(4) FPL Energy Seabrook, LLC, pursuant to the Act and 10 CFR 30, 40, and 70, to receive, possess, and use at any time any byproduct, source, and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(5) FPL Energy Seabrook, LLC, pursuant to the Act and 10 CFR 30, 40, and 70, to receive, possess, and use in amounts as required any byproduct, source, or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components;

(6) FPL Energy Seabrook, LLC, pursuant to the Act and 10 CFR 30, 40, and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility authorized herein; and

(7) DELETED

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; is subject to the additional conditions specified or incorporated below:

(1) <u>Maximum Power Level</u>

FPL Energy Seabrook, LLC, is authorized to operate the facility at reactor core power levels not in excess of 3648 megawatts thermal (100% of rated power).

(2) <u>Technical Specifications</u>

The Technical Specifications contained in Appendix A, as revised through Amendment No. 115 *, and the Environmental Protection Plan contained in Appendix B are incorporated into the Facility License No. NPF-86. FPL Energy Seabrook, LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) <u>License Transfer to FPL Energy Seabrook, LLC</u>

a. On the closing date(s) of the transfer of any ownership interests in Seabrook Station covered by the Order approving the transfer, FPL Energy Seabrook, LLC, shall obtain from each respective transferring owner all of the accumulated decommissioning trust funds for the facility, and ensure the deposit of such funds and additional funds, if necessary, into a decommissioning trust or trusts for Seabrook Station established by FPL Energy Seabrook, LLC, such that the amount of such funds deposited meets or exceeds the amount required under 10 CFR 50.75 with respect to the interest in Seabrook Station FPL Energy Seabrook, LLC, acquires on such dates(s).

* Implemented

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3.0/4.0 BASES

6.0 ADMINISTRATIVE CONTROLS

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of Leakage Detection Systems or not to be PRESSURE BOUNDARY LEAKAGE, or

c. Reactor Coolant System leakage through a steam generator to the Secondary Coolant System (primary to secondary leakage).

MASTER RELAY TEST

1.18 A MASTER RELAY TEST shall be the energization of each master relay and verification of OPERABILITY of each relay. The MASTER RELAY TEST shall include, a continuity check of each associated slave relay.

MEMBER(S) OF THE PUBLIC

1.19 MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the licensee, its contractors, or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational, or other purposes not associated with the plant.

OFFSITE DOSE CALCULATION MANUAL

1.20 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Section 6.7.6 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Annual Radioactive Effluent Release Reports required by Specifications 6.8.1.3 and 6.8.1.4.

OPERABLE - OPERABILITY

1.21 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s), and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its function(s) are also capable of performing their related support function(s).

OPERATIONAL MODE - MODE

1.22 An OPERATIONAL MODE (i.e., MODE) shall correspond to any one inclusive combination of core reactivity condition, power level, and average reactor coolant temperature specified in Table 1.2.

PHYSICS TESTS

1.23 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation: (1) described in Chapter 14.0 of the FSAR, (2) authorized under the provisions of 10 CFR 50.59, or (3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

1.24 PRESSURE BOUNDARY LEAKAGE shall be leakage (except primary to secondary leakage) through a nonisolable fault in a Reactor Coolant System component body, pipe wall, or vessel wall.

PROCESS CONTROL PROGRAM

1.25 The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, tests, 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, 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 3648 Mwt.

REACTOR TRIP SYSTEM (RTS) RESPONSE TIME

1.29 The RTS RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its RTS Trip Setpoint at the channel sensor until loss of stationary gripper coil voltage. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.

REACTOR COOLANT SYSTEM

3/4.4.5 STEAM GENERATORS

LIMITING CONDITION FOR OPERATION

3.4.5 SG tube integrity shall be maintained

AND

All SG tubes satisfying the tube repair criteria shall be plugged in accordance with the Steam Generator Program.

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

ACTION:

..............................NOTE.....................................
Separate action entry is allowed for each steam generator tube.
....

a. With one or more steam generator tubes satisfying the tube repair criteria and not plugged in accordance with the Steam Generator Program,

   1. Within 7 days, verify tube integrity of the affected tube(s) is maintained until the next refueling outage or SG tube inspection, or be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the next 30 hours, and

   2. Plug the affected tube(s) in accordance with the Steam Generator Program prior to entering HOT SHUTDOWN following the next refueling outage or steam generator tube inspection.

b. With steam generator tube integrity not maintained, be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.5.1 Verify SG tube integrity in accordance with the Steam Generator Program.

4.4.5.2 Verify that each inspected SG tube that satisfies the tube repair criteria is plugged in accordance with the Steam Generator Program prior to entering HOT SHUTDOWN following a SG tube insepction.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.6.1 The following Reactor Coolant System Leakage Detection Systems shall be OPERABLE:

a. The Containment Atmosphere Particulate Radioactivity Monitoring System,

b. The Containment Drainage Sump Level Monitoring System, and

c. Containment Radioactive Gas Monitor

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

ACTION:

With only two of the above required Leakage Detection Systems OPERABLE, operation may continue for up to 30 days provided grab samples of the containment atmosphere are obtained and analyzed at least once per 24 hours when the required Gaseous or Particulate Radioactive Monitoring System is inoperable; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.1 The Leakage Detection Systems shall be demonstrated OPERABLE by:

a. Containment Atmosphere Gaseous and Particulate Monitoring Systems - performance of CHANNEL CHECK, CHANNEL CALIBRATION, and CHANNEL OPERATIONAL TEST at the frequencies specified in Table 4.3-3, and

b. Containment Drainage Sump Level Monitoring System - performance of CHANNEL CALIBRATION at least once per 18 months.

REACTOR COOLANT SYSTEM LEAKAGE

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System operational leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 gpm UNIDENTIFIED LEAKAGE,

c. 150 gallons per day primary to secondary leakage through any one steam generator (SG),

d. 10 gpm IDENTIFIED LEAKAGE from the Reactor Coolant System,

e. 40 gpm CONTROLLED LEAKAGE at a Reactor Coolant System pressure of 2235 psig ± 20 psig, and

f. 0.5 gpm leakage per nominal inch of valve size up to a maximum of 5 gpm at a Reactor Coolant System pressure of 2235 ± 20 psig from any Reactor Coolant System Pressure Isolation Valve.*

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

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE or with primary to secondary leakage not within the limit, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any Reactor Coolant System operational leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE, primary to secondary leakage, and leakage from Reactor Coolant System Pressure Isolation Valves, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

*Test pressures less than 2235 psig but greater than 150 psig are allowed. Observed leakage shall be adjusted for the actual test pressure up to 2235 psig assuming the leakage to be directly proportional to pressure differential to the one-half power.

ACTION: (Continued)

c. With any Reactor Coolant System Pressure Isolation Valve leakage greater than the above limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least two closed manual or deactivated automatic valves, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.2.1 Reactor Coolant System operational leakage shall be demonstrated to be within each of the above limits by:

a. Monitoring the containment atmosphere particulate radioactivity monitor at least once per 12 hours;

b. Monitoring the containment drainage sump inventory and discharge at least once per 12 hours;

c. Measurement of the CONTROLLED LEAKAGE to the reactor coolant pump seals when the Reactor Coolant System pressure is 2235 ± 20 psig at least once per 31 days with the modulating valve fully open. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4;

d. Performance of a Reactor Coolant System water inventory balance at least once per 72 hours during steady-state operation, except that not more than 96 hours shall elapse between any two successive inventory balances; [(1)] [(2)]

e. Monitoring the Reactor Head Flange Leakoff System at least once per 24 hours, and

f. Verifying primary to secondary leakage is ≤ 150 gallons per day through any one SG at least once per 72 hours. [(2)]

---

(1) Not applicable to primary to secondary leakage.

(2) Not required to be performed until 12 hours after establishment of steady state operation.

SURVEILLANCE REQUIREMENTS

4.4.6.2.2 Each Reactor Coolant System Pressure Isolation Valve shall be demonstrated OPERABLE by verifying leakage to be within its limit:

a. At least once per 18 months,

b. Prior to entering MODE 2 whenever the plant has been in COLD SHUTDOWN for 7 days or more and if leakage testing has not been performed in the previous 9 months,

c. Prior to returning the valve to service following maintenance, repair, or replacement work on the valve, and

d. Within 24 hours following valve actuation due to automatic or manual action or flow through the valve.*

e. Testing pursuant to Specification 4.0.5.

The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4.

* Not applicable to RHR Pumps 8A and 8B suction isolation valves.

3/4.4.7 (THIS SPECIFICATION NUMBER IS NOT USED)

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/Ē 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 the limit line shown on Figure 3.4-1, be in at least HOT STANDBY with T_{avg} less than 500°F within 6 hours; and

b. With the specific activity of the reactor coolant greater than 100/Ē 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/Ē microCuries per gram, perform the sampling and analysis requirements of Item 4.a) of Table 4.4-3 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 4.4-3.

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

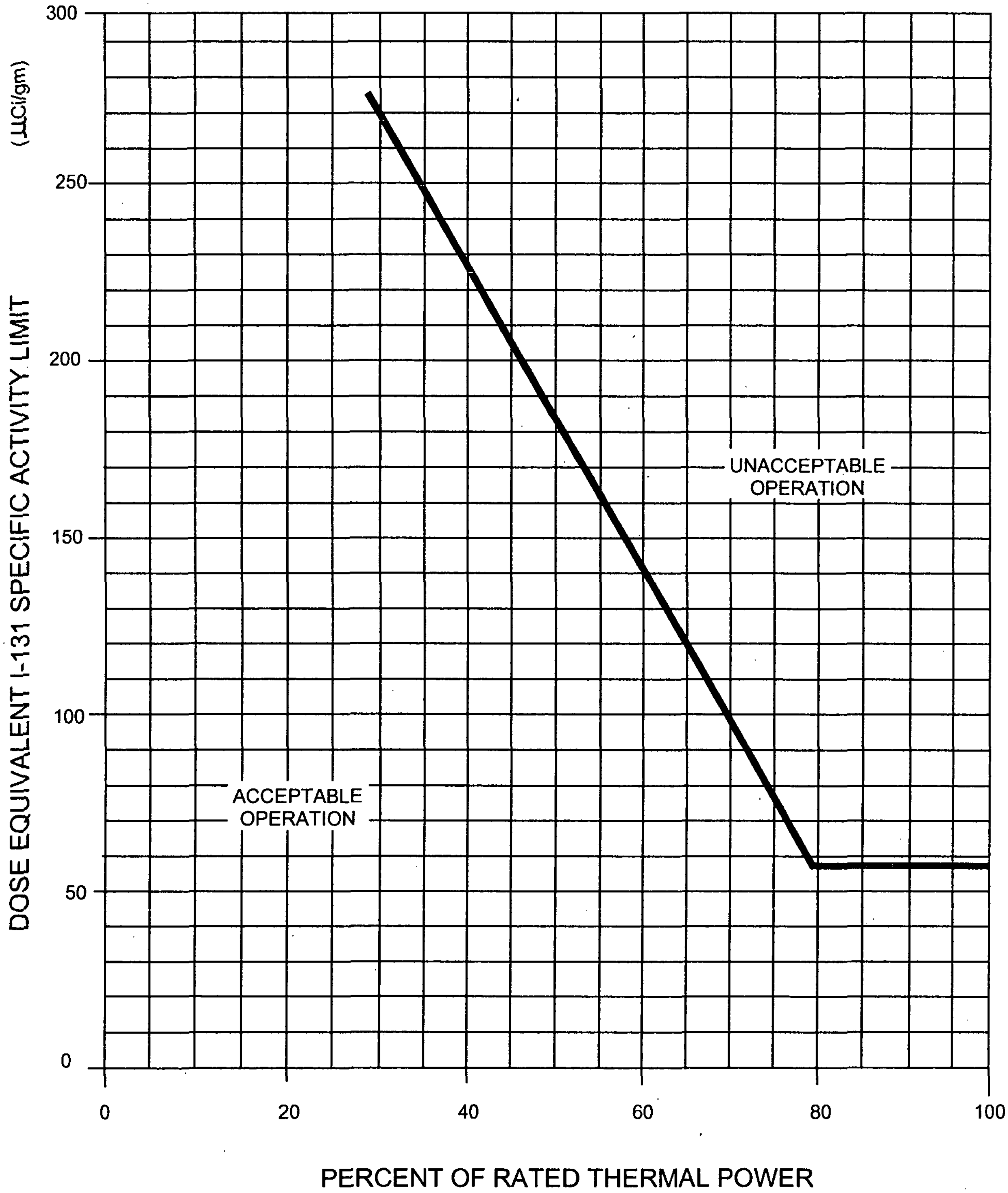

FIGURE 3.4-1

DOSE EQUIVALENT I-131 REACTOR COOLANT SPECIFIC ACTIVITY LIMIT VERSUS PERCENT OF RATED THERMAL POWER WITH THE REACTOR COOLANT SPECIFIC ACTIVITY >1 µCi/gram DOSE EQUIVALENT I-131

TABLE 4.4-3

REACTOR COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY	MODES IN WHICH SAMPLE AND ANALYSIS REQUIRED
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 Ē 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 1 μCi/gram DOSE EQUIVALENT 1-131 or 100/Ē Ci/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

---

* A radiochemical analysis for Ē shall consist of the quantitative measurement of the specific activity for each radionuclide, except for radionuclides with half-lives less than 10 minutes and all radioiodines, which is identified in the reactor coolant. The specific activities for these individual radionuclides shall be used in the determination of Ē for the reactor coolant sample. Determination of the contributors to Ē shall be based upon those energy peaks identifiable with a 95% confidence level.

**Sample to be taken after a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since reactor was last subcritical for 48 hours or longer.

#Until the specific activity of the Reactor Coolant System is restored within its limits.

REACTOR COOLANT SYSTEM

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

GENERAL

LIMITING CONDITION FOR OPERATION

3.4.9.1 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 of 100°F in any 1-hour period,

b. A maximum cooldown of 100°F in any 1-hour period, and

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: At all times.

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 be in at least HOT STANDBY within the next 6 hours and reduce the RCS T_{avg} and pressure to less than 200°F and 500 psig, respectively, within the following 30 hours.

SURVEILLANCE REQUIRMENTS

4.4.9.1 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.

MATERIAL PROPERTY BASIS

Limiting material: LOWER SHELL PLATE R-1808-1
Limiting ART values at 20 EFPY:1/4T, 109°F
3/4T, 88°F

Curves applicable for the first 20 EFPY and contain margins of 20°F and 100 psig for possible instrument errors

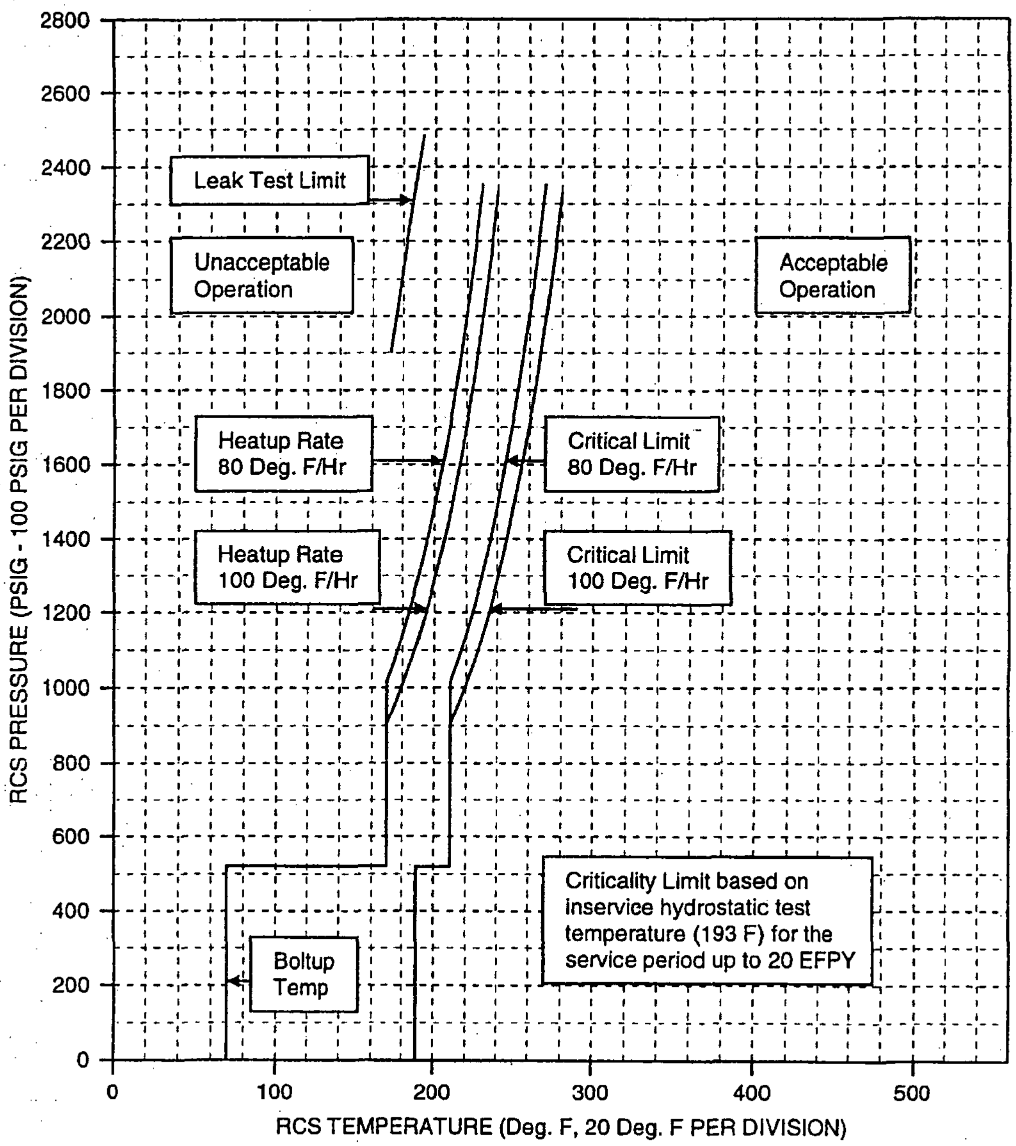

FIGURE 3.4-2
REACTOR COOLANT SYSTEM HEATUP LIMITATIONS - APPLICABLE UP TO 20 EFPY

MATERIAL PROPERTY BASIS

Limiting material: LOWER SHELL PLATE R-1808-1
Limiting ART values at 20 EFPY: 1/4T, 109°F
3/4T, 88°F

Curves applicable for the first 20 EFPY and contain margins of 20°F and 100 psig for possible instrument errors

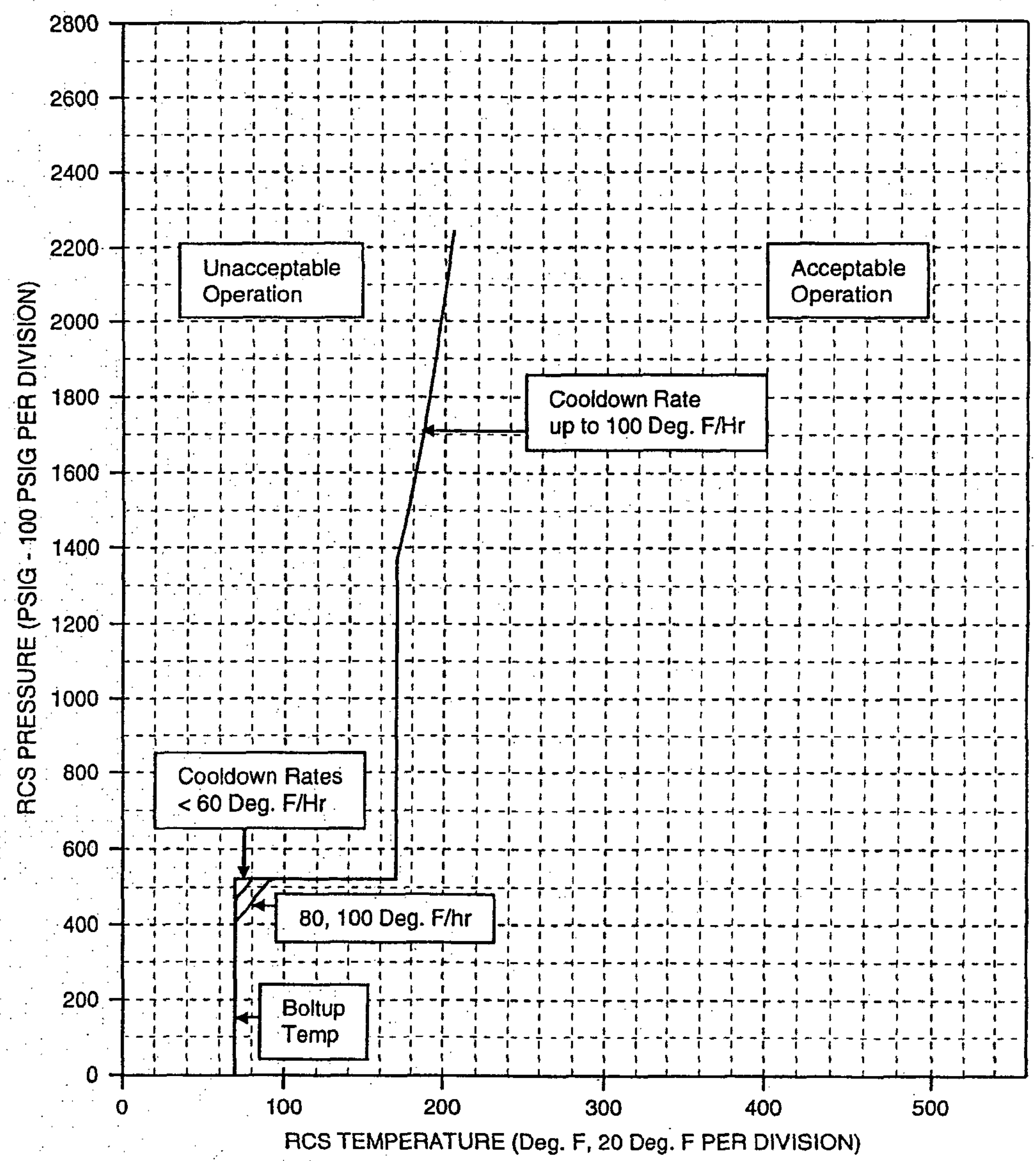

FIGURE 3.4-3
REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS - APPLICABLE UP TO 20 EFPY

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.9.2 The pressurizer temperature shall be limited to:

a. A maximum heatup of 100°F in any 1-hour period,

b. A maximum cooldown of 200°F in any 1-hour period, and

c. A maximum spray water temperature differential of 320°F.

APPLICABILITY: At all times.

ACTION:

With the pressurizer temperature limits in excess of any of the above limits, restore the temperature to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the pressurizer; determine that the pressurizer remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the pressurizer pressure to less than 500 psig within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.2 The pressurizer temperatures shall be determined to be within the limits at least once per 30 minutes during system heatup or cooldown. The spray water temperature differential shall be determined to be within the limit at least once per 12 hours during auxiliary spray operation.

3.4.9.3 The following Overpressure Protection Systems shall be OPERABLE:

a. In MODE 4 when the temperature of any RCS cold leg is less than or equal to 290°F; and in MODE 5 and MODE 6 with all Safety Injection pumps inoperable at least one of the following groups of two overpressure protection devices shall be OPERABLE when the RCS is not depressurized with an RCS vent area of greater than or equal to 1.58 square inches:

 1) Two residual heat removal (RHR) suction relief valves each with a setpoint of 450 psig +0, -3 %; or

 2) Two power-operated relief valves (PORVs) with lift setpoints that vary with RCS temperature which do not exceed the limit established in Figure 3.4-4, or

 3) One RHR suction relief valve and one PORV with setpoints as required above.

b. In MODE 5 and MODE 6 with all Safety Injection pumps except one inoperable:

 1) The Reactor Coolant System (RCS) depressurized with an RCS vent area equal to or greater than 18 square inches, or

 2) The RCS in a reduced inventory condition*.

APPLICABILITY: MODE 4 when the temperature of any RCS cold leg is less than or equal to 290°F; MODE 5 and MODE 6 with the reactor vessel head on and the vessel head closure bolts not fully detensioned.

ACTION:

a) In MODE 4 with all Safety Injection pumps inoperable and with one of the two required overpressure protection devices inoperable, either restore two overpressure protection devices to OPERABLE status within 7 days or within the next 8 hours

 (a) depressurize the RCS and

 (b) vent the RCS through at least a 1.58-square-inch vent.

*A reduced inventory condition exists whenever reactor vessel (RV) water level is lower than 36 inches below the RV flange.

3.4.9.3

ACTION: (Continued)

b) In MODE 5 and MODE 6 with all Safety Injection pumps inoperable and with one of the two required overpressure protection devices inoperable, restore two overpressure protection devices to OPERABLE status within 24 hours or within the next 8 hours

   (a) depressurize the RCS and

   (b) vent the RCS through at least a 1.58-square-inch vent.

c) In MODE 4, MODE 5 and MODE 6 with all Safety Injection pumps inoperable and with both of the two required overpressure protection devices inoperable, within the next 8 hours

   (a) depressurize the RCS and

   (b) vent the RCS through at least a 1.58-square-inch vent.

d) In the event the PORVs, or the RHR suction relief valves, or the RCS vent(s) are used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.8.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs, or the RHR suction relief valves, or RCS vent(s) on the transient, and any corrective action necessary to prevent recurrence.

e) In MODE 5 and MODE 6 with all Safety Injection pumps except one inoperable and with the RCS vent area less than 18 square inches or RCS water level not in a reduced inventory condition, immediately restore all Safety Injection pumps to inoperable status.

OVERPRESSURE PROTECTION SYSTEMS

SURVEILLANCE REQUIREMENTS

4.4.9.3.1 Each PORV shall be demonstrated OPERABLE when the PORV(s) are being used for overpressure protection by:

a. Performance of an ANALOG CHANNEL OPERATIONAL TEST on the PORV actuation channel, but excluding valve operation, at least once per 31 days thereafter when the PORV is required OPERABLE; and

b. Performance of a CHANNEL CALIBRATION on the PORV actuation channel at least once per 18 months; and

c. Verifying the PORV isolation valve is open at least once per 72 hours.

4.4.9.3.2 Each RHR suction relief valve shall be demonstrated OPERABLE when the RHR suction relief valve(s) are being used for overpressure protection as follows:

a. For RHR suction relief valve RC-V89 by verifying at least once per 72 hours that RHR suction isolation valves RC-V87 and RC-V88 are open.

b. For RHR suction relief valve RC-V24 by verifying at least once per 72 hours that RHR suction isolation valves RC-V22 and RC-V23 are open.

c. Testing pursuant to Specification 4.0.5.

4.4.9.3.3 The RCS vent(s) shall be verified to be open at least once per 12 hours** when the vent(s) is being used for overpressure protection.

4.4.9.3.4 The reactor vessel water level shall be verified to be lower than 36 inches below the reactor vessel flange at least once per 12 hours when the reduced inventory condition is being used for overpressure protection.

**Except when the vent pathway is provided with a valve(s) or device(s) that is locked, sealed, or otherwise secured in the open position, then verify this valve(s) or device(s) open at least once per 31 days.

VALID FOR THE FIRST 20 EFPY, SETPOINT CONTAINS MARGIN OF 50°F FOR TRANSIENT EFFECTS

T ≤ 200.0°F , P = 561.0 PSIG;
200.0°F < T ≤ 230.5°F , P = 12.1*(T-200.0) + 926.0 PSIG;
230.5°F < T ≤ 255.0°F , P = 23.15*(T-230.5) + 1295.05 PSIG;
T > 255.0°F , P = 34.5*(T-255.0) + 1862.225 PSIG

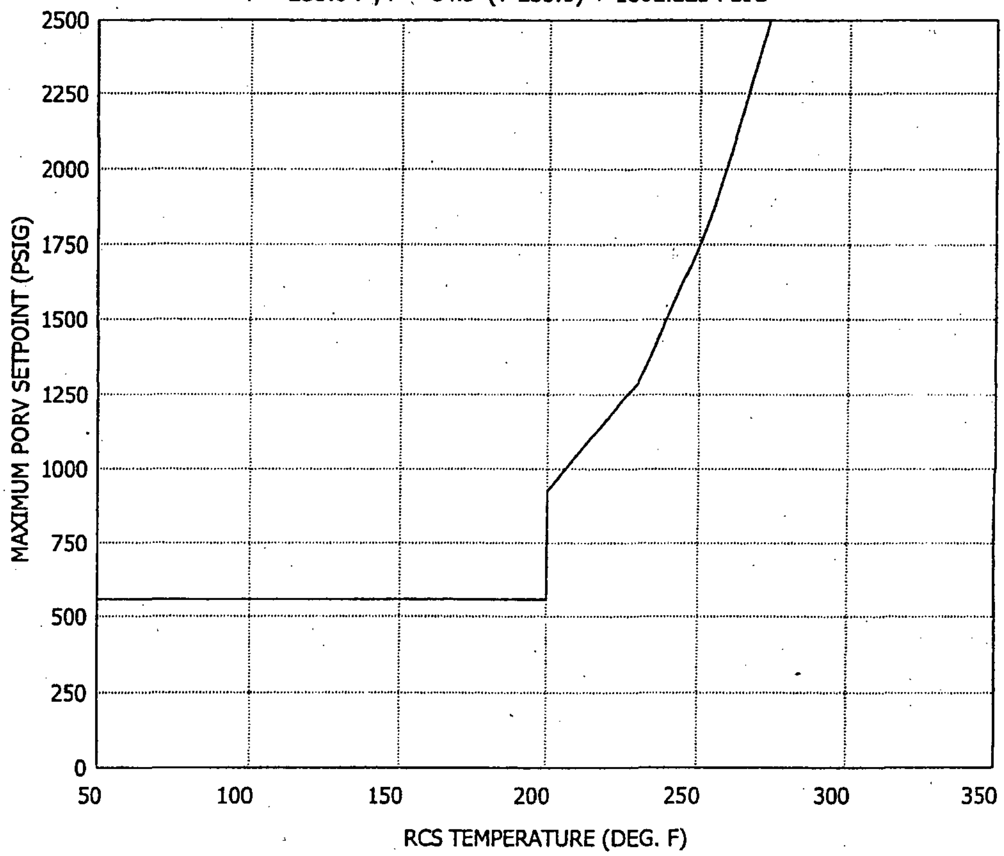

FIGURE 3.4-4 RCS COLD OVERPRESSURE PROTECTION SETPOINTS

LIMITING CONDITION FOR OPERATION

3.4.10 The structural integrity of ASME Code Class 1, 2, and 3 components shall be maintained in accordance with Specification 4.4.10.

APPLICABILITY: All MODES.

ACTION:

a. With the structural integrity of any ASME Code Class 1 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature more than 50°F above the minimum temperature required by NDT considerations.

b. With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 200°F.

c. With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) from service.

SURVEILLANCE REQUIREMENTS

4.4.10 In addition to the requirements of Specification 4.0.5, each reactor coolant pump flywheel shall be inspected at least once every 10 years. This inspection shall be by either of the following examinations:

a. An in-place examination, utilizing ultrasonic testing, over the volume from the inner bore of the flywheel to the circle of one-half the outer radius; or

b. A surface examination, utilizing magnetic particle testing and/or penetrant testing, of the exposed surfaces of the disassembled flywheel.

REACTOR COOLANT SYSTEM

3/4.4.11 REACTOR COOLANT SYSTEM VENTS

LIMITING CONDITION FOR OPERATION

3.4.11 At least one Reactor Coolant System vent path consisting of one vent valve and one block valve powered from emergency busses shall be OPERABLE and closed*at each of the following locations:

a. Reactor vessel head, and

b. Pressurizer steam space.

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

ACTION:

a. With one of the above Reactor Coolant System vent paths inoperable, STARTUP and/or POWER OPERATION may continue provided the inoperable vent path is maintained closed with power removed from the valve actuator of all the vent valves and block valves in the inoperable vent path; restore the inoperable vent path to OPERABLE status within 30 days, or, be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With both Reactor Coolant System vent paths inoperable; maintain the inoperable vent path closed with power removed from the valve actuators of all the vent valves and block valves in the inoperable vent paths, and restore at least one of the vent paths to OPERABLE status within 72 hours or be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.11.1 Each Reactor Coolant System vent path block valve not required to be closed by ACTION a. or b., above, shall be demonstrated OPERABLE at least once per COLD SHUTDOWN, if not performed within the previous 92 days, by operating the valve through one complete cycle of full travel from the control room.

4.4.11.2 Each Reactor Coolant System vent path shall be demonstrated OPERABLE at least once per 18 months by:

a. Verifying all manual isolation valves in each vent path are locked in the open position,

*For an OPERABLE vent path using a power-operated relief valve (PORV) as the vent path, the PORV block valve is not required to be closed.

SURVEILLANCE REQUIREMENTS

4.4.11.2 (Continued)

b. Cycling each vent valve through at least one complete cycle of full travel from the control room, and

c. Verifying flow through the Reactor Coolant System vent paths during venting.

PROCEDURES AND PROGRAMS

6.7.6 (Continued)

j. Technical Specification (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

1. A change in the TS incorporated in the license or

2. A change to the updated FSAR (UFSAR) or Bases that requires NRC approval pursuant to 10 CFR 50.59.

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the UFSAR.

d. Proposed changes that meet the criteria of Specification 6.7.6j.b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

k. Steam Generator (SG) Program

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following provisions:

a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged to confirm that the performance criteria are being met.

PROCEDURES AND PROGRAMS

6.7.6 (Continued)

b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.

   1. Structural integrity performance criterion: All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

   2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 1 gpm total or 500 gpd through any one SG.

   3. The operational LEAKAGE performance criterion is specified in LCO 3.4.6.2, "Reactor Coolant System Operational Leakage."

c. Provisions for SG tube repair criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged.

PROCEDURES AND PROGRAMS

6.7.6 (Continued)

The following alternate tube repair criteria may be applied as an alternative to the 40% depth based criteria:

1. During refueling outage 11 and the subsequent operating cycles until the next scheduled inspection, flaws identified in the portion of the tube below 17 inches from the top of the hot leg tubesheet do not require plugging.

   During refueling outage 11 and the subsequent operating cycles until the next scheduled inspection, all tubes with flaws identified in the portion of the tube within the region from the top of the hot leg tubesheet to 17 inches below the tubesheet shall be plugged.

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube repair criteria. During refueling outage 11 and the subsequent operating cycles until the next scheduled inspection, the portion of the tube below 17 inches from the top of the hot leg tubesheet is excluded from inspection when the alternate tube repair criteria in TS 6.7.6.k.c are implemented. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

   1. Inspect 100% of the tubes in each SG during the first refueling outage following SG replacement.

   2. Inspect 100% of the tubes at sequential periods of 120, 90, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 48 effective full power months or two refueling outages (whichever is less) without being inspected.

PROCEDURES AND PROGRAMS

6.7.6 (Continued)

3. If crack indications are found in any SG tube, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever is less). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

e. Provisions for monitoring operational primary to secondary leakage.

6.8 REPORTING REQUIREMENTS

ROUTINE REPORTS

6.8.1 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.

STARTUP REPORT

6.8.1.1 A summary report of station 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 station.

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

6.8.1.2 Annual Reports covering the activities of the station 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:

The results of specific activity analyses in which the primary coolant exceeded the limits of Specification 3.4.8. The following information shall be included: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded (in graphic and tabular format); (2) Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) Clean-up flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) Graph of the I-131 concentration (μCi/gm) and one other radio-iodine isotope concentration (μCi/gm) as a function of time for the duration of the specific activity above the steady-state level; and (5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT

6.8.1.3 The annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted by May 1 of each year. The report shall include summaries, interpretations, and analysis of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in (1) the ODCM and (2) Sections IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR Part 50.

ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT

6.8.1.4 The Annual Radioactive Effluent Release Report covering the operation of the unit during the previous calendar year of operation shall be submitted by May 1 of each year. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be (1) consistent with the objectives outlined in the ODCM and PCP and (2) in conformance with 10 CFR 50.36a and Section IV.B.1 of Appendix I to 10 CFR Part 50.

6.8.1.5 Deleted

CORE OPERATING LIMITS REPORT

6.8.1.6.a Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT prior to each reload cycle, or prior to any remaining portion of a reload cycle, for the following:

1. Cycle dependent Overpower ,T and Overtemperature ,T trip setpoint parameters and function modifiers for operation with skewed axial power profiles for Table 2.2-1 of Specification 2.2.1.

2. Cycle dependent maximum allowable combination of thermal power, pressurizer pressure and the highest operating loop average temperature (T_{avg}) for Specifications 2.1.1 and 2.1.2.

3. SHUTDOWN MARGIN and minimum boron concentration limits for MODES 1, 2, 3, and 4 for Specification 3.1.1.1.

4. SHUTDOWN MARGIN and minimum boron concentration limits for MODE 5 for Specification 3.1.1.2.

5. Moderator Temperature Coefficient BOL and EOL limits, and 300 ppm surveillance limit for Specification 3.1.1.3.

6. The minimum boron concentration for Modes 4, 5, and 6 for Specification 3.1.2.7.

7. Shutdown Rod Insertion limit for Specification 3.1.3.5.

8. Control Rod Bank Insertion limits for Specification 3.1.3.6.

9. AXIAL FLUX DIFFERENCE limits for Specification 3.2.1

10. Heat Flux Hot Channel Factor, F_Q^{RTP} and K(Z) for Specification 3.2.2.

11. Nuclear Enthalpy Rise Hot Channel Factor, and $F_{\Delta H}^{RTP}$ for Specification 3.2.3.

12. Cycle dependent DNB-related parameters for reactor coolant system average temperature (T_{avg}), and pressurizer pressure for Specification 3.2.5.

13. The boron concentration limits for MODES 1, 2 and 3 for Specification 3.5.1.1.

14. The boron concentration limits for MODES 1, 2, 3 and 4 for Specification 3.5.4.

15. The boron concentration limits for MODE 6 for Specification 3.9.1.

6.8.1.6.b The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC in:

1. WCAP-12945P-A, "Code Qualification Document for Best Estimate LOCA Analysis," Volume 1, Revision 2, and Volumes 2 through 5, Revision 1; Bajorek, S. M., et al, 1998.

   Methodology for Specification:
   3.2.2 - Heat Flux Hot Channel Factor

2. WCAP-10079-P-A, (Proprietary) and WCAP-10080-A (Nonproprietary), "NOTRUMP: A Nodal Transient Small Break and General Network Code", August 1985.

   Methodology for Specification:
   3.2.2 - Heat Flux Hot Channel Factor

3. YAEC-1363-A, "CASMO-3G Validation," April, 1988.

   YAEC-1659-A, "SIMULATE-3 Validation and Verification," September, 1988.

   WCAP-11596-P-A, (Proprietary), "Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores", June, 1988.

   WCAP-10965-P-A, (Proprietary), "ANC: A Westinghouse Advanced Nodal Computer Code", September, 1986.

   Methodology for Specifications:
   3.1.1.1 - SHUTDOWN MARGIN for MODES 1,2, 3, and 4
   3.1.1.2 - SHUTDOWN MARGIN for MODE 5
   3.1.1.3 - Moderator Temperature Coefficient
   3.1.3.5 - Shutdown Rod Insertion Limit
   3.1.3.6 - Control Rod 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

4. Seabrook Station Updated Final Safety Analysis Report, Section 15.4.6, "Chemical and Volume Control System Malfunction That Results in a Decrease in the Boron Concentration in the Reactor Coolant System".

   Methodology for Specifications:
   3.1.1.1 - SHUTDOWN MARGIN for MODES 1, 2, 3, and 4
   3.1.1.2 - SHUTDOWN MARGIN for MODE 5

6.8.1.6.b (Continued)

5. YAEC-1241, "Thermal-Hydraulic Analysis of PWR Fuel Elements Using the CHIC-KIN Code", R. E. Helfrich, March, 1981.

   WCAP-14565-P-A, (Proprietary), "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis", October, 1999.

   WCAP-15025-P-A, "Modified WRB-2 Correlation, WRB-2M, for Predicting Critical Heat Flux in 17x17 Rod Bundles with Modified LPD Mixing Vane Grids," April 1999.

   Methodology for Specification:
   2.1 - Safety Limits
   3.2.1 - AXIAL FLUX DIFFERENCE
   3.2.2 - Heat Flux Hot Channel Factor
   3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor
   3.2.5 - DNB Parameters

6. YAEC-1849P, "Thermal-Hydraulic Analysis Methodology Using VIPRE-01 For PWR Applications," October, 1992.

   WCAP-11397-P-A, (Proprietary), "Revised Thermal Design Procedure", April, 1989.

   WCAP-8745-P-A, Design Basis for the Thermal Overpower ,T and Thermal Overtemperature ,T Trip Functions," September 1986.

   Methodology for Specification:
   2.2.1 - Limiting Safety System Settings
   3.2.1 - AXIAL FLUX DIFFERENCE
   3.2.2 - Heat Flux Hot Channel Factor
   3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor

7. YAEC-1854P, "Core Thermal Limit Protection Function Setpoint Methodology For Seabrook Station," October, 1992

   Methodology for Specification:
   2.2.1 - Limiting Safety System Settings
   3.1.3.5 - Shutdown Rod Insertion Limit
   3.1.3.6 - Control Rod 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

6.8.1.6.b (Continued)

8. YAEC-1856P, "System Transient Analysis Methodology Using RETRAN for PWR Applications," December, 1992.

   Methodology for Specification:
   2.2.1 - Limiting Safety System Settings
   3.1.1.3 - Moderator Temperature Coefficient
   3.1.3.5 - Shutdown Rod Insertion Limit
   3.1.3.6 - Control Rod 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

9. YAEC-1752, "STAR Methodology Application for PWRs, Control Rod Ejection, Main Steam Line Break," October, 1990.

   Methodology for Specification:
   3.1.1.3 - Moderator Temperature Coefficient
   3.1.3.5 - Shutdown Rod Insertion Limit
   3.1.3.6 - Control Rod 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

10. YAEC-1855PA, "Seabrook Station Unit 1 Fixed Incore Detector System Analysis," October, 1992.

   Methodology for Specification:
   3.2.1 - AXIAL FLUX DIFFERENCE
   3.2.2 - Heat Flux Hot Channel Factor
   3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor

11. YAEC-1624P, "Maine Yankee RPS Setpoint Methodology Using Statistical Combination of Uncertainties - Volume 1 - Prevention of Fuel Centerline Melt," March, 1988.

   Methodology for Specification:
   3.2.1 - AXIAL FLUX DIFFERENCE
   3.2.2 - Heat Flux Hot Channel Factor
   3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor

6.8.1.6.b (Continued)

12. NYN-95048, Letter from T. C. Feigenbaum (NAESCo) to NRC, "License Amendment Request 95-05: Positive Moderator Temperature Coefficient", May 30, 1995.

    Methodology for Specification:
    3.1.1.3 - Moderator Temperature Coefficient

13. WCAP-12610-P-A, "VANTAGE + Fuel Assembly Reference Core Report". April, 1995, (Westinghouse Proprietary).

    Methodology for Specification:
    3.2.2 - Heat Flux Hot Channel Factor

14. WCAP-10216-P-A, Revision 1A (Proprietary), "Relaxation of Constant Axial Offset Control F_Q Surveillance Technical Specification", February, 1994.

    Methodology for Specification:
    3.2.1 - AXIAL FLUX DIFFERENCE
    3.2.2 - Heat Flux Hot Channel Factor

15. WCAP-9272-P-A, (Proprietary), "Westinghouse Reload Safety Evaluation Methodology", July, 1985.

    Methodology for Specifications:
    2.1 - Safety Limits
    3.1.1.1 - SHUTDOWN MARGIN for MODES 1,2,3, and 4
    3.1.1.2 - SHUTDOWN MARGIN for MODE 5
    3.1.1.3 - Moderator Temperature Coefficient
    3.1.2.7 - Isolation of Unborated Water Sources - Shutdown
    3.1.3.5 - Shutdown Rod Insertion Limit
    3.1.3.6 - Control Rod 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
    3.2.5 - DNB Parameters
    3.5.1.1 - Accumulators for MODES 1, 2, and 3
    3.5.4 - Refueling Water Storage Tank for MODES 1, 2, 3, and 4
    3.9.1 - Boron Concentration

16. WCAP-13749-P-A, (Proprietary) "Safety Evaluation Supporting the Conditional Exemption of the Most Negative Moderator Temperature Coefficient Measurement," March, 1997.

    Methodology for Specifications:
    3.1.1.3 - Moderator Temperature Coefficient

6.8.1.6.c The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as SHUTDOWN MARGIN, and transient and accident analysis limits) of the safety analysis are met. The CORE OPERATING LIMITS REPORT for each reload cycle, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, to the NRC Document Control Desk with copies to the Regional Administrator and the Resident Inspector.

STEAM GENERATOR TUBE INSPECTION REPORT

6.8.1.7 A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with Specification 6.7.6.k, Steam Generator (SG) Program. The report shall include:

a. The scope of inspections performed on each SG,

b. Active degradation mechanisms found,

c. Nondestructive examination techniques utilized for each degradation mechanism,

d. Location, orientation (if linear), and measured sizes (if available) of service induced indications,

e. Number of tubes plugged during the inspection outage for each active degradation mechanism,

f. Total number and percentage of tubes plugged to date,

g. The results of condition monitoring, including the results of tube pulls and in-situ testing,

h. The effective plugging percentage for all plugging in each SG.

SPECIAL REPORTS

6.8.2 Special reports shall be submitted to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attn: Document Control Desk, with a copy to the NRC Regional Administrator within the time period specified for each report.

6.9 (THIS SPECIFICATION NUMBER IS NOT USED)

6.10 RADIATION PROTECTION PROGRAM

6.10.1 Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained, and adhered to for all operations involving personnel radiation exposure.

6.11 HIGH RADIATION AREA

6.11.1 Pursuant to paragraph 20.1601(c) of 10 CFR Part 20, in lieu of the "control device" or "alarm signal" required by paragraph 20.1601(a) and (b), each high radiation area, as defined in 10 CFR Part 20, in which the intensity of radiation is equal to or less-than 1000 mR/h at 30 cm (12 in.) from the radiation source or from any surface that the radiation penetrates shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP). Individuals qualified in radiation protection procedures (e.g., Health Physics Technician) or personnel continuously escorted by such individuals may be exempt from the RWP issuance requirement during the performance of their assigned duties in high radiation areas with exposure rates equal to or less than 1000 mR/h, provided they are otherwise following plant radiation protection procedures for entry into such high radiation areas. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device that continuously indicates the radiation dose rate in the area; or

b. A radiation monitoring device that continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel have been made knowledgeable of them; or

c. An individual qualified in radiation protection procedures with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified in the Radiation Work Permit.

6.11.2 In addition to the requirements of Specification 6.11.1, areas accessible to personnel with radiation levels greater than 1000 mR/h at 30 cm (12 in.) from the radiation source or from any surface that the radiation penetrates shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the Shift Manager on duty and/or health physics supervision. Doors shall remain locked except during periods of access by personnel under an approved RWP that shall specify the dose rate levels in the immediate work areas and the maximum allowable stay time for individuals in that area. In lieu of the stay time specification of the RWP, direct or remote (such as closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities being performed within the area.

HIGH RADIATION AREA

6.11.2 (Continued)

For individual high radiation areas accessible to personnel with radiation levels of greater than 1000 mR/h that are located within large areas, such as PWR containment, where no enclosure exists for purposes of locking, and where no enclosure can be reasonably constructed around the individual area, that individual area shall be barricaded, conspicuously posted, and a flashing light shall be activated as a warning device.

6.12 PROCESS CONTROL PROGRAM (PCP)

Changes to the PCP:

a. Shall be documented and records of reviews performed shall be retained as required by the Operational Quality Assurance Program (OQAP). This documentation shall contain:

   1) Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and

   2) A determination that the change will maintain the overall conformance of the solidified waste product to existing requirements of Federal, State, or other applicable regulations.

b. Shall become effective after review and acceptance by the SORC and approval of the Station Director.

6.13 OFFSITE DOSE CALCULATION MANUAL (ODCM)

Changes to the ODCM:

a. Shall be documented and records of reviews performed shall be retained as required by the Operational Quality Control Program (OQAP). This documentation shall contain:

   1) Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and

   2) A determination that the change will maintain the level of radioactive effluent control required by 10 CFR 20.1302, 40 CFR Part 190, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50 and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations.

b. Shall become effective after review and acceptance by the SORC and the approval of the Station Director.

OFFSITE DOSE CALCULATION MANUAL (ODCM)

c. Shall be submitted to the Commission in the form of a complete, legible copy of the entire ODCM as part of or concurrent with the Annual Radioactive Effluent Release Report for the period of the report in which any change to the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and each affected page shall indicate the revision number the change was implemented.

6.14 MAJOR CHANGES TO LIQUID, GASEOUS, AND SOLID RADWASTE TREATMENT SYSTEMS*

6.14.1 Licensee-initiated major changes to the Radwaste Treatment Systems (liquid, gaseous, and solid):

a. Shall be reported to the Commission in the Annual Radioactive Effluent Release Report for the period in which the evaluation was reviewed by the SORC. The discussion of each change shall contain:

1) A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR 50.59;

2) Sufficient detailed information to totally support the reason for the change without benefit of additional or supplemental information;

3) A detailed description of the equipment, components, and processes involved and the interfaces with other plant systems;

4) An evaluation of the change, which shows the predicted releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the License application and amendments thereto;

5) An evaluation of the change, which shows the expected maximum exposures to a MEMBER OF THE PUBLIC in the UNRESTRICTED AREA and to the general population that differ from those previously estimated in the License application and amendments thereto;

*Licensees may choose to submit the information called for in this Specification as part of the FSAR update, pursuant to 10 CFR 50.71.

6.14.1 (Continued)

6) A comparison of the predicted releases of radioactive materials, in liquid and gaseous effluents and in solid waste, to the actual releases for the period prior to when the change is to be made;

7) An estimate of the exposure to plant operating personnel as a result of the change; and

8) Documentation of the fact that the change was reviewed and found acceptable by the SORC.

b. Shall become effective upon review and acceptance by the SORC.

6.15 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 accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak Test Program, dated September 1995," as modified by the following exception:

a. NEI 94-01-1995, Section 9.2-3: The first ILRT performed after October 30, 1992 shall be performed no later than April 29, 2008.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 49.6 psig.

The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.15% of primary containment air weight per day.

The provisions of SR 4.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

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

Containment 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 Type B and Type C tests and $\leq 0.75\ L_a$ for Type A tests.

CONTAINMENT LEAKAGE RATE TESTING PROGRAM

6.15 (Continued)

Overall air lock leakage rate acceptance criterion is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

Each containment 8-inch purge supply and exhaust isolation valve leakage rate acceptance criterion is $\leq 0.01\ L_a$ when tested at P_a.