

(1) Maximum Power Level

Carolina Power & Light Company is authorized to operate the facility at reactor core power levels not in excess of 2900 megawatts thermal (100 percent rated core power) in accordance with the conditions specified herein.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, as revised through Amendment No. 107, are hereby incorporated into this license. Carolina Power & Light Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Antitrust Conditions

Carolina Power & Light Company shall comply with the antitrust conditions delineated in Appendix C to this license.

(4) Initial Startup Test Program (Section 14)*

Any changes to the Initial Test Program described in Section 14 of the FSAR made in accordance with the provisions of 10 CFR 50.59 shall be reported in accordance with 50.59(b) within one month of such change.

(5) Steam Generator Tube Rupture (Section 15.6.3)

Prior to startup following the first refueling outage, Carolina Power & Light Company shall submit for NRC review and receive approval of a steam generator tube rupture analysis, including the assumed operator actions, which demonstrates that the consequences of the design basis steam generator tube rupture event for the Shearon Harris Nuclear Power Plant are less than the acceptance criteria specified in the Standard Review Plan, NUREG-0800, at §15.6.3 Subparts II(1) and (2) for calculated doses from radiological releases. In preparing their analysis Carolina Power & Light Company will not assume that operators will complete corrective actions within the first thirty minutes after a steam generator tube rupture.

*The parenthetical notation following the title of many license conditions denotes the section of the Safety Evaluation Report and/or its supplements wherein the license condition is discussed.

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DEFINITIONS

DOSE EQUIVALENT I-131

1.11 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microCurie/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in International Commission on Radiological Protection, "Limits for Intakes of Radionuclides by Workers," ICRP Publication 30, Volume 3 No. 1-4, 1979 (or equivalently, Federal Guidance Report No. 11 "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion." EPA 520/1-88-020, September 1988).

DEFINITIONS

PROCESS CONTROL PROGRAM

1.25 The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, test, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71 and State regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.

PURGE - PURGING

1.26 PURGE or PURGING shall be any controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

QUADRANT POWER TILT RATIO

1.27 QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one excore detector inoperable, the remaining three detectors shall be used for computing the average.

RATED THERMAL POWER

1.28 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 2900 Mwt.

REACTOR TRIP SYSTEM RESPONSE TIME

1.29 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its Trip Setpoint at the channel sensor until loss of stationary gripper coil voltage.

REPORTABLE EVENT

1.30 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 of 10 CFR Part 50.

SHUTDOWN MARGIN

1.31 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be fully withdrawn.

SITE BOUNDARY

1.32 For these Specifications, the SITE BOUNDARY shall be identical to the EXCLUSION AREA BOUNDARY defined above.

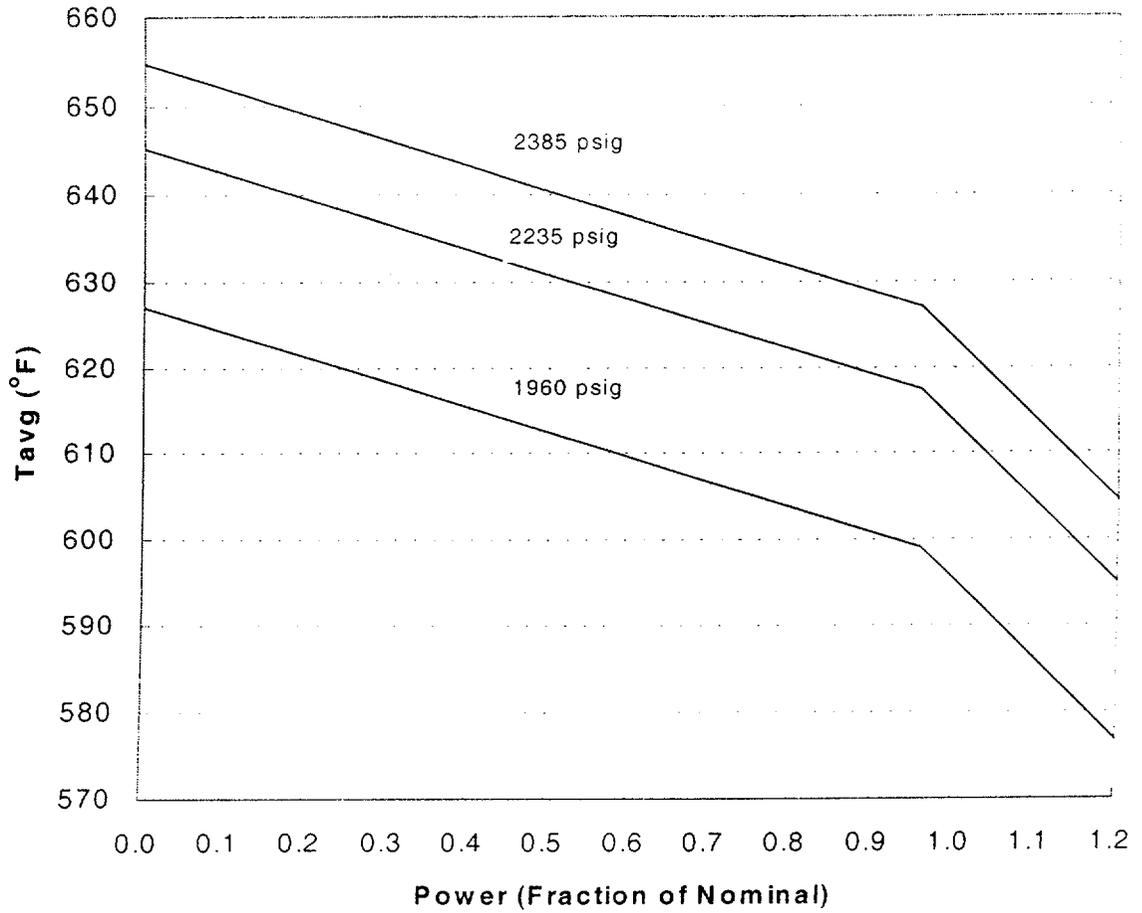


FIGURE 2.1-1
 REACTOR CORE SAFETY LIMITS - THREE LOOPS IN OPERATION
 WITH MEASURED RCS FLOW > [293,540 GPM X (1.0 + C₁)]

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

<u>FUNCTIONAL UNIT</u>	<u>TOTAL ALLOWANCE (TA)</u>	<u>Z</u>	<u>SENSOR ERROR (S)</u>	<u>TRIP SETPOINT</u>	<u>ALLOWABLE VALUE</u>
1. Manual Reactor Trip	N.A.	N.A.	N.A.	N.A.	N.A.
2. Power Range, Neutron Flux					
a. High Setpoint	7.5	4.56	0	≤ 109% of RTP**	≤ 111.1% of RTP**
b. Low Setpoint	8.3	4.56	0	≤ 25% of RTP**	≤ 27.1% of RTP**
3. Power Range, Neutron Flux, High Positive Rate	2.5	0.83	0	≤ 5% of RTP** with a time constant ≥ 2 seconds	≤ 6.3% of RTP** with a time constant ≥ 2 seconds
4. Power Range, Neutron Flux, High Negative Rate	2.5	0.83	0	≤ 5% of RTP** with a time constant ≥ 2 seconds	≤ 6.3% of RTP** with a time constant ≥ 2 seconds
5. Intermediate Range, Neutron Flux	17.0	8.41	0	≤ 25% of RTP**	≤ 30.9% of RTP**
6. Source Range, Neutron Flux	17.0	10.01	0	≤ 10 ⁵ cps	≤ 1.4 x 10 ⁵ cps
7. Overtemperature ΔT	9.0	7.31	Note 5	See Note 1	See Note 2
8. Overpower ΔT	4.0	2.32	1.3	See Note 3	See Note 4
9. Pressurizer Pressure-Low	5.0	1.52	1.5	≥ 1960 psig	≥ 1948 psig
10. Pressurizer Pressure-High	7.5	1.52	1.5	≤ 2385 psig	≤ 2397 psig
11. Pressurizer Water Level-High	8.0	3.42	1.75	≤ 92% of instrument span	≤ 93.5% of instrument span

**RTP = RATED THERMAL POWER

TABLE 2.2-1 (continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

<u>FUNCTIONAL UNIT</u>	<u>TOTAL ALLOWANCE (TA)</u>	<u>Z</u>	<u>SENSOR ERROR (S)</u>	<u>TRIP SETPOINT</u>	<u>ALLOWABLE VALUE</u>
12. Reactor Coolant Flow-Low	4.58	1.98	0.6	≥ 90.5% of loop full indicated flow	≥ 89.5% of loop full indicated flow
13. Steam Generator Water Level Low-Low	25.0	16.85	2.0	≥ 25.0% of narrow range instrument span	≥ 23.5% of narrow range instrument span
14. Steam Generator Water Level - Low Coincident With Steam/Feedwater Flow Mismatch	8.9	5.35	2.0	≥ 25.0% of narrow range instrument span	≥ 23.5% of narrow range instrument span
	20.0	3.01	Note 6	≤ 40% of full steam flow at RTP**	≤ 43.1% of full steam flow at RTP**
15. Undervoltage - Reactor Coolant Pumps	14.0	1.3	0.0	≥ 5148 volts	≥ 4920 volts
16. Underfrequency - Reactor Coolant Pumps	5.0	3.0	0.0	≥ 57.5 Hz	≥ 57.3 Hz
17. Turbine Trip					
a. Low Fluid Oil Pressure	N.A.	N.A.	N.A.	≥ 1000 psig	≥ 950 psig
b. Turbine Throttle Valve Closure	N.A.	N.A.	N.A.	≥ 1% open	≥ 1% open
18. Safety Injection Input from ESF	N.A.	N.A.	N.A.	N.A.	N.A.

**RTP = RATED THERMAL POWER

TABLE 2.2-1 (Continued)
TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE ΔT

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

Where: ΔT = Measured ΔT by RTD 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 , $\tau_1 = 0$ s, $\tau_2 = 0$ s;

$\frac{1}{1 + \tau_3 S}$ = Lag compensator on measured ΔT ;

τ_3 = Time constants utilized in the lag compensator for ΔT , $\tau_3 = 4$ s;

ΔT_o = Indicated ΔT at RATED THERMAL POWER;

K_1 = 1.185;

K_2 = 0.0224/°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} , $\tau_4 = 22$ s, $\tau_5 = 4$ s;

TABLE 2.2-1 (Continued)
TABLE NOTATIONS

NOTE 1: (Continued)

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;
T'	= Reference T_{avg} at RATED THERMAL POWER ($\leq 588.8^\circ\text{F}$);
K_3	= 0.0012/psig;
P	= Pressurizer pressure, psig;
P'	= 2235 psig (Nominal RCS operating pressure);
S	= Laplace transform operator, s^{-1} ;

and $f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range neutron ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

- (1) For $q_t - q_b$ between -21.6% and +12.0%, $f_1(\Delta I) = 0$, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER;
- (2) For each percent that the magnitude of $q_t - q_b$ exceeds -21.6%, the ΔT Trip Setpoint shall be automatically reduced by 1.75% of its value at RATED THERMAL POWER; and
- (3) For each percent that the magnitude of $q_t - q_b$ exceeds + 12.0%, the ΔT Trip Setpoint shall be automatically reduced by 1.50% of its value at RATED THERMAL POWER.

NOTE 2: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 1.4% of ΔT span for ΔT input; 2.0% of ΔT span for T_{avg} input; 0.4% of ΔT span for pressurizer pressure input; and 0.7% of ΔT span for ΔI input.

TABLE 2.2-1 (Continued)

TABLE NOTATIONS

NOTE 3: OVERPOWER ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \frac{(1)}{(1 + \tau_3 S)} \leq \Delta T_o \left\{ K_4 - K_5 \frac{(\tau_7 S)}{(1 + \tau_7 S)} \frac{(1)}{(1 + \tau_6 S)} T - K_6 \left[T \frac{(1)}{(1 + \tau_6 S)} - T'' \right] - f_2(\Delta I) \right\}$$

- Where:
- ΔT = As defined in Note 1.
 - $\frac{1 + \tau_1 S}{1 + \tau_2 S}$ = As defined in Note 1.
 - τ_1, τ_2 = As defined in Note 1.
 - $\frac{1}{1 + \tau_3 S}$ = As defined in Note 1.
 - τ_3 = As defined in Note 1.
 - ΔT_o = As defined in Note 1.
 - K_4 = 1.12.
 - K_5 = 0.02/°F for increasing average temperature and 0 for decreasing average temperature.
 - $\frac{\tau_7 S}{1 + \tau_7 S}$ = The function generated by the rate-lag compensator for T_{avg} dynamic compensation.
 - τ_7 = Time constants utilized in the rate-lag compensator for T_{avg} . $\tau_7 = 13$ s.
 - $\frac{1}{1 + \tau_6 S}$ = As defined in Note 1.
 - τ_6 = As defined in Note 1.

TABLE 2.2-1 (Continued)

TABLE NOTATIONS

NOTE 3: (Continued)

K_6	=	0.002/°F for $T > T''$ and $K_6 = 0$ for $T \leq T''$.
T	=	As defined in Note 1.
T''	=	Reference T_{avg} at RATED THERMAL POWER ($\leq 588.8^\circ\text{F}$).
S	=	As defined in Note 1, and
$f_2(\Delta I)$	=	0 for all ΔI .

NOTE 4: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 1.4% of ΔI span for ΔT input and 0.2% of ΔT span for T_{avg} input.

NOTE 5: The sensor error is: 1.3% of ΔT span for $\Delta T/T_{avg}$ temperature measurements; and 1.0% of ΔT span for pressurizer pressure measurements.

NOTE 6: The sensor error (in % span of Steam Flow) is: 1.1% for steam flow; 1.8% for feedwater flow; and 2.4% for steam pressure.

2.1 SAFETY LIMITS

BASES

2.1.1 REACTOR CORE

The restrictions of this safety limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB. This relation has been developed to predict the DNB flux and the location of DNB for axially uniform and nonuniform heat flux distributions. The local DNB heat flux ratio (DNBR) defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux is indicative of the margin to DNB.

The DNB design basis is as follows: there must be at least a 95 percent probability that the minimum DNBR of the limiting rod during Condition I and II events is greater than or equal to the DNBR limit of the DNB correlation being used (in this application, the HTP correlation for Siemens Fuel. The correlation DNBR limit is established based on the entire applicable experimental data set such that there is a 95 percent probability with 95 percent confidence that DNB will not occur when the minimum DNBR is at the DNBR limit.

The curves of Figure 2.1-1 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature below which the calculated DNBR is no less than the design DNBR value or the average enthalpy at the vessel exit is less than the enthalpy of saturated liquid.

LIMITING SAFETY SYSTEM SETTINGS

BASES

Reactor Coolant Flow (Continued)

(a power level of approximately 49% of RATED THERMAL POWER) an automatic Reactor trip will occur if the flow in any single loop drops below 90.5% of nominal full loop flow. Conversely, on decreasing power between P-8 and the P-7 an automatic Reactor trip will occur on low reactor coolant flow in more than one loop and below P-7 the trip function is automatically blocked.

Steam Generator Water Level

The Steam Generator Water Level Low-Low trip protects the reactor from loss of heat sink in the event of a sustained steam/feedwater flow mismatch resulting from loss of normal feedwater. The specified Setpoint provides allowances for starting delays of the Auxiliary Feedwater System.

Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level

The Steam/Feedwater Flow Mismatch in coincidence with a Steam Generator Low Water Level trip is not used in the transient and accident analyses but is included in Table 2.2-1 to ensure the functional capability of the specified trip settings and thereby enhance the overall reliability of the Reactor Trip System. This trip is redundant to the Steam Generator Water Level Low-Low trip. The Steam/Feedwater Flow Mismatch portion of this trip is activated when the steam flow exceeds the feedwater flow by the setpoint value. The Steam Generator Low Water level portion of the trip is activated when the setpoint value is reached, as indicated by the narrow range instrument. These trip values include sufficient allowance in excess of normal operating values to preclude spurious trips but will initiate a Reactor trip before the steam generators are dry. Therefore, the required capacity and starting time requirements of the auxiliary feedwater pumps are reduced and the resulting thermal transient on the Reactor Coolant System and steam generators is minimized.

Undervoltage and Underfrequency - Reactor Coolant Pump Buses

The Undervoltage and Underfrequency Reactor Coolant Pump Bus trips provide core protection against DNB as a result of complete loss of forced coolant flow. The specified Setpoints assure a Reactor trip signal is generated before the Low Flow Trip Setpoint is reached. Time delays are incorporated in the Underfrequency and Undervoltage trips to prevent spurious Reactor trips from momentary electrical power transients. For undervoltage, the delay is set so that the time required for a signal to reach the Reactor trip breakers following the simultaneous trip of two or more reactor coolant pump bus circuit breakers shall not exceed 1.2 seconds. For underfrequency, the delay is set so that the time required for a signal to reach the Reactor trip breakers after the Underfrequency Trip Setpoint is reached shall not exceed 0.3 second.

On decreasing power the Undervoltage and Underfrequency Reactor Coolant Pump Bus trips are automatically blocked by the loss of P-7 (a power-level of approximately 10% of RATED THERMAL POWER or a turbine impulse chamber pressure

REACTIVITY CONTROL SYSTEMS

BORATED WATER SOURCE - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.5 As a minimum, one of the following borated water sources shall be OPERABLE:

- a. The boric acid tank with:
 1. A minimum contained borated water volume of 7150 gallons which is ensured by maintaining indicated level of greater than or equal to 23%.
 2. A boron concentration of between 7000 and 7750 ppm, and
 3. A minimum solution temperature of 65°F.
- b. The refueling water storage tank (RWST) with:
 1. A minimum contained borated water volume of 106,000 gallons, which is equivalent to 12% indicated level.
 2. A boron concentration of between 2400 and 2600 ppm, and
 3. A minimum solution temperature of 40°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water source OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.5 The above required borated water source shall be demonstrated OPERABLE:

- a. At least once per 7 days by:
 1. Verifying the boron concentration of the water.
 2. Verifying the contained borated water volume, and
 3. Verifying the boric acid tank solution temperature when it is the source of borated water.
- b. At least once per 24 hours by verifying the RWST temperature when it is the source of borated water and the outside air temperature is less than 40°F.

POWER DISTRIBUTION LIMITS

3/4.2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB-related parameters shall be maintained within the following limits:

- a. Reactor Coolant System $T_{avg} \leq 594.8^{\circ}\text{F}$ after addition for instrument uncertainty, and
- b. Pressurizer Pressure ≥ 2185 psig* after subtraction for instrument uncertainty, and
- c. RCS total flow rate $\geq 293,540$ gpm after subtraction for instrument uncertainty.

APPLICABILITY: MODE 1.

ACTION:

With any of the above parameters not within its specified limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.2.5.1 Each of the parameters shown in Specification 3.2.5 shall be verified to be within its limit at least once per 12 hours.

4.2.5.2 Verify, by precision heat balance, that RCS total flow rate is within its limit at least once per 18 months.**

* This limit is not applicable during either a THERMAL POWER Ramp in excess of $\pm 5\%$ RATED THERMAL POWER per minute or a THERMAL POWER step change in excess of $\pm 10\%$ RATED THERMAL POWER.

** Required to be performed within 24 hours after $\geq 95\%$ RATED THERMAL POWER.

TABLE 3.3-4

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

<u>FUNCTIONAL UNIT</u>	<u>TOTAL ALLOWANCE (TA)</u>	<u>Z</u>	<u>SENSOR ERROR (S)</u>	<u>TRIP SETPOINT</u>	<u>ALLOWABLE VALUE</u>
1. Safety Injection (Reactor Trip, Feedwater Isolation, Control Room Isolation, Start Diesel Generators, Containment Ventilation Isolation, Phase A Containment Isolation, Start Auxiliary Feedwater System Motor-Driven Pumps, Start Containment Fan Coolers, Start Emergency Service Water Pumps, Start Emergency Service Water Booster Pumps)					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Containment Pressure--High-1	3.64	0.71	1.5	≤ 3.0 psig	≤ 3.6 psig
d. Pressurizer Pressure--Low	18.75	10.47	1.5	≥ 1850 psig	≥ 1838 psig
e. Steam Line Pressure--Low	4.52	0.71	2.0	≥ 601 psig	≥ 581.5 psig
2. Containment Spray					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Containment Pressure--High-3	3.64	0.71	1.5	≤ 10.0 psig	≤ 11.0 psig

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

<u>FUNCTIONAL UNIT</u>	<u>TOTAL ALLOWANCE (TA)</u>	<u>Z</u>	<u>SENSOR ERROR (S)</u>	<u>TRIP SETPOINT</u>	<u>ALLOWABLE VALUE</u>
3. Containment Isolation (Continued)					
3) Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.				
4) Containment Radioactivity					
a) Area Monitors (both preentry and normal purges)	See Table 3.3-6, Item 1.a, for trip setpoint.				
b) Airborne Gaseous Radioactivity					
(1) RCS Leak Detection (normal purge)	See Table 3.3-6, Item 1.b.1, for trip setpoint.				
(2) Preentry Purge Detector	See Table 3.3-6, Item 1.b.2, for trip setpoint.				
c) Airborne Particulate Radioactivity					
(1) RCS Leak Detection (normal purge)	See Table 3.3-6, Item 1.c.1, for trip setpoint.				
(2) Preentry Purge Detector	See Table 3.3-6, Item 1.c.2, for trip setpoint.				
5) Manual Phase "A" Isolation	N.A.	N.A.	N.A.	N.A.	N.A.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

<u>FUNCTIONAL UNIT</u>	<u>TOTAL ALLOWANCE (TA)</u>	<u>Z</u>	<u>SENSOR ERROR (S)</u>	<u>TRIP SETPOINT</u>	<u>ALLOWABLE VALUE</u>
4. Main Steam Line Isolation					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Containment Pressure--High-2	3.64	0.71	1.5	≤ 3.0 psig	≤ 3.6 psig
d. Steam Line Pressure--Low	See Item 1.e. above for Steam Line Pressure--Low Trip Setpoints and Allowable Values.				
e. Negative Steam Line Pressure Rate--High	2.3	0.5	0	≤ 100 psi [#]	≤ 119.5 psi ^{###}
5. Turbine Trip and Feedwater Isolation					
a. Automatic Actuation Logic Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

<u>FUNCTIONAL UNIT</u>	<u>TOTAL ALLOWANCE (TA)</u>	<u>Z</u>	<u>SENSOR ERROR (S)</u>	<u>TRIP SETPOINT</u>	<u>ALLOWABLE VALUE</u>
5. Turbine Trip and Feedwater Isolation (Continued)					
b. Steam Generator Water Level--High-High (P-14)	22.0	8.05	2.0	≤ 78.0% of narrow range instrument span.	≤ 79.5% of narrow range instrument span.
c. Safety Injection	See Item 1. above for Safety Injection Trip Setpoints and Allowable Values.				
6. Auxiliary Feedwater					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Steam Generator Water Level--Low-Low	25.0	16.85	2.0	≥ 25.0% of narrow range instrument span.	≥ 23.5% of narrow range instrument span.
d. Safety Injection Start Motor-Driven Pumps	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.				
e. Loss-of-Offsite Power Start Motor-Driven Pumps and Turbine-Driven Pump	See Item 9. below for all Loss-of-Offsite Trip Setpoint and Allowable Values.				
f. Trip of All Main Feedwater Pumps Start Motor-Driven Pumps	N.A.	N.A.	N.A.	N.A.	N.A.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

<u>FUNCTIONAL UNIT</u>	<u>TOTAL ALLOWANCE (TA)</u>	<u>Z</u>	<u>SENSOR ERROR (S)</u>	<u>TRIP SETPOINT</u>	<u>ALLOWABLE VALUE</u>
6. Auxiliary Feedwater (Continued)					
g. Steam Line Differential Pressure--High	5.0	0.87	3.0	≤ 100 psi	≤ 127.4 psi
Coincident With Main Steam Line Isolation (Causes A/W Isolation)	See Item 4. above for Main Steam Line Isolation Trip Setpoints and Allowable Values.				
7. Safety Injection Switchover to Containment Sump					
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
b. RWST Level--Low-Low	N.A.	N.A.	N.A.	≥ 23.4%	≥ 20.4%
Coincident With Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.				
8. Containment Spray Switchover to Containment Sump					
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
b. RWST--Low-Low	See Item 7.b. above for all RWST--Low-Low Trip Setpoints and Allowable Values.				
Coincident With Containment Spray	See Item 2. above for all Containment Spray Trip Setpoints and Allowable Values.				

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

<u>FUNCTIONAL UNIT</u>	<u>TOTAL ALLOWANCE (TA)</u>	<u>Z</u>	<u>SENSOR ERROR (S)</u>	<u>TRIP SETPOINT</u>	<u>ALLOWABLE VALUE</u>
9. Loss-of-Offsite Power					
a. 6.9 kV Emergency Bus Undervoltage--Primary	N.A.	N.A.	N.A.	≥ 4830 volts with a ≤ 1.0 second time delay.	≥ 4692 volts with a time delay ≤ 1.5 seconds
b. 6.9 kV Emergency Bus Undervoltage--Secondary	N.A.	N.A.	N.A.	≥ 6420 volts with a ≤ 16 second time delay (with Safety Injection). ≥ 6420 volts with a ≤ 54.0 second time delay (without Safety Injection).	≥ 6392 volts with a time delay ≤ 18 seconds (with Safety Injection). ≥ 6392 volts with a ≤ 60 second time delay (without Safety Injection).
10. Engineered Safety Features Actuation System Interlocks					
a. Pressurizer Pressure, P-11	N.A.	N.A.	N.A.	≥ 2000 psig	≥ 1988 psig
Not P-11	N.A.	N.A.	N.A.	≤ 2000 psig	≤ 2012 psig
b. Low-Low T _{avg} , P-12	N.A.	N.A.	N.A.	≥ 553°F	≥ 549.3°F

REACTOR COOLANT SYSTEM

HOT STANDBY

LIMITING CONDITION FOR OPERATION

3.4.1.2 At least two of the reactor coolant loops listed below shall be OPERABLE with two reactor coolant pumps in operation when the Reactor Trip System breakers are closed or with one reactor coolant pump in operation when the Reactor Trip System breakers are open:

- 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.

APPLICABILITY: MODE 3.

ACTION:

- a. With less than the above required reactor coolant loops OPERABLE, restore the required loops to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.
- b. With only one reactor coolant loop in operation and the Reactor Trip System breakers in the closed position, within 1 hour open the Reactor Trip System breakers.
- c. With no reactor coolant loop in operation, immediately open the Reactor Trip System breakers, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required reactor coolant loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.2.1 At least the above required reactor coolant pumps, if not in operation, shall be determined OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

All reactor coolant 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.

REACTOR COOLANT SYSTEM

HOT STANDBY

SURVEILLANCE REQUIREMENTS (Continued)

4.4.1.2.2 The required steam generators shall be determined OPERABLE by verifying narrow range secondary side water level to be greater than or equal to 30% at least once per 12 hours.

4.4.1.2.3 The required reactor coolant loops shall be verified in operation and circulating reactor coolant at least once per 12 hours.

REACTOR COOLANT SYSTEM

HOT SHUTDOWN

SURVEILLANCE REQUIREMENTS

4.4.1.3.1 The required reactor coolant pump(s), if not in operation, shall be determined OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.3.2 The required steam generator(s) shall be determined OPERABLE by verifying wide range (WR) secondary side water level is greater than 74% or, if WR channel is inoperable, narrow range (NR) secondary side water level is greater than 30% at least once per 12 hours.

4.4.1.3.3 At least one reactor coolant or RHR loop shall be verified in operation and circulating reactor coolant at least once per 12 hours.

REACTOR COOLANT SYSTEM

COLD SHUTDOWN - LOOPS FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.4.1 At least one residual heat removal (RHR) loop shall be OPERABLE and in operation, and either:

- a. One additional RHR loop shall be OPERABLE**, or
- b. The wide range (WR) secondary side water level of at least two steam generators shall be greater than 74% or, if WR channel is inoperable, narrow range (NR) secondary side water level shall be greater than 30%.

APPLICABILITY: MODE 5 with reactor coolant loops filled***.

ACTION:

- a. With one of the RHR loops inoperable and with less than the required steam generator water level, immediately initiate corrective action to return the inoperable RHR loop to OPERABLE status or restore the required steam generator water level as soon as possible.
- b. With no RHR 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 RHR loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.4.1.1 The secondary side water level of at least two steam generators when required shall be determined to be within limits at least once per 12 hours.

4.4.1.4.1.2 At least one RHR loop shall be determined to be in operation and circulating reactor coolant at least once per 12 hours.

*The RHR pump 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.

**One RHR loop may be inoperable for up to 2 hours for surveillance testing provided the other RHR loop is OPERABLE and in operation.

***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 325°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.

3/4.4.5 STEAM GENERATORS

LIMITING CONDITION FOR OPERATION

3.4.5 Each steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one or more steam generators inoperable, restore the inoperable generator(s) to OPERABLE status prior to increasing T_{avg} above 200°F.

SURVEILLANCE REQUIREMENTS

4.4.5.0 Each steam generator shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

4.4.5.1 Steam Generator Sample Selection and Inspection - Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 4.4-1.

4.4.5.2 Steam Generator Tube Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 4.4-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.4.5.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.4.5.4. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

- a. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas;
- b. The first sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:
 1. All nonplugged tubes that previously had detectable wall penetrations (greater than 20%),
 2. Tubes in those areas where experience has indicated potential problems, and

REACTOR COOLANT SYSTEM

STEAM GENERATORS

SURVEILLANCE REQUIREMENTS (Continued)

4.4.5.2 (Continued)

3. A tube inspection (pursuant to Specification 4.4.5.4a.8) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.
- c. The tubes selected as the second and third samples (if required by Table 4.4-2) during each inservice inspection may be subjected to a partial tube inspection provided:
 1. The tubes selected for these samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found, and
 2. The inspections include those portions of the tubes where imperfections were previously found.

The results of each sample inspection shall be classified into one of the following three categories:

<u>Category</u>	<u>Inspection Results</u>
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
C-2	One or more tubes, but not more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.
C-3	More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

Note: In all inspections, previously degraded tubes must exhibit significant (greater than 10%) further wall penetrations to be included in the above percentage calculations.

STEAM GENERATORS

SURVEILLANCE REQUIREMENTS (Continued)

4.4.5.3 Inspection Frequencies - The above required inservice inspections of steam generator tubes shall be performed at the following frequencies:

- a. The first inservice inspection shall be performed after 6 Effective Full Power Months but within 24 calendar months of Steam Generator Replacement. 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;
- b. If the results of the inservice inspection of a steam generator conducted in accordance with Table 4.4-2 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 4.4.5.3a.; the interval may then be extended to a maximum of once per 40 months; and
- c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 4.4-2 during the shutdown subsequent to any of the following conditions:
 1. Reactor-to-secondary tubes leak (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.6.2, or
 2. A seismic occurrence greater than the Operating Basis Earthquake, or
 3. A loss-of-coolant accident requiring actuation of the Engineered Safety Features, or
 4. A main steam line or feedwater line break.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.5.4 Acceptance Criteria

a. As used in this specification:

1. Imperfection means an exception to the dimensions, finish, or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections;
2. Degradation means a service-induced cracking, wastage, wear, or general corrosion occurring on either inside or outside of a tube;
3. Degraded Tube means a tube containing imperfections greater than or equal to 20% of the nominal wall thickness caused by degradation;
4. % Degradation means the percentage of the tube wall thickness affected or removed by degradation;
5. Defect means an imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective;
6. Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service and is equal to 40% of the nominal tube wall thickness.
7. Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in Specification 4.4.5.3c., above;
8. Tube Inspection means an inspection of the steam generator tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg.
9. Preservice Inspection means an inspection of the full length of each tube in each steam generator performed by eddy current techniques prior to service to establish a baseline condition of the tubing. This inspection shall be performed prior to POWER OPERATION with the replacement of steam generators using equipment and techniques expected to be used during subsequent inservice inspections.

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) required by Table 4.4-2.

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 pursuant to Specification 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.

TABLE 4.4-2
STEAM GENERATOR TUBE INSPECTION

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 S.G.	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug defective tubes and inspect additional 2S tubes in this S.G.	C-1	None	N/A	N/A
			C-2	Plug defective tubes and inspect additional 4S tubes in this S.G.	C-1	None
			C-3	Perform action for C-3 result of first sample	C-2	Plug defective tubes
					C-3	Perform action for C-3 result of first sample
	C-3	Inspect all tubes in this S.G., plug defective tubes and inspect 2S tubes in each other S.G. Notification to NRC pursuant to Specification 4.4.5.5.c.	All other S.G.s are C-1	None	N/A	N/A
					Some S.G.s C-2 but no additional S.G.s are C-3	Perform action for C-2 result of second sample
			Additional S.G. is C-3	Inspect all tubes in each S.G. and plug defective tubes. Notification to NRC pursuant to Specification 4.4.5.5.c.		

S = $\frac{g}{n}$ % where n is the number of steam generators inspected during an inspection.

TABLE 4.4-2B Deleted

TABLE 4.4-2C Deleted

REACTOR COOLANT SYSTEM

3/4.4.8 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.4.8 The specific activity of the reactor coolant shall be limited to:

- a. Less than or equal to 1 microCurie per gram DOSE EQUIVALENT I-131, and
- b. Less than or equal to $100/E$ microCuries per gram of gross radioactivity.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTION:

MODES 1, 2 and 3*:

- a. With the specific activity of the reactor coolant greater than 1 microCurie per gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding 60.0 microCurie per gram DOSE EQUIVALENT I-131, be in at least HOT STANDBY with T_{avg} less than 500°F within 6 hours. The provisions of Specification 3.0.4 are not applicable.
- b. With the specific activity of the reactor coolant greater than $100/E$ microCuries per gram, be in at least HOT STANDBY with T_{avg} less than 500°F within 6 hours.

MODES 1, 2, 3, 4, and 5:

With the specific activity of the reactor coolant greater than 1 microcurie per gram DOSE EQUIVALENT I-131 or greater than $100/E$ microCuries per gram, perform the sampling and analysis requirements of Item 4.a) of Table 4.4-4 until the specific activity of the reactor coolant is restored to within its limits.

SURVEILLANCE REQUIREMENTS

4.4.8 The specific activity of the reactor coolant shall be determined to be within the limits by performance of the sampling and analysis program of Table 4.4-4.

*With T_{avg} greater than or equal to 500°F.

FIGURE 3.4-1 Deleted

TABLE 4.4-4
REACTOR COOLANT SPECIFIC ACTIVITY SAMPLE
AND ANALYSIS PROGRAM

<u>TYPE OF MEASUREMENT AND ANALYSIS</u>	<u>SAMPLE AND ANALYSIS FREQUENCY</u>	<u>MODES IN WHICH SAMPLE AND ANALYSIS REQUIRED</u>
1. Gross Radioactivity Determination	At least once per 72 hours.	1, 2, 3, 4
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	1 per 14 days.	1
3. Radiochemical for E Determination	1 per 6 months**	1
4. Isotopic Analysis for Iodine Including I-131, I-133, and I-135	a. Once per 4 hours, whenever the specific activity exceeds 0.35 $\mu\text{Ci}/\text{gram}$ DOSE EQUIVALENT I-131 or 100/E $\mu\text{Ci}/\text{gram}$ of gross radioactivity, and	1#, 2#, 3#, 4#, 5#
	b. One sample between 2 and 6 hours following a THERMAL POWER change exceeding 15% of the RATED THERMAL POWER within a 1-hour period.	1, 2, 3

TABLE 3.7-1

MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH
INOPERABLE STEAM LINE SAFETY VALVES DURING 3 LOOP OPERATION

MAXIMUM NUMBER OF INOPERABLE
SAFETY VALVES ON ANY
OPERATING STEAM GENERATOR

MAXIMUM ALLOWABLE POWER RANGE
NEUTRON FLUX HIGH SETPOINT
(PERCENT OF RATED THERMAL POWER)

1	50
2	33
3	16

SPECIAL TEST EXCEPTIONS

3/4.10.4 REACTOR COOLANT LOOPS

LIMITING CONDITION FOR OPERATION

3.10.4 The limitations of Specification 3.4.1.1 may be suspended during the performance of startup and PHYSICS TESTS in MODE 1 or 2 provided:

- a. The THERMAL POWER does not exceed the P-7 Interlock Setpoint, and
- b. The Reactor Trip Setpoints on the OPERABLE Intermediate and Power Range channels are set less than or equal to 25% of RATED THERMAL POWER.

APPLICABILITY: During operation below the P-7 Interlock Setpoint.

ACTION:

With the THERMAL POWER greater than the P-7 Interlock Setpoint, immediately open the reactor trip breakers.

SURVEILLANCE REQUIREMENTS

4.10.4.1 The THERMAL POWER shall be determined to be less than P-7 Interlock Setpoint at least once per hour during startup and PHYSICS TESTS.

4.10.4.2 Each Intermediate and Power Range channel, and P-7 Interlock shall be subjected to an ANALOG CHANNEL OPERATIONAL TEST within 12 hours prior to initiating startup and PHYSICS TESTS.

REACTIVITY CONTROL SYSTEMS

BASES

BORATION SYSTEMS (Continued)

condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity changes in the event the single boron injection flow path becomes inoperable.

The limitation for a maximum of one charging/safety injection pump (CSIP) to be OPERABLE and the Surveillance Requirement to verify all CSIPs except the required OPERABLE pump to be inoperable below 325°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

The boron capability required below 200°F is sufficient to provide the required SHUTDOWN MARGIN as defined by Specification 3/4.1.1.2 after xenon decay and cooldown from 200°F to 140°F. This condition requires either 7150 gallons of 7000 ppm borated water be maintained in the boric acid storage tanks or 106,000 gallons of 2400-2600 ppm borated water be maintained in the RWST.

The gallons given above are the amounts that need to be maintained in the tank in the various circumstances. To get the specified indicated levels used for surveillance testing, each value had added to it an allowance for the unusable volume of water in the tank, allowances for other identified needs, and an allowance for possible instrument error. In addition, for human factors purposes, the percent indicated levels were then raised to either the next whole percent or the next even percent and the gallon figures rounded off. This makes the LCO values conservative to the analyzed values.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 8.5 and 11.0 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

The BAT minimum temperature of 65°F ensures that boron solubility is maintained for concentrations of at least the 7750 ppm limit. The RWST minimum temperature is consistent with the STS value and is based upon other considerations since solubility is not an issue at the specified concentration levels. The RWST high temperature was selected to be consistent with analytical assumptions for containment heat load.

The OPERABILITY of one Boron Injection System during REFUELING ensures that this system is available for reactivity control while in MODE 6.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that: (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) the potential effects of rod misalignment on associated accident analyses are limited. OPERABILITY of the control rod position indicators is required to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits.

POWER DISTRIBUTION LIMITS

BASES

3/4.2.4 QUADRANT POWER TILT RATIO

The QUADRANT POWER TILT RATIO limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during STARTUP testing and periodically during power operation.

The limit of 1.02, at which corrective action is required, provides DNB and linear heat generation rate protection with x-y plane power tilts. A limiting tilt of 1.025 can be tolerated before the margin for uncertainty in F_Q is depleted. A limit of 1.02 was selected to provide an allowance for the uncertainty associated with the indicated power tilt.

The 2-hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned control rod. In the event such action does not correct the tilt, the margin for uncertainty on F_Q is reinstated by reducing the maximum allowed power by 3% for each percent of tilt in excess of 1.

For purposes of monitoring QUADRANT POWER TILT RATIO when one excore detector is inoperable, the movable 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 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 that is equal to or greater than the design DNBR value throughout each analyzed transient. The indicated T_{avg} value and the indicated pressurizer pressure value are compared to analytical limits of 594.8°F and 2185 psig, respectively, after an allowance for measurement uncertainty is included.

When RCS flow rate is measured, an additional allowance is necessary prior to comparison with the limit of Specification 3.2.5.c. Specifically for the precision calorimetric heat balance, a normal RCS flow rate error of 2.1% will be included.

Potential fouling of the feedwater venturi, which might not be detected, could bias the result from the precision heat balance in a non-conservative manner. Therefore, a penalty of 0.1% for undetected fouling of the feedwater venturi, raises the nominal flow measurement allowance to 2.2% for no venturi fouling. Any fouling which might bias the RCS flow rate measurement greater than 0.1% can be detected by monitoring and trending various plant parameters.

BASES

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with all reactor coolant loops in operation and maintain DNBR above the design DNBR value during all normal operations and anticipated transients. In MODES 1 and 2 with one reactor coolant loop not in operation this specification requires that the plant be in at least HOT STANDBY within 6 hours.

In MODE 3, two reactor coolant loops provide sufficient heat removal capability for removing core decay heat even in the event of a bank withdrawal accident; however, a single reactor coolant loop provides sufficient heat removal capacity if a bank withdrawal accident can be prevented, i.e., by opening the Reactor Trip System breakers. Single failure considerations require that two loops be OPERABLE at all times.

In MODE 4, and in MODE 5 with reactor coolant loops filled, a single reactor coolant loop or RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops (either RHR or RCS) be OPERABLE.

Surveillance Requirements for MODES 3, 4, and 5 with reactor coolant loops filled require verification of steam generator (SG) OPERABILITY. Verification of adequate level in the applicable steam generator ensures an adequate heat sink for the removal of decay heat. If the SG tubes become uncovered, the associated loop may not be capable of providing the heat sink for the removal of the decay heat. The level values include allowances for channel uncertainty and process measurement effects. The 12 hour frequency is considered adequate in view of other indications available in the control room to alert the operator to a loss of SG level.

In MODE 5 with reactor coolant loops not filled, a single RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations, and the unavailability of the steam generators as a heat removing component, require that at least two RHR loops be OPERABLE.

The operation of one reactor coolant pump (RCP) or one RHR pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reduction will, therefore, be within the capability of operator recognition and control.

The restrictions on starting an RCP with one or more RCS cold legs less than or equal to 325°F are provided to prevent RCS pressure transients, caused by energy additions from the Secondary Coolant System, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

3/4.4.2 SAFETY VALVES

The pressurizer Code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 380,000 lbs per hour of saturated steam at the valve Setpoint. The relief capacity of a single safety valve is adequate to relieve any

SAFETY VALVES (Continued)

overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating RHR loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization. In addition, the Overpressure Protection System provides a diverse means of protection against RCS overpressurization at low temperatures.

During operation, all pressurizer Code safety valves must be OPERABLE to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from a complete loss-of-load assuming no reactor trip until the second Reactor Trip System trip setpoint is reached (i.e., no credit is taken for a direct Reactor trip on the loss-of-load) and also assuming no operation of the power-operated relief valves or steam dump valves.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Code.

3/4.4.3 PRESSURIZER

The limit on the maximum water volume in the pressurizer assures that the parameter is maintained within the normal steady-state envelope of operation assumed in the SAR. The limit is consistent with the initial SAR assumptions. The 12-hour periodic surveillance is sufficient to ensure that the parameter is restored to within its limit following expected transient operation. The maximum water volume also ensures that a steam bubble is formed and thus the RCS is not a hydraulically solid system. The requirement that a minimum number of pressurizer heaters be OPERABLE enhances the capability of the plant to control Reactor Coolant System pressure and establish natural circulation.

3/4.4.4 RELIEF VALVES

In MODES 1, 2, and 3 the power-operated relief valves (PORVs) provide an RCS pressure boundary, manual RCS pressure control for mitigation of accidents, and automatic RCS pressure relief to minimize challenges to the safety valves.

Providing an RCS pressure boundary and manual RCS pressure control for mitigation of a steam generator tube rupture (SGTR) are the safety-related functions of the PORVs in MODES 1, 2, and 3. The capability of the PORV to perform its function of providing an RCS pressure boundary requires that the PORV or its associated block valve is closed. The capability of the PORV to perform manual RCS pressure control for mitigation of a SGTR accident is based on manual actuation and does not require the automatic RCS pressure control function. The automatic RCS pressure control function of the PORVs is not a safety-related function in MODES 1, 2, and 3. The automatic pressure control function limits the number of challenges to the safety valves, but the safety valves perform the safety function of RCS overpressure protection. Therefore, the automatic RCS pressure control function of the PORVs does not have to be available for the PORVs to be operable.

STEAM GENERATORS (Continued)

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the Reactor Coolant System and the Secondary Coolant System (reactor-to-secondary leakage = 150 gallons per day per steam generator). Cracks having a reactor-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that reactor-to-secondary leakage of 150 gallons per day per steam generator can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging will be required for all tubes with imperfections exceeding the plugging limit of 40% of the tube nominal wall thickness. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be reported to the Commission in a Special Report pursuant to Specification 4.4.5.5.c within 30 days and prior to resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

The RCS Leakage Detection Systems required by this specification are provided to monitor and detect leakage from the reactor coolant pressure boundary. These Detection Systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.

3/4.4.6.2 OPERATIONAL LEAKAGE

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDARY LEAKAGE requires the unit to be promptly placed in COLD SHUTDOWN.

CHEMISTRY (Continued)

that operation may be continued with contaminant concentration levels in excess of the Steady-State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady-State Limits.

The Surveillance Requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.4.8 SPECIFIC ACTIVITY

The limitations on the specific activity of the reactor coolant ensure that the resulting 2-hour doses at the SITE BOUNDARY will not exceed an appropriately small fraction of 10 CFR Part 100 dose guideline values following a steam generator tube rupture accident in conjunction with an assumed steady-state reactor-to-secondary steam generator leakage rate of 1 gpm. The values for the limits on specific activity represent limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the SHEARON HARRIS site, such as SITE BOUNDARY location and meteorological conditions, were not considered in this evaluation.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the reactor coolant's specific activity greater than 0.35 microCurie/gram DOSE EQUIVALENT I-131, but less than 60.0 microCurie/gram DOSE EQUIVALENT I-131, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER. See Generic Letter 85-19 for additional information.

The sample analysis for determining the gross specific activity and \bar{E} can exclude the radioiodines because of the low reactor coolant limit of 0.35 microCurie/gram DOSE EQUIVALENT I-131, and because, if the limit is exceeded, the radioiodine level is to be determined every 4 hours. If the gross specific activity level and radioiodine level in the reactor coolant were at their limits, the radioiodine contribution would be approximately 1%. In a release of reactor coolant with a typical mixture of radioactivity, the actual radioiodine contribution would probably be about 20%. The exclusion of radionuclides with half-lives less than 15 minutes from these determinations has been made for several reasons. The first consideration is the difficulty to identify short-lived radionuclides in a sample that requires a significant time to collect, transport, and analyze. The second consideration is the predictable delay time between the postulated release of radioactivity from the reactor coolant to its release to the environment and transport to the SITE BOUNDARY, which is relatable to at least 30 minutes decay time. The choice of 15 minutes for the half-life cutoff was made because of the nuclear characteristics of the typical reactor coolant radioactivity. The radionuclides in the typical reactor coolant have half-lives of less than 4 minutes or half-lives of greater than 14 minutes, which allows a

3/4.6 CONTAINMENT SYSTEMS

BASES

CONTAINMENT AIR LOCKS (Continued)

Maintaining containment air locks OPERABLE requires compliance with the leakage rate test requirements of 10 CFR 50, Appendix J, as modified by approved exemptions. HNP has an approved exemption to Appendix J Option A, paragraph III.D.2 of 10 CFR 50 in that the Overall air lock leakage test is required to be performed if maintenance has been performed that could affect the air lock sealing capability prior to establishing CONTAINMENT INTEGRITY. This is in contrast to the Appendix J requirement if air locks are opened during periods when containment integrity is not required by the plant's Technical Specifications shall be tested at the end of such periods.

3/4.6.1.4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that: (1) the containment structure is prevented from exceeding its design negative pressure differential with respect to the outside atmosphere of -2 psig, and (2) the containment peak pressure does not exceed the design pressure of 45 psig.

The maximum peak pressure expected to be obtained from a postulated LOCA is 41.8 psig using a value of 1.6 psig for initial positive containment pressure. The -1" wg was chosen to be consistent with the initial assumptions of the accident analyses.

3/4.6.1.5 AIR TEMPERATURE

The limitations on containment average air temperature ensure that the overall containment average air temperature does not exceed the initial temperature condition assumed in the safety analysis for a LOCA or steam line break accident. Measurements shall be made at all listed locations, whether by fixed or portable instruments, prior to determining the average air temperature.

3/4.6.1.6 CONTAINMENT STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the containment will withstand the maximum pressure of a postulated LOCA (41.8 psig). A visual inspection in conjunction with the Containment Leakage Rate Testing Program is sufficient to demonstrate this capability.

3/4.6.1.7 CONTAINMENT VENTILATION SYSTEM

The 42-inch containment preentry purge makeup and exhaust isolation valves are required to be sealed closed during plant operations in MODES 1, 2, 3 and 4 since these valves have not been demonstrated capable of closing during a LOCA or steam line break accident. Maintaining these valves sealed closed during these MODES ensures that excessive quantities of radioactive materials will not be released via the Pre-entry Containment Purge System. To provide assurance that these containment valves cannot be inadvertently opened, the valves are sealed closed in accordance with Standard Review Plan 6.2.4 which includes mechanical devices to seal or lock the valve closed, or prevents power from being supplied to the valve operator.

The use of the Normal Containment Purge System is restricted to the 8-inch purge makeup and exhaust isolation valves since, unlike the 42-inch valves, the 8-inch valves are capable of closing during a LOCA or steam line break accident. Therefore, the SITE BOUNDARY dose guideline of 10 CFR Part 100 would not be exceeded in the event of an accident during normal containment PURGING operation. The total time the Normal Containment Purge System isolation valves may be open during MODES 1, 2, 3, and 4 in a calendar year is a function of anticipated need and operating experience. Only safety-related reasons; e.g., containment pressure control or the reduction of airborne radioactivity to facilitate personnel access for surveillance and maintenance activities, may be used to justify the opening of these isolation valves during MODES 1, 2, 3, and 4.

Leakage integrity tests with a maximum allowable leakage rate for containment purge makeup and exhaust supply valves will provide early indication of resilient material seal degradation and will allow opportunity for repair before

BASES

3/4.7.1 TURBINE CYCLE3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line Code safety valves ensures that the Secondary System pressure will be limited to within 110% (1305 psig) of its design pressure of 1185 psig during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a Turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, 1971 Edition. The total relieving capacity for all valves on all of the steam lines is 1.36×10^7 lbs/h which is in excess of 105% of the maximum calculated steam flow of 12.9×10^6 lbs/h at 100% RATED THERMAL POWER. A minimum of two OPERABLE safety valves per steam generator ensures that sufficient relieving capacity is available for the allowable THERMAL POWER restriction in Table 3.7-1.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in Secondary Coolant System steam flow and THERMAL POWER required by the reduced Reactor trip settings of the Power Range Neutron Flux channels. The Reactor Trip Setpoint reductions are derived on the following bases:

For 3 loop operation

$$Hi\phi = (100/Q) \frac{(w_s h_{fg} N)}{K}$$

Where:

- Hi ϕ = Safety Analysis power range high neutron flux setpoint, percent
- Q = Nominal NSSS power rating of the plant (including reactor coolant pump heat), Mwt
- K = Conversion factor, $947.82 \frac{\text{Btu/sec}}{\text{Mwt}}$
- w_s = Minimum total steam flow rate capability of the operable MSSVs or any one steam generator at the highest MSSV opening pressure including tolerance and accumulation, as appropriate, in lb/sec.
- h_{fg} = Heat of vaporization for steam at the highest MSSV opening pressure including tolerance and accumulation, as appropriate, Btu/lbm
- N = Number of loops in plant

The values from this algorithm must then be adjusted lower to account for instrument and channel uncertainties. This adjustment will be 9% power.

AUXILIARY FEEDWATER SYSTEM

operation. The AFW System provides decay heat removal immediately following a station blackout event, and is required to mitigate the Loss of Normal Feedwater and Feedwater Line break accidents analyzed in FSAR Chapter 15. The minimum pump performance requirements are based upon a maximum allowable degradation of the pump performance curves. Pump operation at this level has been demonstrated by calculation to deliver sufficient AFW flow to satisfy the accident analysis acceptance criteria.

With regard to the periodic AFW valve position verification of Surveillance Requirement 4.7.1.2.1 Sub-paragraph b.1, this requirement does not include in its scope the AFW flow control valves inline from the AFW motor-driven pump discharge header to each steam generator when they are equipped with an auto-open feature. The auto-open logic feature is designed to automatically open these valves upon receipt of an Engineered Safety Features System AFW start signal. As a consequence, valves with an auto-open feature do not have a "correct position" which must be verified. The valves may be in any position, in any MODE of operation thereby allowing full use of the AFW System for activities such as to adjust steam generator water levels prior to and during plant start-up, as an alternate feedwater system during hot standby, for cooldown operations, and to establish and maintain wet layup conditions in the steam generators.

3/4.7.1.3 CONDENSATE STORAGE TANK

The OPERABILITY of the condensate storage tank with the minimum water volume ensures that sufficient water is available to maintain the RCS at HOT STANDBY conditions for 6 hours with steam discharge to the atmosphere concurrent with total loss-of-offsite power. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics, and the value has also been adjusted in a manner similar to that for the RWST and BAT, as discussed on page B 3/4 1-3.

3/4.7.1.4 SPECIFIC ACTIVITY

The limitations on Secondary Coolant System specific activity ensure that the resultant offsite radiation dose will be limited to a small fraction of 10 CFR Part 100 dose guideline values in the event of a steam line rupture. This dose also includes the effects of a coincident 1 gpm reactor-to-secondary tube leak in the steam generator of the affected steam line. These values are consistent with the assumptions used in the safety analyses.

3/4.7.1.5 MAIN STEAM LINE ISOLATION VALVES

The OPERABILITY of the main steam line isolation valves ensures that no more than one steam generator will blow down in the event of a steam line rupture. This restriction is required to: (1) minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and (2) limit the pressure rise within containment in the event the steam line rupture occurs within containment. The OPERABILITY of the main steam isolation valves within the closure times of the Surveillance Requirements are consistent with the assumptions used in the safety analyses.

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

The limitation on steam generator pressure and temperature ensures that the pressure-induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The limitations of 70°F and 200 psig are based on a steam generator RT_{NDT} of 60°F (a generic maximum) and are sufficient to prevent brittle fracture. The Shearon Harris specific RT_{NDT} is limited to a maximum value of 10°F.

DESIGN FEATURES

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The containment building is designed and shall be maintained for a maximum internal pressure of 45.0 psig and a peak air temperature of 380°F.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The core shall contain 157 fuel assemblies with each fuel assembly normally containing 264 fuel rods clad with Zircaloy-4 except that limited substitution of fuel rods by filler rods consisting of Zircaloy-4, stainless steel, or by vacancies may be made in fuel assemblies if justified by a cycle-specific evaluation. Should more than a total of 30 fuel rods or more than 10 fuel rods in any one assembly be replaced per refueling a Special Report describing the number of rods replaced will be submitted to the Commission, pursuant to Specification 6.9.2, within 30 days after cycle startup. Each fuel rod shall have a nominal active fuel length of 144 inches. The initial core loading shall have a maximum enrichment of 3.5 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 5.0 weight percent U-235. Fuel with enrichments greater than 4.20 weight percent U-235 shall contain sufficient integral burnable absorbers such that the requirement of Specification 5.6.1.1.b is met.

CONTROL ROD ASSEMBLIES

5.3.2 The core shall contain 52 shutdown and control rod assemblies. The shutdown and rod assemblies shall contain a nominal 142 inches of absorber material. The nominal values of absorber material shall be 80% silver, 15% indium, and 5% cadmium, or 95% hafnium with the remainder zirconium. All control rods shall be clad with stainless steel tubing.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

- 5.4.1 The Reactor Coolant System is designed and shall be maintained:
- In accordance with the Code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements.
 - For a pressure of 2485 psig, and
 - For a temperature of 650°F, except for the pressurizer which is 680°F.

VOLUME

5.4.2 The total water and steam volume of the Reactor Coolant System is approximately 10,300 cubic feet at a nominal T_{avg} of 588.8°F.

PROCEDURES AND PROGRAMS (Continued)

k. Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54 (o) and 10 CFR 50 Appendix J, Option B, as modified by approved exemptions. This program shall be in conformance with the NRC Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, with the following exception noted:

- 1) The above Containment Leakage Rate Testing Program is only applicable to Type A testing. Type B and C testing shall continue to be conducted in accordance with the original commitment to 10 CFR 50 Appendix J, Option A.

The calculated peak containment internal pressure related to the design basis loss-of-coolant accident is 41.8 psig. The calculated peak containment internal pressure related to the design basis main steam line break is 41.3 psig. P_a will be assumed to be 41.8 psig for the purpose of containment testing in accordance with this Technical Specification.

The maximum allowable containment leakage rate, L_a at P_a , shall be 0.1 % of containment air weight per day.

The containment overall leakage rate acceptance criterion is $\leq 1.0 L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60 L_a$ for the combined Type B and Type C tests, and $\leq 0.75 L_a$ for Type A tests.

The provisions of Surveillance Requirement 4.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program. However, test frequencies specified in this Program may be extended consistent with the guidance provided in Nuclear Energy Institute (NEI) 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50 Appendix J," as endorsed by Regulatory Guide 1.163. Specifically, NEI 94-01 has this provision for test frequency extension:

- 1) Consistent with standard scheduling practices for Technical Specifications Required Surveillances, intervals for recommended Type A testing may be extended by up to 15 months. This option should be used only in cases where refueling schedules have been changed to accommodate other factors.

The provisions of Surveillance Requirement 4.0.3 are applicable to the Containment Leakage Rate Testing Program.

6.9.1.6 CORE OPERATING LIMITS REPORT (Continued)

- d. XN-75-32(P)(A). "Computational Procedure for Evaluating Fuel Rod Bowing," approved version as specified in the COLR.

(Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).

- e. EMF-84-093(P)(A). "Steam Line Break Methodology for PWRs," approved version as specified in the COLR.

(Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).

- f. EMF-2087(P)(A), "SEM/PWR-98: ECCS Evaluation Model for PWR LBLOCA Applications," approved version as specified in the COLR.

XN-NF-81-58(P)(A), "RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model," approved version as specified in the COLR.

ANF-81-58(P)(A), "RODEX2 Fuel Rod Thermal Mechanical Response Evaluation Model," approved version as specified in the COLR.

XN-NF-82-06(P)(A). "Qualification of Exxon Nuclear Fuel for Extended Burnup," approved version as specified in the COLR.

ANF-88-133(P)(A), "Qualification of Advanced Nuclear Fuels' PWR Design Methodology for Rod Burnups of 62 Gwd/MTU," approved version as specified in the COLR.

XN-NF-85-92(P)(A). "Exxon Nuclear Uranium Dioxide/Gadolinia Irradiation Examination and Thermal Conductivity Results," approved version as specified in the COLR.

EMF-92-116(P)(A). "Generic Mechanical Design Criteria for PWR Fuel Designs," approved version as specified in the COLR.

(Methodologies for Specification 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).

- g. XN-NF-78-44(NP)(A). "A Generic Analysis of the Control Rod Ejection Transient for Pressurized Water Reactors," approved version as specified in the COLR.

(Methodology for Specification 3.1.3.5 - Shutdown Bank Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, and 3.2.2 - Heat Flux Hot Channel Factor).

6.9.1.6 CORE OPERATING LIMITS REPORT (Continued)

- h. ANF-88-054(P)(A). "PDC-3: Advanced Nuclear Fuels Corporation Power Distribution Control for Pressurized Water Reactors and Application of PDC-3 to H. B. Robinson Unit 2." approved version as specified in the COLR.

(Methodology for Specification 3.2.1 - Axial Flux Difference, and 3.2.2 - Heat Flux Hot Channel Factor).

- i. EMF-92-081(P)(A). "Statistical Setpoint/Transient Methodology for Westinghouse Type Reactors." approved version as specified in the COLR.

(Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).

- j. EMF-92-153(P)(A). "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel." approved version as specified in the COLR.

(Methodology for Specification 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).

- k. XN-NF-82-49(P)(A). "Exxon Nuclear Company Evaluation Model EXEM PWR Small Break Model." approved version as specified in the COLR.

(Methodology for Specification 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).

6.9.1.6 CORE OPERATING LIMITS REPORT (Continued)

1. EMF-96-029(P)(A). "Reactor Analysis Systems for PWRs." approved version as specified in the COLR.

(Methodology for Specification 3.1.1.2 - SHUTDOWN MARGIN - MODES 3, 4 and 5, 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor, and 3.9.1 - Boron Concentration).

6.9.1.6.3 The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

6.9.1.6.4 The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements, shall be provided, upon issuance for each reload cycle, to the NRC Document Control Desk, with copies to the Regional Administrator and Resident Inspector.

SPECIAL REPORTS

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

6.10 DELETED

(PAGE 6-25 DELETED)