LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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TABLE 2.2-1 (Continued)

TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \left(\frac{1}{1 + \tau_3 S}\right) \le \Delta T_o \left\{K_1 - K_2 \frac{(1 + \tau_4 S)}{(1 + \tau_5 S)} \left[T \left(\frac{1}{1 + \tau_6 S}\right) - T'\right] + K_3(P - P') - f_1(\Delta I)\right\}$$

Where:

ΔT	=	Measured ΔT by RTD Manifold Instrumentation;
$\frac{1 + \tau_1 S}{1 + \tau_2 S}$	=	Lead-lag compensator on measured ΔT;
τ_1, τ_2	=	Time constants utilized in lead-lag compensator for ΔT, $\tau_1 = 8$ s, $\tau_2 = 3$ s;
$\frac{1}{1 + \tau_3 S}$	=	Lag compensator on measured ΔT;
τ_3	=	Time constants utilized in the lag compensator for ΔT, $\tau_3 = 0$ s;
ΔT_o	=	Indicated ΔT at RATED THERMAL POWER;
K_1	=	1.09;
K_2	=	0.0182/°F;
$\frac{1 + \tau_4 S}{1 + \tau_5 S}$	=	The function generated by the lead-lag compensator for T_{avg} dynamic compensation;
τ_4, τ_5	=	Time constants utilized in the lead-lag compensator for T_{avg}, $\tau_4 = 20$ s, $\tau_5 = 4$ s;
T	=	Average temperature, °F;
$\frac{1}{1 + \tau_6 S}$	=	Lag compensator on measured T_{avg};
τ_6	=	Time constant utilized in the measured T_{avg} lag compensator, $\tau_6 = 0$ s;

TABLE 3.3-2 REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

This table is deleted from Technical Specifications.

The information in this table is controlled by plant procedure PLP-106, Technical Specification Equipment List Program.

~~TABLE 3.3-2 (pages 3/4 3-9 and 3-10) has been deleted~~
~~Refer to plant procedure PLP-106~~

~~TABLE 3.3-5 (pages 3/4 3-37 through 40) has been deleted. Refer to plant procedure PLP-106.~~

TABLE 3.3-5 ENGINEERED SAFETY FEATURES RESPONSE TIMES

This table is deleted from Technical Specifications.

The information in this table is controlled by plant procedure PLP-106, Technical Specification Equipment List Program.

INSTRUMENTATION

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.6 The accident monitoring instrumentation channels shown in Table 3.3-10 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With the number of OPERABLE accident monitoring instrumentation channels less than the Total Required Number of Channels shown in Table 3.3-10, except for the pressurizer safety valve position indicator or the sub-cooling margin monitor, restore the inoperable channel(s) to OPERABLE status within 7 days, or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours; or

b. With the number of OPERABLE accident monitoring instrumentation channels, except the radiation monitors, the pressurizer safety valve position indicator, or the sub-cooling margin monitor, less than the Minimum Channels OPERABLE requirements of Table 3.3-10, restore the inoperable channel(s) to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours; or

c. With the number of OPERABLE channels for the radiation monitors, the pressurizer safety valve position indicator*, or the sub-cooling margin monitor#, less than required by the Minimum Channels OPERABLE requirements, initiate the preplanned alternate method of monitoring the appropriate parameter(s), within 72 hours, and either restore the inoperable channel(s) to OPERABLE status within 7 days or prepare and submit a Special Report to the Commission, pursuant to Specification 6.9.2, within the next 14 days, that provides actions taken, cause of the inoperability, and the plans and schedule for restoring the channels to OPERABLE status.

d. The provisions of Specification 3.0.4 are not applicable.

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* The alternate method shall be a check of safety valve piping temperatures and evaluation to determine position.

The alternate method shall be the initiation of the backup method as required by Specification 6.8.4.d.

TABLE 4.3-8

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	DIGITAL CHANNEL OPERATIONAL TEST
1. Radioactivity Monitors Providing Alarm and Automatic Termination of Release				
a. Liquid Radwaste Effluent Lines				
1) Treated Laundry and Hot Shower Tanks Discharge Monitor	D	P	R(3)	Q(1)
2) Waste Monitor Tanks and Waste Evaporator Condensate Tanks Discharge Monitor	D	P	R(3)	Q(1)
3) Secondary Waste Sample Tank Discharge Monitor	D	P, M(5)	R(3)	Q(1)
b. Turbine Building Floor Drains Effluent Line	D	M	R(3)	Q(1)
c. Outdoor Tank Area Drain Transfer Pump Monitor	D	M	R(3)	Q(1)
2. Radioactivity Monitors Providing Alarm But Not Providing Automatic Termination of Release				
a. Normal Service Water System Return From the Waste Processing Building to the Circulating Water System	D	M	R(3)	Q(2)

REACTOR COOLANT SYSTEM

HOT SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.1.3 At least two of the loops listed below shall be OPERABLE and at least one of these loops shall be in operation:*

a. Reactor Coolant Loop A and its associated steam generator and reactor coolant pump,**

b. Reactor Coolant Loop B and its associated steam generator and reactor coolant pump,**

c. Reactor Coolant Loop C and its associated steam generator and reactor coolant pump,**

d. RHR Loop ~~[~~A~~]~~, or

e. RHR Loop ~~[~~B~~]~~.

APPLICABILITY: MODE 4.

ACTION:

a. With less than the above required loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible; if the remaining OPERABLE loop is an RHR loop, be in COLD SHUTDOWN within 24 hours.

b. With no loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required loop to operation.

*All reactor coolant pumps and RHR pumps may be deenergized for up to 1 hour provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

**A reactor coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to 335°F unless the secondary water temperature of ~~each~~ steam generator is less than 50°F above each of the Reactor Coolant System cold leg temperatures.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.5.4 Acceptance Criteria (Continued)

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Tables 4.4-2A and B.

4.4.5.5 Reports

a. Within 15 days following the completion of each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Commission in a Special Report pursuant to Specification 6.9.2;

b. The complete results of the steam generator tube inservice inspection shall be submitted to the Commission in a Special Report pursuant to Specification 6.9.2 within 12 months following the completion of the inspection. This Special 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.

c. Results of steam generator tube inspections which fall into Category C-3 shall be reported in a Special Report ~~to the Regional Administrator with a copy to NRR~~ pursuant to Spec. 6.9.2 within 30 days and prior to resumption of plant operation. This report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.9.2 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figures 3.4-2 and 3.4-3 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing with:

a. A maximum heatup rate as shown on Table 4.4-6.

b. A maximum cooldown rate as shown on Table 4.4-6.

c. A maximum temperature change of less than or equal to 10°F in any 1-hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.

APPLICABILITY: MODES 4, 5, and 6 with reactor vessel head on.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operation or maintain the RCS T_{avg} and pressure at less than 200°F and 500 psig, respectively.

SURVEILLANCE REQUIREMENTS

4.4.9.2.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

4.4.9.2.2 Deleted. from Technical Specifications. Refer to plant procedure PLP-106, Technical Specification Equipment List Program.

TABLE 4.4-5 REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM

This table is deleted from Technical Specifications

The information in this table is controlled by plant procedure PLP-106, Technical Specification Equipment List Program.

~~TABLE 4.4-5 has been deleted. Refer to plant procedures PLP-106.~~

SURVEILLANCE REQUIREMENTS (Continued)

4.6.3.2 Each isolation valve ~~specified in Table 3.6-1~~ shall be demonstrated OPERABLE during the COLD SHUTDOWN or REFUELING MODE at least once per 18 months by:

a. Verifying that on a Phase "A" Isolation test signal, each Phase "A" isolation valve actuates to its isolation position;

b. Verifying that on a Phase "B" Isolation test signal, each Phase "B" isolation valve actuates to its isolation position; and

c. Verifying that on a Containment Ventilation Isolation test signal, each normal, preentry purge makeup and exhaust, and containment vacuum relief valve actuates to its isolation position, and

d. Verifying that, on a Safety Injection "S" test signal, each containment isolation valve receiving an "S" signal actuates to its isolation position, and

e. Verifying that, on a Main Steam Isolation test signal, each main steam isolation valve actuates to its isolation position, and

f. Verifying that, on a Main Feedwater Isolation test signal, each feedwater isolation valve actuates to its isolation position.

4.6.3.3 The isolation time of each power-operated or automatic valve ~~of Table 3.6-1~~ shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

~~Table 3.6-1 (pages 3/4 6-16 through 29) has been deleted. Refer to plant procedure PLP-106.~~

TABLE 3.6-1 CONTAINMENT ISOLATION VALVES

This table is deleted from Technical Specifications.

The information in this table is controlled by plant procedure PLP-106, Technical Specification Equipment List Program.

~~PLANT SYSTEMS~~

~~SNUBBERS~~

~~Pages 3/4 7-20 through 23 have been deleted. Refer to plant procedure PLP-106.~~

The inservice inspection program for SNUBBERS is deleted from Technical Specifications, and is controlled in plant procedure PLP-106, Technical Specification Equipment List Program.

FIGURE 4.7-1 SAMPLE PLAN (2) FOR SNUBBER FUNCTIONAL TEST

This figure is deleted from Technical Specifications, and is controlled in plant procedure PLP-106, Technical Specification Equipment List Program.

~~FIGURE 4.7-1 has been deleted. Refer to plant procedure PLP-106.~~

~~TABLE 3.8-1 (pages 3/4 8-21 through 38B) has been deleted.~~
~~Refer to plant procedure PLP-106.~~

TABLE 3.8-1 CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

This table is deleted from Technical Specifications.

The information in this table is controlled by plant procedure PLP-106, Technical Specification Equipment List Program.

~~TABLE 3.8-2 (pages 3/4 8-40 through 43) has been deleted. Refer to plant procedure PLP-106.~~

TABLE 3.8-2 MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION

This table is deleted from Technical Specifications.

The information in this table is controlled by plant procedure PLP-106, Technical Specification Equipment List Program.

QUADRANT POWER TILT RATIO (Continued)

For purposes of monitoring QUADRANT POWER TILT RATIO when one excore detector is inoperable, the moveable incore detectors are used to confirm that the normalized symmetric power distribution is consistent with the QUADRANT POWER TILT RATIO. The incore detector monitoring is done with a full incore flux map or two sets of four symmetric thimbles. The preferred sets of four symmetric thimbles is a unique set of eight detector locations. These locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11, N-8. If other locations must be used, a special report to ~~NRR~~ the NRC should be submitted within 30 days. in accordance with 10CFR50.4.

3/4.2.5 DNB PARAMETERS

The limits on the DNB-related parameters assure that each of the parameters are maintained within the normal steady-state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR of 1.30 throughout each analyzed transient. The indicated T_{avg} value and the indicated pressurizer pressure value are compared to analytical limits of 592.6°F and 2205 psig, respectively, with allowance for measurement uncertainty.

The 12-hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the ~~Regional Administrator of the Regional Office of the NRC unless otherwise noted.~~ NRC in accordance with 10CFR50.4.

STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following: (1) receipt of an Operating License, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the unit.

The Startup Report shall address each of the tests identified in the Final Safety Analysis Report and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

Startup Reports shall be submitted within: (1) 90 days following completion of the Startup Test Program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of Startup Test Program, and resumption or commencement of commercial operation), supplementary reports shall be submitted at least every 3 months until all three events have been completed.

ANNUAL REPORTS

6.9.1.2 Annual Reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality.

Reports required on an annual basis shall include:

a. A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man-rem exposure according to work and job functions* (e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance [describe maintenance], waste processing, and refueling). The dose assignments to various duty functions may be estimated based on pocket dosimeter, thermoluminescent dosimeter (TLD), or

*This tabulation supplements the requirements of §20.407 of 10 CFR Part 20.

RADIAL PEAKING FACTOR LIMIT REPORT

6.9.1.6 The F_{xy} limits for RATED THERMAL POWER (F_{xy}^{RTP}) shall be provided to the NRC in accordance with 10CFR50.4 ~~Regional Administrator with a copy to the Director of Nuclear Reactor Regulation, Attention: Chief, Reactor Systems Branch, Division of PWR Licensing-A, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555~~, for all core planes containing Bank "D" control rods and all unrodded core planes and the plot of predicted ($F_q^T \cdot P_{Rel}$) vs Axial Core Height with the limit envelope at least 60 days prior to each cycle initial criticality unless otherwise approved by the Commission by letter. In addition, in the event that the limit should change requiring a new substantial or an amended submittal to the Radial Peaking Factor Limit Report, it will be submitted 60 days prior to the date the limit would become effective unless otherwise approved by the Commission by letter. Any information needed to support F_{xy}^{RTP} will be by request from the NRC and need not be included in this report.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the NRC in accordance with 10CFR50.4 ~~Regional Administrator of the Regional Office of the NRC~~ within the time period specified for each report.

6.10 RECORD RETENTION

6.10.1 In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

6.10.2 The following records shall be retained for at least 5 years:

a. Records and logs of unit operation covering time interval at each power level;

b. Records and logs of principal maintenance activities, inspections, repair, and replacement of principal items of equipment related to nuclear safety;

c. All REPORTABLE EVENTS;

d. Records of surveillance activities, inspections, and calibrations required by these Technical Specifications;

e. Records of changes made to the procedures required by Specification 6.8.1;

f. Records of radioactive shipments;

g. Records of sealed source and fission detector leak tests and results; and