subsystem to OPERABLE status ensures that prompt action is taken to restore the required recirculation cooling capacity. Normally, in MODE 4, reactor decay heat is removed from the RCS by an RHR loop. If no RHR loop is OPERABLE for this function, reactor

(continued)

BASES

ACTIONS

A.1 (continued)

decay heat must be removed by some alternate method, such as use of the steam generators. The alternate means of heat removal must continue until the inoperable RHR loop components can be restored to operation so that decay heat removal is continuous.

With both RHR pumps and heat exchangers inoperable for decay heat removal, it would be unwise to require the plant to go to MODE 5, where the only available heat removal system is the RHR. Therefore, the appropriate action is to initiate measures to restore one ECCS RHR subsystem and to continue the actions until the subsystem is restored to OPERABLE status.

The Note allows the required ECCS RHR subsystem to be inoperable due to surveillance testing of RCS Pressure Isolation Valve leakage (FCV-74-2 and FCV-74-8). This allows testing while the RCS pressure is high enough to obtain valid leakage data and following valve closure for the RHR decay heat removal path. The condition requiring alternate heat removal methods ensures that the RCS heatup rate can be controlled to prevent Mode 3 entry and thereby ensure that the reduced ECCS operational requirements are maintained. The condition requiring manual realignment capability from the main control room ensures that in the unlikely event of a Design Basis Accident during the 1 hour of Surveillance testing, the RHR subsystem can be placed in ECCS recirculation mode when required to mitigate the event.

B.1

With no ECCS centrifugal charging subsystem OPERABLE, due to the inoperability of the centrifugal charging pump or flow path from the RWST, the plant is not prepared to provide high pressure response to Design Basis Events requiring SI. The 1 hour Completion Time to restore at least one ECCS centrifugal charging subsystem to OPERABLE status ensures that prompt action is taken to provide the required cooling capacity or to initiate actions to place the plant in MODE 5, where an ECCS train is not required.

(continued)

BASES

ACTIONS (continued)

C.1

When the Required Actions of Condition B cannot be completed within the required Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status the plant must be brought to MODE 5 within 24 hours. The allowed Completion Time is reasonable, based on operating experience, to reach the required plant condition in an orderly manner and without challenging plant systems or operators.

SURVEILLANCE REQUIREMENTS

SR 3.5.3.1

The applicable Surveillance descriptions from Bases 3.5.2 apply. This SR is modified by a Note that allows an RHR train to be considered OPERABLE during alignment and operation for decay heat removal, if capable of being manually realigned (remote or local) to the ECCS mode of operation and not otherwise inoperable. This allows operation in the RHR mode during MODE 4, if necessary.

REFERENCES

The applicable references from Bases 3.5.2 apply.

BASES

LCO (continued)

The normally closed containment isolation valves are considered OPERABLE when manual valves are closed, automatic valves are de-activated and secured in their closed position, blind flanges are in place, and closed systems are intact. These passive isolation valves/devices are those listed in Reference 2.

Purge valves with resilient seals and shield building bypass valves must meet additional leakage rate requirements. The other containment isolation valve leakage rates are addressed by LCO 3.6.1, "Containment," as Type C testing.

This LCO provides assurance that the containment isolation valves will perform their designed safety functions to minimize the loss of reactor coolant inventory and establish the containment boundary during accidents.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the containment isolation valves are not required to be OPERABLE in MODE 5. The requirements for containment isolation valves during MODE 6 are addressed in LCO 3.9.4, "Containment Penetrations."

ACTIONS

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

A second Note has been added to provide clarification that, for this LCO, separate Condition entry is allowed for each penetration flow path. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable containment isolation valve. Complying with the Required Actions may

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

within 18 seconds (20 seconds from the initiating event.) This does not include 10 seconds for diesel generator startup. The analysis shows that the annulus pressure will rise to a value above the EGTS negative pressure control setpoint (become less negative) but will not go positive.

The EGTS satisfies Criterion 3 of the NRC Policy Statement.

LCO

In the event of a DBA, one EGTS train is required to provide the minimum particulate iodine removal assumed in the safety analysis. Two trains of the EGTS must be OPERABLE to ensure that at least one train will operate, assuming that the other train is disabled by a single active failure.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could lead to fission product release to containment that leaks to the shield building. The large break LOCA, on which this system's design is based, is a full power event. Less severe LOCAs and leakage still require the system to be OPERABLE throughout these MODES. The probability and severity of a LOCA decrease as core power and Reactor Coolant System pressure decrease. With the reactor shut down, the probability of release of radioactivity resulting from such an accident is low.

In MODES 5 and 6, the probability and consequences of a DBA are low due to the pressure and temperature limitations in these MODES. Under these conditions, the Filtration System is not required to be OPERABLE (although one or more trains may be operating for other reasons, such as habitability during maintenance in the shield building annulus).

ACTIONS

A.1

With one EGTS train inoperable, the inoperable train must be restored to OPERABLE status within 7 days. The components in this degraded condition are capable of providing 100% of the iodine removal needs after a DBA. The 7 day Completion Time is based on consideration of such factors as the availability of the OPERABLE redundant EGTS train and the

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.15.1

Verifying that shield building annulus negative pressure is within limit ensures that operation remains within the limit assumed in the containment analysis. The 12 hour Frequency of this SR was developed considering operating experience related to shield building annulus pressure variations and pressure instrument drift during the applicable MODES.

SR 3.6.15.2

Maintaining shield building OPERABILITY requires maintaining each door in the access opening closed, except when the access opening is being used for normal transient entry and exit. The 31 day Frequency of this SR is based on engineering judgment and is considered adequate in view of the other indications of door status that are available to the operator.

SR 3.6.15.3

This SR would give advance indication of gross deterioration of the concrete structural integrity of the shield building. The Frequency of this SR is the same as that of SR 3.6.1.1. The verification is done during shutdown.

SR 3.6.15.4

The EGTS is required to maintain a pressure equal to or more negative than -0.50 inches of water gauge ("wg) in the annulus at an elevation equivalent to the top of the Auxiliary Building. At elevations higher than the Auxiliary Building, the EGTS is required to maintain a pressure equal to or more negative than -0.25 "wg. The low pressure sense line for the pressure controller is located in the annulus at elevation 783. By verifying that the annulus pressure is equal to or more negative than -0.61 "wg at elevation 783, the annulus pressurization requirements stated above are met. The ability of a EGTS train with final flow $\geq$ 3600 and $\leq$ 4400 cfm to produce the required negative pressure during the test operation provides assurance that the building is adequately sealed. The negative pressure prevents leakage from the building, since outside air will be drawn in by the low pressure at a maximum rate $\leq$ 250 cfm. The 18 month Frequency on a STAGGERED TEST BASIS is consistent with Regulatory Guide 1.52 (Ref. 1) guidance for functional testing.

(continued)

BASES

LCO (continued)

This LCO provides assurance that the MSSVs will perform their designed safety functions to mitigate the consequences of accidents that could result in a challenge to the RCPB.

APPLICABILITY

In MODE 1 above 26% RTP, the number of MSSVs per steam generator required to be OPERABLE must be according to Table 3.7.1-1 in the accompanying LCO. Below 26% RTP in MODES 1, 2, and 3, only two MSSVs per steam generator are required to be OPERABLE.

In MODES 4 and 5, there are no credible transients requiring the MSSVs. The steam generators are not normally used for heat removal in MODES 5 and 6, and thus cannot be overpressurized; there is no requirement for the MSSVs to be OPERABLE in these MODES.

ACTIONS

The ACTIONS table is modified by a Note indicating that separate Condition entry is allowed for each MSSV.

A.1

With one or more MSSVs inoperable, reduce power so that the available MSSV relieving capacity meets Reference 2 requirements for the applicable THERMAL POWER.

Operation with less than all five MSSVs OPERABLE for each steam generator is permissible, if THERMAL POWER is proportionally limited to the relief capacity of the remaining MSSVs. With an upper MTC limit of 0 $\Delta k/k/°F$, this is accomplished by maintaining THERMAL POWER at or below the power levels specified in Table 3.7.1.1. This ensures that the energy transfer to the most limiting steam generator is not greater than the available relief capacity in that steam generator. The reduced THERMAL POWER level for a reduced steam relieving capacity can be determined by performing a energy balance between the reactor coolant system heat

(continued)

BASES

LCO (continued)

OPERABLE. The MFIVs and MFRVs and the associated bypass valves are considered OPERABLE when isolation times are within limits and they close on an isolation actuation signal.

Failure to meet the LCO requirements can result in additional mass and energy being released to containment following an SLB or FWLB inside containment. If a feedwater isolation signal on high-high steam generator level is relied on to terminate an excess feedwater flow event, failure to meet the LCO may result in the introduction of water into the main steam lines.

APPLICABILITY

The MFIVs and MFRVs and the associated bypass valves must be OPERABLE whenever there is significant mass and energy in the Reactor Coolant System and steam generators. This ensures that, in the event of an HELB, a single failure cannot result in the blowdown of more than one steam generator. In MODES 1, 2, and 3, the MFIVs and MFRVs and the associated bypass valves are required to be OPERABLE, except when closed and de-activated to limit the amount of available fluid that could be added to containment in the case of a secondary system pipe break inside containment. When the valves are closed and de-activated or isolated by a closed manual valve, they are already performing their safety function.

In MODES 4, 5, and 6, steam generator energy is low. Therefore, the MFIVs, MFRVs, and the associated bypass valves are normally closed since MFW is not required.

ACTIONS

The ACTIONS table is modified by a Note indicating that separate Condition entry is allowed for each valve.

A.1 and A.2

With one MFIV in one or more flow paths inoperable, action must be taken to restore the affected valves to OPERABLE status, or to close or isolate inoperable affected valves within 72 hours. When these valves are closed or isolated, they are performing their required safety function.

(continued)

BASES

LCO (continued)

a. Two pumps, aligned to separate shutdown boards, are OPERABLE; and

b. The associated piping, valves, heat exchanger, and instrumentation and controls required to perform the safety related function are OPERABLE.

APPLICABILITY

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

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

ACTIONS

A.1

If one ERCW train is inoperable, action must be taken to restore OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE ERCW train is adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure in the OPERABLE ERCW train could result in loss of ERCW System function. Required Action A.1 is modified by two Notes. The first Note indicates that the applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources Operating," should be entered if an inoperable ERCW train results in an inoperable emergency diesel generator. The second Note indicates that the applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops MODE 4," should be entered if an inoperable ERCW train results in an inoperable decay heat removal train. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components. The 72 hour Completion Time is based on the redundant capabilities afforded by the OPERABLE train, and the low probability of a DBA occurring during this time period.

B.1 and B.2

If the ERCW train cannot be restored to OPERABLE status within the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.10.2

This SR verifies that the required CREVS testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The CREVS filter tests are in accordance with Regulatory Guide 1.52 (Ref. 3). The VFTP includes testing the performance of the HEPA filter, charcoal adsorber efficiency, minimum flow rate, and the physical properties of the activated charcoal. Specific test Frequencies and additional information are discussed in detail in the VFTP.

SR 3.7.10.3

This SR verifies that each CREVS train starts and operates on an actual or simulated actuation signal. The Frequency of 18 months is specified in Regulatory Guide 1.52 (Ref. 3).

SR 3.7.10.4

This SR verifies the integrity of the control room enclosure, and the assumed inleakage rates of the potentially contaminated air. The control room positive pressure, with respect to potentially contaminated adjacent areas, is periodically tested to verify proper functioning of the CREVS. During the emergency mode of operation, the CREVS is designed to pressurize the control room $\geq$ 0.125 inches water gauge positive pressure with respect to the outside atmosphere and adjacent areas in order to prevent unfiltered inleakage. The CREVS is designed to maintain this positive pressure with one train at a makeup flow rate $\leq$ 711 cfm and a recirculation flow rate $\geq$ 2960 and $\leq$ 3618 cfm. The Frequency of 18 months on a STAGGERED TEST BASIS is consistent with the guidance provided in NUREG-0800 (Ref. 4).

REFERENCES

1. Watts Bar FSAR, Section 6.4, "Habitability Systems."

2. Watts Bar FSAR, Section 15.5.3, "Environmental Consequences of a Postulated Loss of Coolant Accident."

(continued)

BASES

LCO (continued)

anticipated operational occurrence (AOO) or a postulated DBA.

Qualified offsite circuits are those that are described in the FSAR and are part of the licensing basis for the plant.

Each offsite circuit must be capable of maintaining acceptable frequency and voltage, and accepting required loads during an accident, while connected to the 6.9 kV shutdown boards.

Offsite power from the Watts Bar Hydro 161 kV switchyard to the onsite Class 1E distribution system is from two independent immediate access circuits. Each of the two circuits are routed from the switchyard through a 161 kV transmission line and 161 to 6.9 kV transformer (common station service transformers) to the onsite Class 1E distribution system. The medium voltage power system starts at the low-side of the common station service transformers.

Each DG must be capable of starting, accelerating to rated speed and voltage, and connecting to its respective 6.9 kV shutdown board on detection of loss-of-voltage. This will be accomplished within 10 seconds. Each DG must also be capable of accepting required loads within the assumed loading sequence intervals, and continue to operate until offsite power can be restored to the 6.9 kV shutdown boards. These capabilities are required to be met from a variety of initial conditions such as DG in standby with the engine hot and DG in standby with the engine at ambient conditions. Additional DG capabilities must be demonstrated to meet required Surveillances, e.g., capability of the DG to revert to standby status on an accident signal while operating in parallel test mode.

Proper sequencing of loads, including tripping of nonessential loads, is a required function for DG OPERABILITY.

A Note has been added to indicate that the C-S DG may be substituted for any of the required DGs. However, the C-S DG cannot be declared OPERABLE until it is connected electrically in place of another DG, and it has satisfied applicable Surveillance Requirements.

(continued)