ATTACHMENT TO LICENSE AMENDMENT NO. 213

TO RENEWED FACILITY OPERATING LICENSE NO. DPR-67

DOCKET NO. 50-335

Replace Pages 3 and 5 of Renewed Operating License DPR-67 with the attached Pages 3 and 5. Insert the attached Page 6 of Renewed Operating License DPR-67.

Replace the following pages of the Appendix "A" Technical Specifications with the attached pages. The revised pages are identified by amendment number and contain vertical lines indicating the area of change.

Remove Pages	Insert Pages
I	I
III	III
VIII	VIII
1-3	1-3
1-4	1-4
1-6	1-6
2-2	2-2
2-4	2-4
3/4 1-1	3/4 1-1
3/4 1-3	3/4 1-3
3/4 1-8	3/4 1-8
3/4 1-9a	3/4 1-9a
3/4 1-10	3/4 1-10
3/4 1-12	3/4 1-12
3/4 1-16	3/4 1-16
3/4 1-17	3/4 1-17
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3/4 4-21	3/4 4-21
3/4 4-22	3/4 4-22
3/4 4-23a	3/4 4-23a
3/4 4-23b	3/4 4-23b
3/4 4-23c	3/4 4-23c
3/4 4-59	3/4 4-59
3/4 4-60	3/4 4-60
3/4 5-1	3/4 5-1
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3/4 11-15	3/4 11-15
5-4	5-4
5-5	5-5
5-6	5-6
5-6b	5-6b
5-6c	5-6c
5-6d	5-6d
5-6e	5-6e
	5-6f
	5-6g
6-15b	6-15b
6-19	6-19
6-19a	6-19a
6-19b	6-19b

applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

A. Maximum Power Level

FPL is authorized to operate the facility at steady state reactor core power levels not in excess of 2700 megawatts (thermal).

B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 213 are hereby incorporated in the renewed license. FPL shall operate the facility in accordance with the Technical Specifications.

Appendix B, the Environmental Protection Plan (Non-Radiological), contains environmental conditions of the renewed license. If significant detrimental effects or evidence of irreversible damage are detected by the monitoring programs required by Appendix B of this license, FPL will provide the Commission with an analysis of the problem and plan of action to be taken subject to Commission approval to eliminate or significantly reduce the detrimental effects or damage.

C. Updated Final Safety Analysis Report

The Updated Final Safety Analysis Report supplement submitted pursuant to 10 CFR 54.21(d), as revised on March 28, 2003, describes certain future activities to be completed before the period of extended operation. FPL shall complete these activities no later than March 1, 2016, and shall notify the NRC in writing when implementation of these activities is complete and can be verified by NRC inspection.

The Updated Final Safety Analysis Report supplement as revised on March 28, 2003, described above, shall be included in the next scheduled update to the Updated Final Safety Analysis Report required by 10 CFR 50.71(e)(4), following issuance of this renewed license. Until that update is complete, FPL may make changes to the programs described in such supplement without prior Commission approval, provided that FPL evaluates each such change pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

D. Sustained Core Uncovery Actions

Procedural guidance shall be in place to instruct operators to implement actions that are designed to mitigate a small-break loss-of-coolant accident prior to a calculated time of sustained core uncovery.

(c) Actions to minimize release to include consideration of:
1. Water spray scrubbing
2. Dose to onsite responders

H. Control Room Habitability

Upon implementation of Amendment No. 205, adopting TSTF-448, Revision 3, the determination of control room envelope (CRE) unfiltered air inleakage as required by SR 4.7.7.1.e, in accordance with TS 6.8.4.m, the assessment of CRE habitability as required by Specification 6.8.4.m.c. (ii), and the measurement of CRE pressure as required by Specification 6.8.4.m.d, shall be considered met. Following implementation:

(a) The first performance of SR 4.7.7.1.e, in accordance with Specification 6.8.4.m.c(i), shall be within the specified Frequency of 6 years, plus the 18-month allowance of SR 4.0.2, as measured from September 2003, the date of the most recent successful tracer gas test, as stated in FPL letters to NRC dated December 9, 2003, and October 29, 2004, in response to Generic Letter 2003-01.

(b) The first performance of the periodic assessment of CRE habitability, Specification 6.8.4.m.c(ii), shall be within 3 years, plus the 9-month allowance of SR 4.0.2, as measured from September 2003, the date of the most recent successful tracer gas test, as stated in FPL letters to NRC dated December 9, 2003, and October 29, 2004, in response to Generic Letter 2003-01, or within the next 9 months if the time period since the most recent successful tracer gas test is greater than 3 years.

(c) The first performance of the periodic measurement of CRE pressure, Specification 6.8.4.c.d, shall be within 36 months in a staggered test basis, plus the 138 days allowed by SR 4.0.2, as measured from June 30, 2006, which is the date of the most recent successful pressure measurement test, or within 138 days if not performed previously.

I. RODEX2 Safety Analyses

RODEX2 has been specifically approved for use for St. Lucie Unit 1 licensing basis analyses. Upon NRC's approval of a generic supplement to the RODEX2 code and associated methods that accounts for thermal conductivity degradation (TCD), FPL will within six months:

(a) Demonstrate that St. Lucie Unit 1 safety analyses remain conservatively bounded in licensing basis analyses when compared to the NRC-approved generic supplement to the RODEX2 methodology, or

(b) Provide a schedule for the re-analysis using the NRC-approved generic supplement to the RODEX2 methodology for any of the affected licensing basis analyses.

4. This renewed license is effective as of the date of issuance and shall expire at midnight on March 1, 2036.

FOR THE NUCLEAR REGULATORY COMMISSION

ORIGINAL SIGNED BY
J. E. Dyer, Director
Office of Nuclear Reactor Regulation

Attachments:
1. Appendix A, Technical Specifications
2. Appendix B, Environmental Protection Plan

INDEX

DEFINITIONS

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

DOSE EQUIVALENT I-131

1.10 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (μCi/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 Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion."

DOSE EQUIVALENT XE-133

1.11 DOSE EQUIVALENT XE-133 shall be that concentration of Xe-133 (μCi/gram) that alone would produce the same acute dose to the whole body as the combined activities of noble gas nuclides Kr-85m, Kr-85, Kr-87, Kr-88, Xe-131m, Xe-133m, Xe-133, Xe-135m, Xe-135, and Xe-138 actually present. If a specific noble gas nuclide is not detected, it should be assumed to be present at the minimum detectable activity. The determination of DOSE EQUIVALENT XE-133 shall be performed using effective dose conversion factors for air submersion listed in Table III.1 of EPA Federal Guidance Report No. 12, 1993, "External Exposure to Radionuclides in Air, Water, and Soil."

ENGINEERED SAFETY FEATURES RESPONSE TIME

1.12 The ENGINEERED SAFETY FEATURES REPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays where applicable. 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.

FREQUENCY NOTATION

1.13 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.1.

GASEOUS RADWASTE TREATMENT SYSTEM

1.14 A GASEOUS RADWASTE TREATMENT SYSTEM is any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system offgases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

IDENTIFED LEAKAGE

1.15 IDENTIFIED LEAKAGE shall be:

a. Leakage (except CONTROLLED LEAKAGE) into closed systems, such as pump seal or valve packing leaks that are captured, and conducted to a sump or collecting tank, or

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 system (Primary-to-secondary leakage).

1.16 Deleted

MEMBER(S) OF THE PUBLIC

1.17 MEMBER OF THE PUBLIC means an individual in a controlled or unrestricted area. However, an individual is not a member of the public during any period in which the individual receives an occupational dose.

OFFSITE DOSE CALCULATION MANUAL (ODCM)

1.18 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 6.8.4 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 6.9.1.7 and 6.9.1.8.

RATED THERMAL POWER

1.25 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3020 MWt.

REACTOR TRIP SYSTEM RESPONSE TIME

1.26 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 electrical power to the CEA drive mechanism is interrupted. 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.

REPORTABLE EVENT

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

SHIELD BUILDING INTEGRITY

1.28 SHIELD BUILDING INTEGRITY shall exist when:

a. Each door is closed except when the access opening is being used for normal transit entry and exit;

b. The shield building ventilation system is in compliance with Specification 3.6.6.1, and

c. The sealing mechanism associated with each penetration (e.g., welds, bellows or O-rings) is OPERABLE.

SHUTDOWN MARGIN

1.29 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 full-length control element assemblies (shutdown and regulating) are fully inserted except for the single assembly of highest reactivity worth which is assumed to be fully withdrawn.

SITE BOUNDARY

1.30 SITE BOUNDARY means that line beyond which the land or property is not owned, leased, or otherwise controlled by the licensee.

SOURCE CHECK

1.31 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a radioactive source.

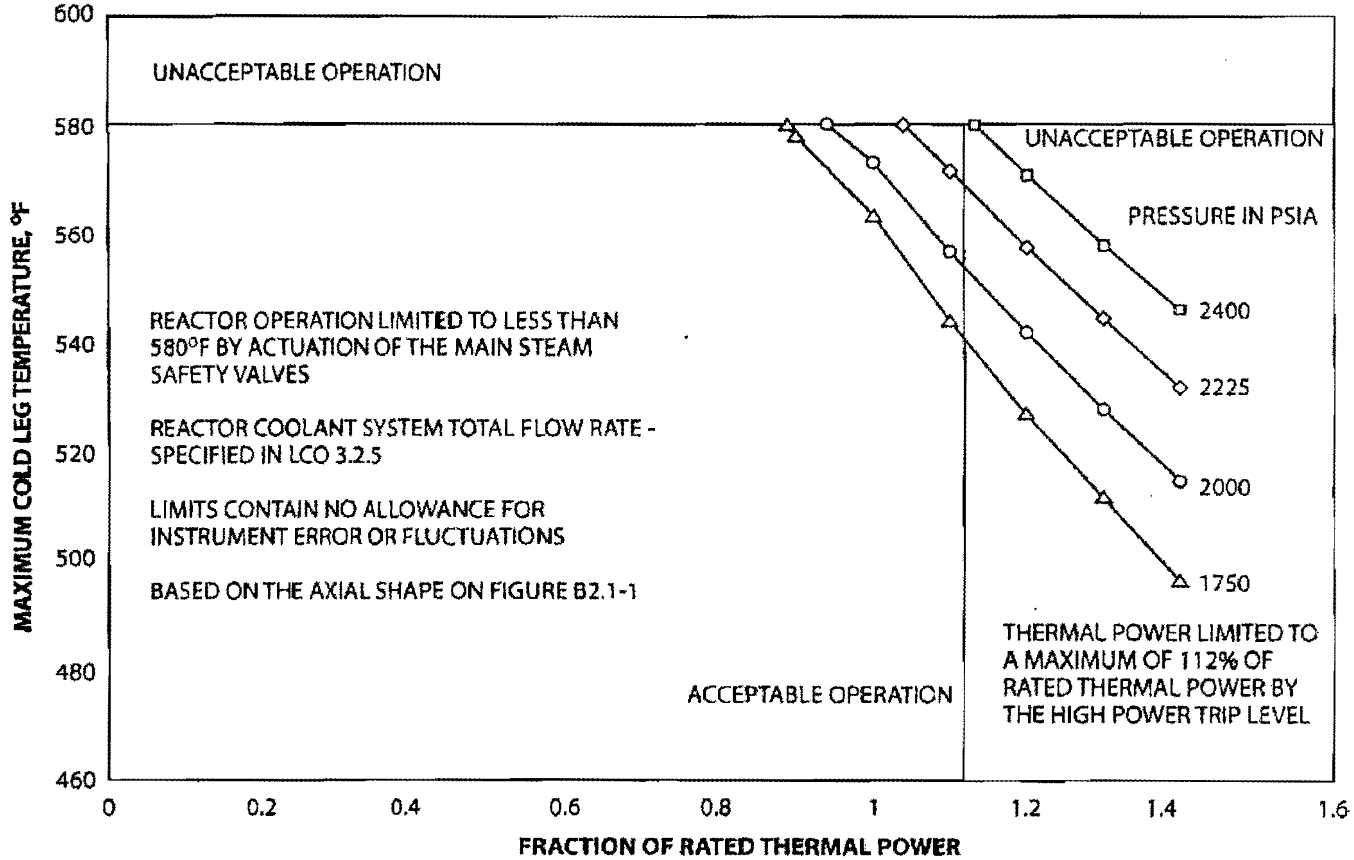

FIGURE 2.1-1: REACTOR CORE THERMAL MARGIN SAFETY LIMIT – FOUR REACTOR COOLANT PUMPS OPERATING

TABLE 2.2-1

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Power Level – High (1) Four Reactor Coolant Pumps Operating	≤ 9.61% above THERMAL POWER, with a minimum setpoint of 15% of RATED THERMAL POWER, and a maximum of < 107.0% of RATED THERMAL POWER.	≤ 9.61% above THERMAL POWER, and a minimum setpoint of 15% of RATED THERMAL POWER and a maximum of ≤ 107.0% of RATED THERMAL POWER.
3. Reactor Coolant Flow – Low (1) Four Reactor Coolant Pumps Operating	≥ 95% of minimum reactor coolant flow with 4 pumps operating *	≥ 95% of minimum reactor coolant flow with 4 pumps operating *
4. Pressurizer Pressure – High	≤ 2400 psia	≤ 2400 psia
5. Containment Pressure – High	≤ 3.3 psig	≤ 3.3 psig
6. Steam Generator Pressure – Low (2)	≥ 600 psia	≥ 600 psia
7. Steam Generator Water Level – Low	≥ 35.0% Water Level – each steam generator	≥ 35.0% Water Level – each steam generator
8. Local Power Density – High (3)	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-1 and 2.2-2.	Trip set point adjusted to not exceed the limit lines of Figures 2.2-1 and 2.2-2.

* For minimum reactor coolant flow with 4 pumps operating, refer to Technical Specification LCO 3.2.5.

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - T_{avg} > 200 °F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be within the limits specified in the COLR.

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

ACTION:

With the SHUTDOWN MARGIN not within limits immediately initiate and continue boration at ≥ 40 gpm of greater than or equal to 1900 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be within the COLR limits:

a. Within one hour after detection of an inoperable CEA(s) and at least once per 12 hours thereafter while the CEA(s) is inoperable. If the inoperable CEA is not fully inserted, and is immovable as a result of excessive friction or mechanical interference or is known to be untrippable, the above required SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable CEA(s).

b. When in MODES 1 or 2#, at least once per 12 hours by verifying that CEA group withdrawal is within the Power Dependent Insertion Limits of Specification 3.1.3.6.

c. When in MODE 2## at least once during CEA withdrawal and at least once per hour thereafter until the reactor is critical.

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e below, with the CEA groups at the Power Dependent Insertion Limits of Specification 3.1.3.6.

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* See Special Test Exception 3.10.1.

\# With $K_{eff} \geq 1.0$.

\## With $K_{eff} < 1.0$.

SHUTDOWN MARGIN - $T_{avg} \leq 200$ °F

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be:

Within the limits specified in the COLR, and in addition with the Reactor Coolant System drained below the hot leg centerline, one charging pump shall be rendered inoperable.*

APPLICABILITY: MODE 5.

ACTION:

If the SHUTDOWN MARGIN requirements cannot be met, immediately initiate and continue boration at $\geq$ 40 gpm of greater than or equal to 1900 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN requirements of Specification 3.1.1.2 shall be determined:

a. Within one hour after detection of an inoperable CEA(s) and at least once per 12 hours thereafter while the CEA(s) is inoperable. If the inoperable CEA is immovable or untrippable, the above required SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable CEA(s).

b. At least once per 24 hours by consideration of the following factors:

1. Reactor coolant system boron concentration,
2. CEA position,
3. Reactor coolant system average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

c. At least once per 24 hours, when the Reactor Coolant System is drained below the hot leg centerline, by consideration of the factors in 4.1.1.2.b and by verifying at least one charging pump is rendered inoperable.*

* Breaker racked-out.

3/4.1.2 BORATION SYSTEMS

FLOW PATHS – SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.1 As a minimum, one of the following boron injection flow paths shall be OPERABLE and capable of being powered from an OPERABLE emergency power source.

a. A flow path from the boric acid makeup tank via either a boric acid pump or a gravity feed connection and any charging pump to the Reactor Coolant System if only the boric acid makeup tank in Specification 3.1.2.7a is OPERABLE, or

b. The flow path from the refueling water tank via either a charging pump or a high pressure safety injection pump* to the Reactor Coolant System if only the refueling water tank in Specification 3.1.2.7b is OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

With none of the above flow paths OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes** until at least one injection path is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.2.1 At least one of the above required flow paths shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

* The flow path from the RWT to the RCS via a single HPSI pump shall only be established if: (a) the RCS pressure boundary does not exist, or (b) RCS pressure boundary integrity exists and no charging pumps are operable. In the latter case, all charging pumps shall be disabled.

** Plant temperature changes are allowed provided the temperature change is accounted for in the calculated SHUTDOWN MARGIN.

DELETED

REACTIVITY CONTROL SYSTEMS

FLOW PATHS – OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.2 At least two of the following three boron injection flow paths shall be OPERABLE:

a. One flow path from the boric acid makeup tank(s) with the tank meeting Specification 3.1.2.8 part a) or b), via a boric acid makeup pump through a charging pump to the Reactor Coolant System.

b. One flow path from the boric acid makeup tank(s) with the tank meeting Specification 3.1.2.8 part a) or b), via a gravity feed valve through a charging pump to the Reactor Coolant System.

c. The flow path from the refueling water storage tank via a charging pump to the Reactor Coolant System.

OR

At least two of the following three boron injection flow paths shall be OPERABLE:

d. One flow path from each boric acid makeup tank with the combined tank contents meeting Specification 3.1.2.8 c), via both boric acid makeup pumps through a charging pump to the Reactor Coolant System.

e. One flow path from each boric acid makeup tank with the combined tank contents meeting Specification 3.1.2.8 c), via both gravity feed valves through a charging pump to the Reactor Coolant System.

f. The flow path from the refueling water storage tank, via a charging pump to the Reactor Coolant System.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one of the above required boron injection flow paths to the Reactor Coolant System OPERABLE, restore at least two boron injection flow paths to the Reactor Coolant System to OPERABLE status within 72 hours or make the reactor subcritical within the next 2 hours and borate to a SHUTDOWN MARGIN equivalent to the requirements of Specification 3.1.1.2 at 200°F; restore at least two flow paths to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

CHARGING PUMPS – SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.3 At least one charging pump or high pressure safety injection pump* in the boron injection flow path required OPERABLE pursuant to Specification 3.1.2.1 shall be OPERABLE and capable of being powered from an OPERABLE emergency bus.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no charging pump or high pressure safety injection pump* OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes** until at least one of the required pumps is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.2.3 At least one of the above required pumps shall be demonstrated OPERABLE by verifying the charging pump develops a flow rate of greater than or equal to 40 gpm or the high pressure safety injection pump develops a total head of greater than or equal to 2571 ft. when tested pursuant to the Inservice Testing Program.

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* The flow path from the RWT to the RCS via a single HPSI pump shall be established only if: (a) the RCS pressure boundary does not exist, or (b) RCS pressure boundary integrity exists and no charging pumps are operable. In the latter case, all charging pumps shall be disabled.

** Plant temperature changes are allowed provided the temperature change is accounted for in the calculated SHUTDOWN MARGIN.

REACTIVITY CONTROL SYSTEMS

BORATED WATER SOURCES – SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

a. One boric acid makeup tank with a minimum borated water volume of 3650 gallons of 3.0 to 3.5 weight percent boric acid (5245 to 6119 ppm boron).

b. The refueling water tank with:

1. A minimum contained volume of 125,000 gallons,

2. A minimum boron concentration of 1900 ppm, and

3. A minimum solution temperature of 40°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water sources OPERABLE, suspend all operations involving positive reactivity changes* until at least one borated water source is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.2.7 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 water level of the tank, and.

b. At least once per 24 hours by verifying the RWT temperature when it is the source of borated water and the site ambient air temperature is < 40°F.

c. At least once per 24 hours when the Reactor Auxiliary Building air temperature is less than 55°F by verifying that the Boric Acid Makeup Tank solution temperature is greater than 55°F when that Boric Acid Makeup Tank is required to be OPERABLE.

* Plant temperature changes are allowed provided the temperature change is accounted for in the calculated SHUTDOWN MARGIN.

BORATED WATER SOURCES – OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.8 At least two of the following four borated water sources shall be OPERABLE:

a. Boric Acid Makeup Tank 1A in accordance with Figure 3.1-1.

b. Boric Acid Makeup Tank 1B in accordance with Figure 3.1-1.

c. Boric Acid Makeup Tanks 1A and 1B with a minimum combined contained borated water volume in accordance with Figure 3.1-1.

d. The refueling water tank with:

1. A minimum contained volume of 477,360 gallons of water,

2. A minimum boron concentration of 1900 ppm,

3. A maximum solution temperature of 100°F,

4. A minimum solution temperature of 55°F when in MODES 1 and 2, and

5. A minimum solution temperature of 40°F when in MODES 3 and 4.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one borated water source OPERABLE, restore at least two borated water sources to OPERABLE status within 72 hours or make the reactor subcritical within the next 2 hours and borate to a SHUTDOWN MARGIN equivalent to the requirements of Specification 3.1.1.2 at 200°F; restore at least two borated water sources to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.8 At least two borated water sources shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the boron concentration of the water source,

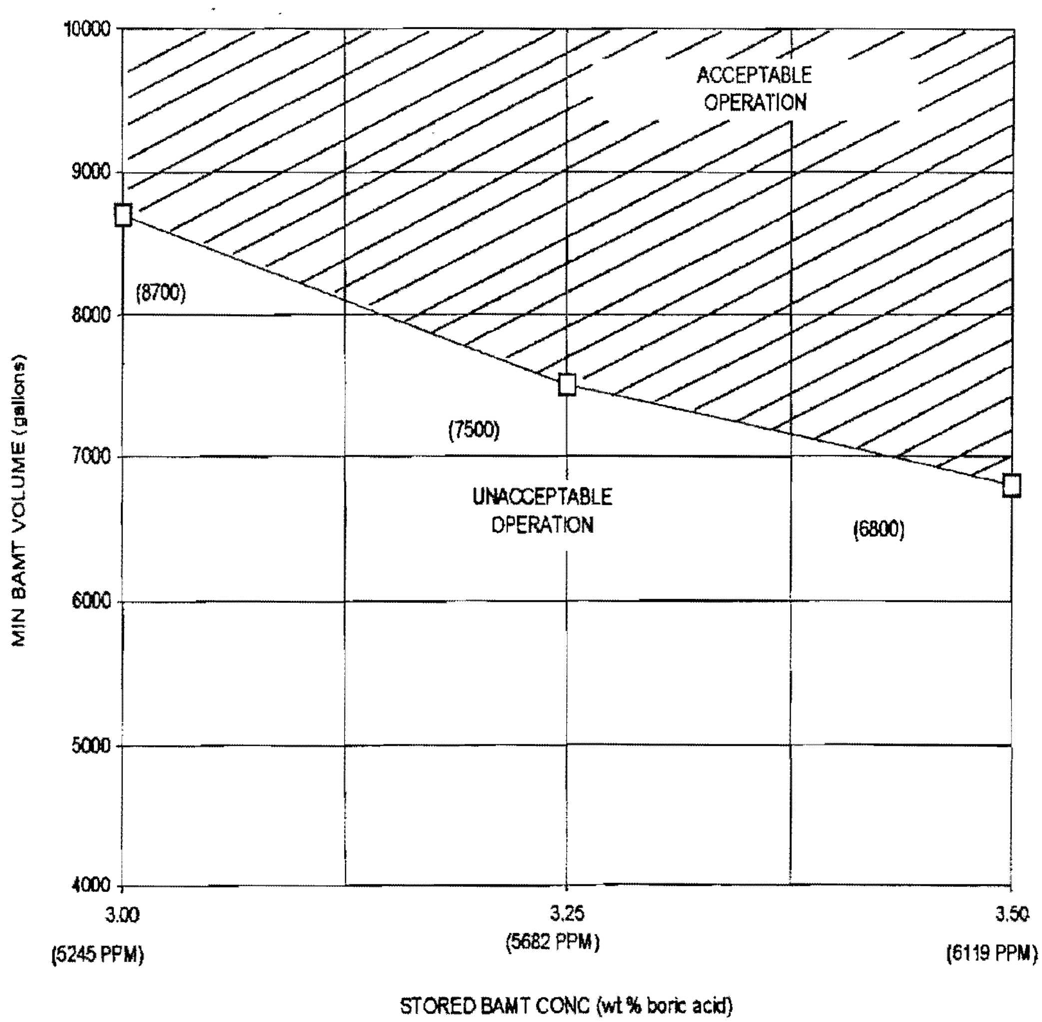
SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying the water level in each water source.

b. At least once per 24 hours by verifying the RWT temperature.

c. At least once per 24 hours by verifying that the Boric Acid Makeup Tank solution temperature is greater than 55°F when the Reactor Auxiliary Building air temperature is below 55°F.

FIGURE 3.1-1 ST. LUCIE 1 MIN BAMT VOLUME
VS STORED BAMT CONCENTRATION
(MODES 1, 2, 3 and 4)



POWER DISTRIBUTION LIMITS

DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

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

a. Cold Leg Temperature as shown on Table 3.2-1 of the COLR,

b. Pressurizer Pressure* as shown on Table 3.2-1 of the COLR,

c. Reactor Coolant System Total Flow Rate - greater than or equal to 375,000 gpm, and

d. AXIAL SHAPE INDEX as shown on Figure 3.2-4 of the COLR.

APPLICABILITY: MODE 1.

ACTION:

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to ≤ 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5.1 Each of the DNB related parameters shall be verified to be within their limits by instrument readout at least once per 12 hours.

4.2.5.2 The Reactor Coolant System total flow rate shall be determined to be within its limit by measurement** at least once per 18 months.

* Limit not applicable during either a THERMAL POWER ramp increase in excess of 5% per minute of RATED THERMAL POWER or a THERMAL POWER step increase of greater than 10% of RATED THERMAL POWER.

** Not required to be performed until THERMAL POWER is ≥ 90% of RATED THERMAL POWER.

Relocated to the COLR

TABLE 4.3-1

REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. Manual Reactor Trip	N/A	N.A.	S/U(1)	N/A
2. Power Level – High				
a. Nuclear Power	S	D(2), M(3), Q(5)	M	1,2
b. ΔT Power	S	D(4), Q	M	1
3. Reactor Coolant Flow – Low	S	R	M	1, 2
4. Pressurizer Pressure – High	S	R	M	1, 2
5. Containment Pressure – High	S	R	M	1, 2
6. Steam Generator Pressure – Low	S	R	M	1, 2
7. Steam Generator Water Level – Low	S	R	M(6, 7)	1, 2
8. Local Power Density – High	S	R	M	1
9. Thermal Margin/Low Pressure	S	R	M	1, 2
9a. Steam Generator Pressure Difference – High	S	R	M	1, 2
10. Loss of Turbine -- Hydraulic Fluid Pressure – Low	N.A.	N.A.	S/U(1)	N.A.
11. Wide Range Logarithmic Neutron Flux Monitor	S	N.A.	S/U(1)	1, 2, 3, 4, 5 and *
12. Reactor Protection System Logic	N.A.	N.A.	M and S/U(1)	1, 2 and *
13. Reactor Trip Breakers	N.A.	N.A.	M	1, 2 and *

TABLE 4.3-1 (Continued)

TABLE NOTATION

* - With reactor trip breaker closed.

(1) - If not performed in previous 7 days.

(2) - Heat balance only, above 15% of RATED THERMAL POWER; adjust "Nuclear Power Calibrate" potentiometer to null "Nuclear Pwr – ΔT Pwr." During PHYSICS TESTS, these daily calibrations of nuclear power and ΔT power may be suspended provided these calibrations are performed upon reaching each major test power plateau and prior to proceeding to the next major test power plateau.

(3) - Above 15% of RATED THERMAL POWER, recalibrate the excore detectors which monitor the AXIAL SHAPE INDEX by using the incore detectors or restrict THERMAL POWER during subsequent operations to $\leq$ 90% of the maximum allowed THERMAL POWER level with the existing Reactor Coolant Pump combination.

(4) - Adjust "ΔT Pwr Calibrate" potentiometers to make ΔT power signals agree with calorimetric calculation.

(5) - Neutron detectors may be excluded from CHANNEL CALIBRATION.

(6) - If the as-found setpoint is either outside its predefined as-found acceptance criteria band or is not conservative with respect to the Allowable Value, then the channel shall be declared inoperable and shall be evaluated to verify that it is functioning as required before returning the channel to service.

(7) - The instrument channel setpoint shall be reset to a value that is within the as-left tolerance of the Field Trip Setpoint, otherwise that channel shall not be returned to OPERABLE status. The Field Trip Setpoint and the methodology used to determine the Field Trip Setpoint, the as-found acceptance criteria band, and the as-left acceptance criteria are specified in the UFSAR Section 7.2.

REACTOR COOLANT SYSTEM

SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

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

a. $\leq$ 1.0 μCi/gram DOSE EQUIVALENT I-131, and

b. $\leq$ 518.9 μCi/gram DOSE EQUIVALENT XE-133.

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

ACTION:

a. With the specific activity of the primary coolant >1.0 μCi/gram DOSE EQUIVALENT I-131, verify DOSE EQUIVALENT I-131 is $\leq$ 60.0 μCi/gram once per four hours.

b. With the specific activity of the primary coolant > 1.0 μCi/gram DOSE EQUIVALENT I-131, but $\leq$ 60.0 μCi/gram DOSE EQUIVALENT I-131, operation may continue for up to 48 hours while efforts are made to restore DOSE EQUIVALENT I-131 to within the 1.0 μCi/gram limit. Specification 3.0.4 is not applicable.

c. With the specific activity of the primary coolant > 1.0 μCi/gram DOSE EQUIVALENT I-131 for greater than 48 hours during one continuous time interval, or > 60.0 μCi/gram DOSE EQUIVALENT I-131, be in HOT STANDBY within 6 hours and COLD SHUTDOWN within the following 30 hours.

d. With the specific activity of the primary coolant > 518.9 μCi/gram DOSE EQUIVALENT XE-133, operation may continue for up to 48 hours while efforts are made to restore DOSE EQUIVALENT XE-133 to within the 518.9 μCi/gram DOSE EQUIVALENT XE-133 limit. Specification 3.0.4 is not applicable.

e. With the specific activity of the primary coolant > 518.9 μCi/gram DOSE EQUIVALENT XE-133 for greater than 48 hours during one continuous time interval, be in HOT STANDBY within 6 hours and COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

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

TABLE 4.4-4

PRIMARY COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	MINIMUM FREQUENCY	MODES IN WHICH SAMPLE AND ANALYSIS REQUIRED
1. DOSE EQUIVALENT XE-133 Determination	1 per 7 days	1, 2, 3 and 4
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	1 per 14 days	1
3. Isotopic Analysis for Iodine Including I-131, I-132, I-133, I-134, and I-135	a) Once per 4 hours, whenever the DOSE EQUIVALENT I-131 exceeds 1.0 μCi/gram, and	1#, 2#, 3#, and 4#
	b) One sample between 2 and 6 hours following a THERMAL POWER change exceeding 15 percent of the RATED THERMAL POWER within a one hour period.	1, 2, 3

Until the specific activity of the primary coolant system is restored within its limits.

DELETED

REACTOR COOLANT SYSTEM

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

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 3.4-2a and 3.4-2b during heatup, cooldown, criticality, and inservice leak and hydrostatic testing.

APPLICABILITY: At all times.*

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limits within 30 minutes; perform an analysis to determine the effects of the out-of-limit condition on the fracture toughness properties of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operations or be in at least HOT STANDBY within the next 6 hours and reduce the RCS T_{avg} to less than 200°F within the following 30 hours in accordance with Figure 3.4-2b.

* During hydrostatic testing operations above system design pressure, a maximum temperature change in any one hour period shall be limited to 5°F.

SURVEILLANCE REQUIREMENTS

4.4.9.1

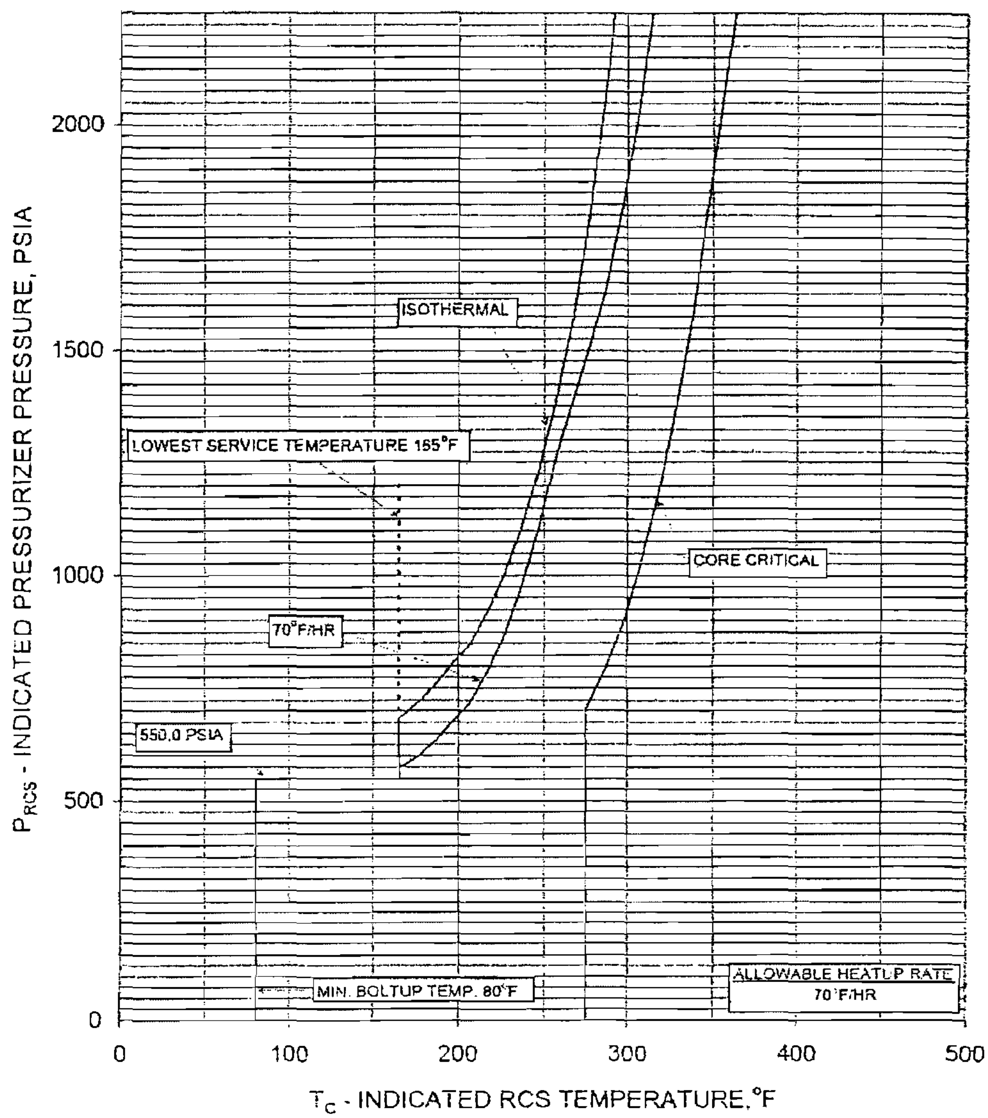
a. The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

b. The Reactor Coolant System temperature and pressure conditions shall be determined to be to the right of the criticality limit line within 15 minutes prior to achieving reactor criticality.

c. The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties as required by 10 CFR 50 Appendix H. The results of these examinations shall be used to update Figures 3.4-2a and 3.4-2b.

FIGURE 3.4-2a

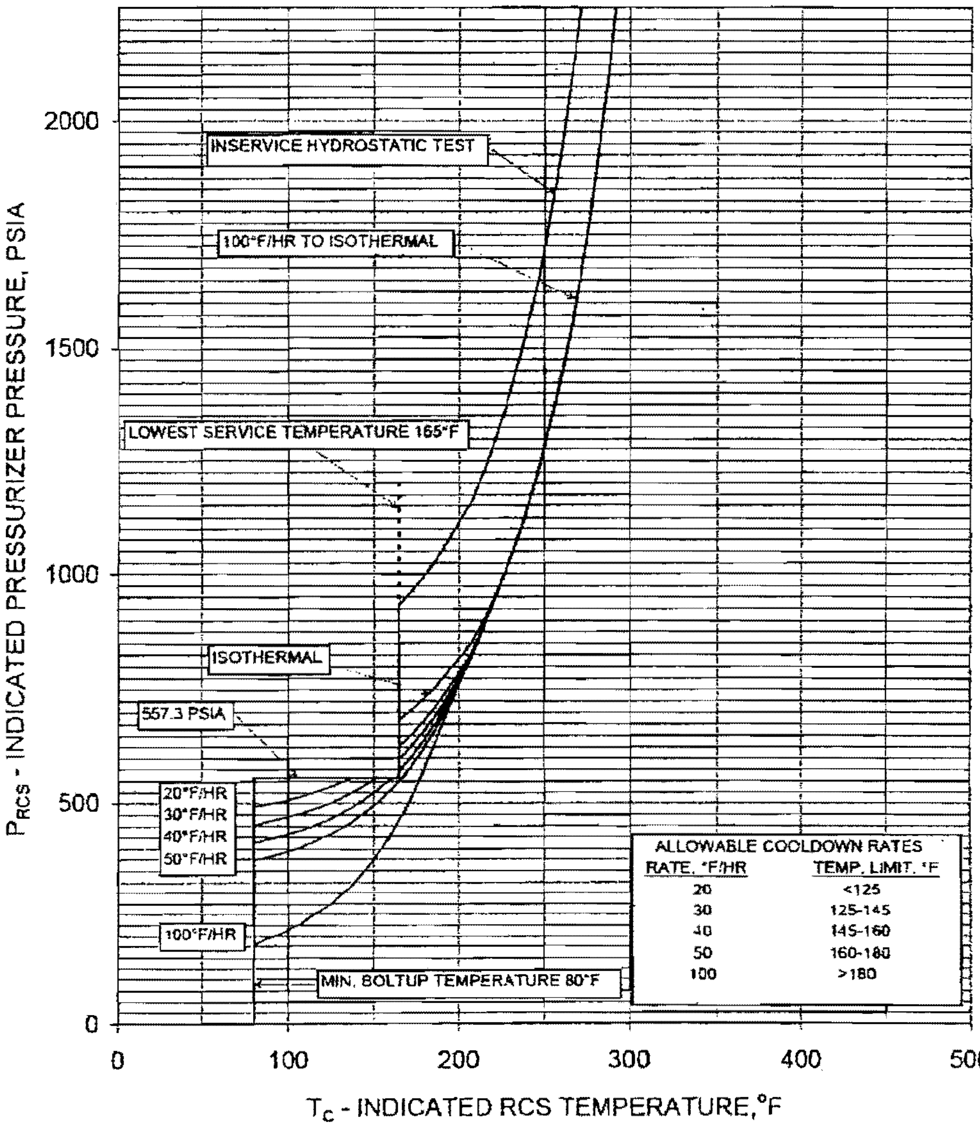
ST. LUCIE UNIT 1 P/T LIMITS, 54 EFPY
HEATUP AND CORE CRITICAL



Limiting Material: Lower Shell Axial Welds (Ht. #305424)
Limiting ART Values at 54 EFPY: 1/4T, 210°F
3/4T, 156°F

FIGURE 3.4-2b

ST. LUCIE UNIT 1 P/T LIMITS, 54 EFPY
COOLDOWN AND INSERVICE TEST



Limiting Material: Lower Shell Axial Welds (Ht. #305424)
Limiting ART Values at 54 EFPY: 1/4T, 210°F
3/4T, 156°F

DELETED

POWER OPERATED RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.13 Two power operated relief valves (PORVs) shall be OPERABLE, with their setpoints selected to the low temperature mode of operation as follows:

a. A setpoint of less than or equal to 350 psia shall be selected during heatup, cooldown and isothermal conditions when the temperature of any RCS cold leg is less than or equal to 200°F.

b. A setpoint of less than or equal to 530 psia shall be selected during heatup, cooldown and isothermal conditions when the temperature of any RCS cold leg is greater than 200°F and less than or equal to 300°F.

APPLICABILITY: MODE 4 when the temperature of any RCS cold leg is less than or equal to 300°F, MODE 5, and MODE 6 when the head is on the reactor vessel; and the RCS is not vented through greater than a 1.75 square inch vent.

ACTION:

a. With one PORV inoperable in MODE 4, restore the inoperable PORV to OPERABLE status within 7 days; or depressurize and vent the RCS through greater than a 1.75 square inch vent within the next 8 hours.

b. With one PORV inoperable in MODES 5 or 6, either (1) restore the inoperable PORV to OPERABLE status within 24 hours, or (2) complete depressurization and venting of the RCS through greater than a 1.75 square inch vent within a total of 32 hours.

c. With both PORVs inoperable, restore at least one PORV to operable status or complete depressurization and venting of the RCS through greater than a 1.75 square inch vent within 24 hours.

d. With the RCS vented per ACTIONS a, b, or c, verify the vent pathway at least once per 31 days when the pathway is provided by a valve(s) that is locked, sealed, or otherwise secured in the open position; otherwise, verify the vent pathway every 12 hours.

e. In the event either the PORVs 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 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs or RCS vent(s) on the transient, and any corrective action necessary to prevent recurrence.

f. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.13 Each PORV shall be demonstrated OPERABLE by:

a. Verifying the PORV isolation valve is open at least once per 72 hours; and

b. Performance of a CHANNEL FUNCTION TEST, but excluding valve operation, at least once per 31 days; and

c. Performance of a CHANNEL CALIBRATION at least once per 18 months.

REACTOR COOLANT SYSTEM

REACTOR COOLANT PUMP - STARTING

LIMITING CONDITION FOR OPERATION

3.4.14 If the steam generator temperature exceeds the primary temperature by more than 30°F, the first idle reactor coolant pump shall not be started.

APPLICABILITY: MODES 4# and 5.

ACTION:

If a reactor coolant pump is started when the steam generator temperature exceeds primary temperature by more than 30°F, evaluate the subsequent transient to determine compliance with Specification 3.4.9.1.

SURVEILLANCE REQUIREMENTS

4.4.14 Prior to starting a reactor coolant pump, verify that the steam generator temperature does not exceed primary temperature by more than 30°F.

Reactor Coolant System Cold Leg Temperature is less than 300°F.

3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

SAFETY INJECTION TANKS (SIT)

LIMITING CONDITION FOR OPERATION

3.5.1 Each reactor coolant system safety injection tank shall be OPERABLE with:

a. The isolation valve open,

b. Between 1090 and 1170 cubic feet of borated water,

c. A minimum boron concentration of 1900 ppm, and

d. A nitrogen cover-pressure of between 230 and 280 psig.

APPLICABILITY: MODES 1, 2 and 3.*

ACTION:

a. With one SIT inoperable due to boron concentration not within limits, or due to an inability to verify the required water volume or nitrogen cover-pressure, restore the inoperable SIT to OPERABLE status with 72 hours; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With one SIT inoperable due to reasons other than those stated in ACTION-a, restore the inoperable SIT to OPERABLE status within 24 hours; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.1 Each safety injection tank shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1. Verifying that the borated water volume and nitrogen cover-pressure in the tanks are within their limits, and

2. Verifying that each safety injection tank isolation valve is open.

---

* With pressurizer pressure $\geq$ 1750 psia.

ECCS SUBSYSTEMS - OPERATING

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent ECCS subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE high-pressure safety injection (HPSI) pump,

b. One OPERABLE low-pressure safety injection pump,

c. An independent OPERABLE flow path capable of taking suction from the refueling water tank on a Safety Injection Actuation Signal and automatically transferring suction to the containment sump on a Recirculation Actuation Signal, and

d. One OPERABLE charging pump*.

APPLICABILITY: MODES 1, 2 and 3**.

ACTION:

a. 1. With one ECCS subsystem inoperable only because its associated LPSI train is inoperable, restore the inoperable subsystem to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

2. With one ECCS subsystem inoperable for reasons other than condition a.1., restore the inoperable subsystem to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date.

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* One ECCS subsystem charging pump shall satisfy the flow path requirements of Specification 3.1.2.2.a or 3.1.2.2.d. The second ECCS subsystem charging pump shall satisfy the flow path requirements of Specification 3.1.2.2.b or 3.1.2.2.e.

** With pressurizer pressure $\geq$ 1750 psia.

SURVEILLANCE REQUIREMENTS (continued)

e. At least once per 18 months, during shutdown, by:

1. Verifying that each automatic valve in the flow paths actuates to its correct position on a Safety Injection Actuation Signal.

2. Verifying that each of the following pumps start automatically upon receipt of a Safety Injection Actuation Signal;

   a. High-Pressure Safety Injection Pumps.

   b. Low-Pressure Safety Injection Pumps.

   c. Charging Pumps.

3. Verifying that upon receipt of an actual or simulated Recirculation Actuation Signal: each low-pressure safety injection pump stops, each containment sump isolation valve opens, each refueling water tank outlet valve closes, and each safety injection system recirculation valve to the refueling water tank closes.

f. By verifying that each of the following pumps develops the specified total developed head when tested pursuant to the Inservice Testing Program.

1. High-Pressure Safety Injection pumps.

2. Low-Pressure Safety Injection pumps.

EMERGENCY CORE COOLING SYSTEMS

REFUELING WATER TANK

LIMITING CONDITION FOR OPERATION

3.5.4 The refueling water tank shall be OPERABLE with:

a. A minimum contained volume 477,360 gallons of borated water,

b. A minimum boron concentration of 1900 ppm,

c. A maximum water temperature of 100°F,

d. A minimum water temperature of 55°F when in MODES 1 and 2, and

e. A minimum water temperature of 40°F when in MODES 3 and 4

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the refueling water tank inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.5.4 The RWT shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the water level in the tank, and

2. Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWT temperature.

CONTAINMENT SYSTEMS

INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.4 Primary containment internal pressure shall be maintained between –0.7 and +0.5 psig.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the containment internal pressure outside of the limits above, restore the internal pressure to within the limits within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.4 The primary containment internal pressure shall be determined to be within the limits at least once per 12 hours.

TABLE 3.7-1

MAXIMUM ALLOWABLE POWER LEVEL-HIGH TRIP SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING OPERATION WITH BOTH STEAM GENERATORS

Maximum Number of Inoperable Safety Valves on Any Operating Steam Generator	Maximum Allowable Power Level-High Trip Setpoint (Percent of RATED THERMAL POWER)
1	88.5
2	79.8
3	66.5

TABLE 4.7-1

STEAM LINE SAFETY VALVES PER LOOP

	VALVE NUMBER		LIFT SETTING*
	Header A	Header B	
a.	8201	8205	≥ 955.3 psig and ≤ 1015.3 psig
b.	8202	8206	≥ 955.3 psig and ≤ 1015.3 psig
c.	8203	8207	≥ 955.3 psig and ≤ 1015.3 psig
d.	8204	8208	≥ 955.3 psig and ≤ 1015.3 psig
e.	8209	8213	≥ 994.1 psig and ≤ 1046.1 psig
f.	8210	8214	≥ 994.1 psig and ≤ 1046.1 psig
g.	8211	8215	≥ 994.1 psig and ≤ 1046.1 psig
h.	8212	8216	≥ 994.1 psig and ≤ 1046.1 psig

* +/-3% for valves a through d and +2%/-3% for valves e through h

PLANT SYSTEMS

CONDENSATE STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.7.1.3 The condensate storage tank shall be OPERABLE with a minimum contained volume of 153,400 gallons.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With the condensate storage tank inoperable, restore the condensate storage tank to OPERABLE status within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.3 The condensate storage tank shall be demonstrated OPERABLE at least once per 12 hours by verifying the water level.

3/4.8.1 A.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.1.1 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. Two physically independent circuits between the offsite transmission network and the onsite Class 1E distribution system, and

b. Two separate and independent diesel generator sets each with:

1. Engine-mounted fuel tanks containing a minimum of 152 gallons of fuel,

2. A separate fuel storage system containing a minimum of 19,000 gallons of fuel, and

3. A separate fuel transfer pump.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one offsite circuit of 3.8.1.1.a inoperable, except as provided in Action f. below, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter. Restore the offsite circuit to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

b. With one diesel generator of 3.8.1.1.b inoperable, demonstrate the OPERABILITY of the A.C. sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter; and if the EDG became inoperable due to any cause other than an inoperable support system, an independently testable component, or preplanned preventative maintenance or testing, demonstrate the OPERABILITY of the remaining OPERABLE EDG by performing Surveillance Requirement 4.8.1.1.2.a.4 within 8 hours, unless it can be confirmed that the cause of the inoperable EDG does not exist on the remaining EDG*; restore the diesel generator to OPERABLE status within 14 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Additionally, within 4 hours from the discovery of concurrent inoperability of required redundant feature(s) (including the steam driven auxiliary feed pump in MODE 1, 2, and 3), declare required feature(s) supported by the inoperable EDG inoperable if its redundant required feature(s) is inoperable.

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* If the absence of any common-cause failure cannot be confirmed, this test shall be completed regardless of when the inoperable EDG is restored to OPERABILITY.

SURVEILLANCE REQUIREMENTS (Continued)

b) Verifying the diesel starts on the auto-start signal****, energizes the emergency busses with permanently connected loads within 10 seconds, energizes the auto-connected shutdown loads through the load sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the shutdown loads. After energization, the steady-state voltage and frequency of the emergency busses shall be maintained at 4160 $\pm$ 210 volts and 60 $\pm$ 0.6 Hz during this test.

4. Verifying that on an ESF actuation test signal (without loss-of-offsite power) the diesel generator starts**** on the auto-start signal, and:

   a) Within 10 seconds, generator voltage and frequency shall be 4160 $\pm$ 420 volts and 60 $\pm$ 1.2 Hz.

   b) Operates on standby for greater than or equal to 5 minutes.

   c) Steady-state generator voltage and frequency shall be 4160 $\pm$ 210 volts and 60 $\pm$ 0.6 Hz and shall be maintained throughout this test.

5. Simulating a loss-of-offsite power in conjunction with an ESF actuation test signal, and

   a) Verifying deenergization of the emergency busses and load shedding from the emergency busses.

   b) Verifying the diesel starts on the auto-start signal****, energizes the emergency busses with permanently connected loads within 10 seconds, energizes the auto-connected emergency (accident) loads through the auto-sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the emergency loads. After energization, the steady-state voltage and frequency of the emergency busses shall be maintained at 4160 $\pm$ 210 volts and 60 $\pm$ 0.6 Hz during this test.

   c) Verifying that all automatic diesel generator trips, except engine overspeed and generator differential, are automatically bypassed upon loss of voltage on the emergency bus concurrent with a safety injection signal.

**** This test may be conducted in accordance with the manufacturer's recommendations concerning engine prelube period.

ELECTRICAL POWER SYSTEMS

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.1.2 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. One circuit between the offsite transmission network and the onsite Class 1E distribution system, and

b. One diesel generator set with:

1. Engine-mounted fuel tanks containing a minimum of 152 gallons of fuel,

2. A fuel storage system containing a minimum of 19,000 gallons of fuel, and

3. A fuel transfer pump.

APPLICABILITY: MODES 5 and 6.

ACTION:

With less than the above minimum required A.C. electrical power sources OPERABLE, immediately suspend all operations involving CORE ALTERATIONS, operations involving positive reactivity additions that could result in loss of required SHUTDOWN MARGIN or boron concentration, movement of irradiated fuel, or crane operation with loads over the fuel storage pool. In addition, when in MODE 5 with the reactor coolant loops not filled, or in MODE 6 with the water level less than 23 feet above the top of irradiated fuel assemblies seated within the reactor vessel, immediately initiate corrective action to restore the required sources to OPERABLE status as soon as possible.

SURVEILLANCE REQUIREMENTS

4.8.1.2.1 The above required A.C. electrical power sources shall be demonstrated OPERABLE by the performance of each of the Surveillance Requirements of 4.8.1.1.1 and 4.8.1.1.2 except for requirement 4.8.1.1.2a.5.

BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1 With the reactor vessel head unbolted or removed, the boron concentration of all filled portions of the Reactor Coolant System and the refueling cavity shall be maintained within the limit specified in the COLR.

APPLICABILITY: MODE 6*.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and initiate and continue boration at $\geq$ 40 gpm of greater than or equal to 1900 ppm boron or its equivalent to restore boron concentration to within limits.

SURVEILLANCE REQUIREMENTS

4.9.1.1 The boron concentration limit shall be determined prior to:

a. Removing or unbolting the reactor vessel head, and

b. Withdrawal of any full length CEA in excess of 3 feet from its fully inserted position.

4.9.1.2 The boron concentration of the refueling cavity shall be determined by chemical analysis at least 3 times per 7 days with a maximum time interval between samples of 72 hours.

* The reactor shall be maintained in MODE 6 when the reactor vessel head is unbolted or removed.

SPENT FUEL STORAGE POOL

LIMITING CONDITION FOR OPERATION

3.9.11 The Spent Fuel Pool shall be maintained with:

a. The fuel storage pool water level greater than or equal to 23 ft over the top of irradiated fuel assemblies seated in the storage racks, and

b. The fuel storage pool boron concentration greater than or equal to 1900 ppm.

APPLICABILITY: Whenever irradiated fuel assemblies are in the spent fuel storage pool.

ACTION:

a. With the water level requirement not satisfied, immediately suspend all movement of fuel assemblies and crane operations with loads in the fuel storage areas and restore the water level to within its limit within 4 hours.

b. With the boron concentration requirement not satisfied, immediately suspend all movement of fuel assemblies in the fuel storage pool and initiate action to restore the fuel storage pool boron concentration to within the required limit.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.11 The water level in the spent fuel storage pool shall be determined to be at least its minimum required depth at least once per 7 days when irradiated fuel assemblies are in the fuel storage pool.

4.9.11.1 Verify the fuel storage pool boron concentration is within limit at least once per 7 days.

DELETED

3/4.10 SPECIAL TEST EXCEPTIONS

SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.10.1 The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 may be suspended for measurement of CEA worth and shutdown margin provided reactivity equivalent to at least the highest estimated CEA worth is available for trip insertion from OPERABLE CEA(s).

APPLICABILITY: MODE 2.

ACTION:

a. With any full length CEA not fully inserted and with less than the above reactivity equivalent available for trip insertion, immediately initiate and continue boration at $\geq$ 40 gpm of 1900 ppm boric acid solution or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all full length CEAs inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at $\geq$ 40 gpm of 1900 ppm boric acid solution or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

SURVEILLANCE REQUIREMENTS

4.10.1.1 The position of each full length CEA required either partially or fully withdrawn shall be determined at least once per 2 hours.

4.10.1.2 Each CEA not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 7 days prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

<u>RADIOACTIVE EFFLUENTS</u>

<u>GAS STORAGE TANKS</u>

<u>LIMITING CONDITION FOR OPERATION</u>

3.11.2.6 The quantity of radioactivity contained in each gas storage tank shall be limited to less than or equal to 165,000 curies noble gases (considered as Xe-133).

<u>APPLICABILITY</u>: At all times.

<u>ACTION</u>:

a. With the quantity of radioactive material in any gas storage tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

<u>SURVEILLANCE REQUIREMENTS</u>

4.11.2.6 The quantity of radioactive material contained in each gas storage tank shall be determined to be within the above limit at least once per 24 hours when radioactive materials are being added to the tank when reactor coolant system activity exceeds 518.9 μCi/gram DOSE EQUIVALENT XE-133.

2.1.2 SHIELD BUILDING

a. Minimum annular space = 4 feet

b. Annulus nominal volume = 543,000 cubic feet

c. Nominal outside height (measured from top of foundation base to the top of the dome) = 230.5 feet

d. Nominal inside diameter = 148 feet

e. Cylinder wall minimum thickness = 3 feet

f. Dome minimum thickness = 2.5 feet

g. Dome inside radius = 112 feet

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The containment vessel is designed and shall be maintained for a maximum internal pressure of 44 psig and a temperature of 264°F.

PENETRATIONS

5.2.3 Penetrations through the containment structure are designed and shall be maintained in accordance with the original design provisions contained in Sections 3.8.2.1.10 and 6.2.4 of the FSAR with allowance for normal degradation pursuant to the applicable Surveillance Requirements.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 217 fuel assemblies. Each assembly shall consist of a matrix of Zircaloy clad fuel rods and/or poison rods, with fuel rods having an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in non-limiting core regions.

5.3.1.1 Except for special test as authorized by the NRC, all fuel assemblies under control element assemblies shall be sleeved with a sleeve design previously approved by the NRC.

CONTROL ELEMENT ASSEMBLIES

5.3.2 The reactor core shall contain 73 full length and no part length control element assemblies. The control element assemblies shall be designed and maintained in accordance with the original design provisions contained in Section 4.2.3.2 of the FSAR with allowance for normal degradation pursuant to the applicable Surveillance Requirements.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained:

a. 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,

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 700°F.

5.5 EMERGENCY CORE COOLING SYSTEMS

5.5.1 The emergency core cooling systems are designed and shall be maintained in accordance with the original design provisions contained in Section 6.3 of the FSAR with allowance for normal degradation pursuant to the applicable Surveillance Requirements.

5.6 FUEL STORAGE

CRITICALITY

5.6.1.a The spent fuel pool and spent fuel storage racks shall be maintained with:

1. k_{eff} less than 1.0 when fully flooded with unborated water, which includes an allowance for biases and uncertainties as described in Section 9.1 of the Updated Final Safety Analysis Report.

2. A nominal 10.12 inches center to center distance between fuel assemblies in Region 1 of the spent fuel pool storage racks, a nominal 10.30 inches center to center distance between fuel assemblies in the Region 1 cask pit storage rack, and a nominal 8.86 inches center to center distance between fuel assemblies in Region 2 of the spent fuel pool storage racks.

3. A k_{eff} less than or equal to 0.95 when flooded with water containing 500 ppm boron, including an allowance for biases and uncertainties as described in Section 9.1 of the Updated Final Safety Analysis Report.

CRITICALITY (Continued)

4. For storage of enriched fuel assemblies, requirements of Criteria in 5.6.1.a.1 and 5.6.1.a.3 shall be met by positioning fuel in the spent fuel storage racks consistent with the requirements of Specification 5.6.1.c.

5. Vessel Flux Reduction Assemblies (VFRAs), as defined in Section 9.1 of the Updated Final Safety Analysis Report, may be placed in any allowable fuel storage location.

6. Fissile material, not contained in a fuel assembly lattice, shall be stored in accordance with the requirements of Criteria in 5.6.1.a.1 and 5.6.1.a.3.

7. The Metamic neutron absorber inserts shall have a ^{10}B areal density greater than or equal to 0.015 grams $^{10}B/cm^2$.

b. The Region 1 cask pit storage rack shall contain neutron absorbing material (Boral) between stored fuel assemblies when installed in the spent fuel pool.

c. Loading of spent fuel storage racks shall be controlled as described below. Criteria in 5.6.1.c.2, 5.6.1.c.3, 5.6.1.c.5 and 5.6.1.c.6 do not apply to the Region 1 cask pit storage rack.

1. The maximum initial planar average U-235 enrichment of any fuel assembly inserted in a spent fuel storage rack shall be less than or equal to 4.6 weight percent.

2. Fuel placed in Region 1 of the spent fuel pool storage racks shall comply with the storage patterns and alignment restrictions of Figure 5.6-1 and the minimum burnup requirements of Table 5.6-1.

3. Fuel placed in Region 2 of the spent fuel pool storage racks shall comply with the storage patterns or allowed special arrangements of Figure 5.6-2 and the minimum burnup requirements of Table 5.6-1. The allowed special arrangement for fresh fuel may be repeated, provided the applicable interface requirements specified by the safety analysis are met.

4. Any fuel satisfying criteria 5.6.1.c.1, including fresh fuel, may be placed in the Region 1 cask pit storage rack.

5. The same directional orientation for Metamic inserts is required for contiguous groups of 2x2 arrays where Metamic inserts are required.

6. Any 2x2 array of Region 2 storage cells that interface with Region 1 shall comply with the rules of Figure 5.6-3. The allowed special arrangement in Region 2 as shown in Figure 5.6-2 shall not be placed adjacent to Region 1.

d. The new fuel storage racks are designed for dry storage of unirradiated fuel assemblies having a maximum planar average U-235 enrichment less than or equal to 4.6 weight percent, while maintaining a k_{eff} of less than or equal to 0.98 under the most reactive condition.

Allowable Checkerboard Storage Patterns
(See Notes 4 and 5)

Pattern "A"
See Note 1

F	WH
WH	F

Pattern "B"
See Note 2

1	1
1	1

Pattern "C"
See Note 3

2	2
WH	2

NOTES:

1. F represents Fresh Fuel. WH represents an empty cell. Allowable Pattern is Fresh Fuel checkerboarded with empty cells. Diagram is for illustration only.

2. Numbering denotes fuel assembly type. Minimum burnup for fuel assembly type 1 is defined in Table 5.6-1. Allowable pattern is at least one insert [either Metamic or full-length full-strength CEA] in any one of the 2x2 array locations. Diagram is for illustration only.

3. Numbering denotes fuel assembly type. WH represents an empty cell. Minimum burnup for fuel assembly type 2 is defined in Table 5.6-1. Allowable pattern is at least one empty cell in any of the 2x2 array locations. Diagram is for illustration only.

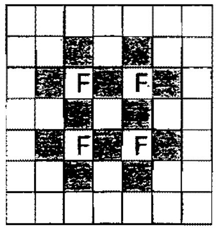
4. The storage arrangements of fuel within a rack module may contain more than one pattern. Each cell is a part of up to four 2x2 arrays, and each cell must simultaneously meet the requirements of all those arrays of which it is a part.

5. Empty cells within any pattern are acceptable.

FIGURE 5.6-1
Allowable Region 1 Storage Patterns and Fuel Arrangements

ALLOWED SPECIAL ARRANGEMENT

Fresh Fuel Assemblies in Pattern "C", "D", or "E" Racks



F = FRESH FUEL ASSEMBLY

= EMPTY CELL

ALLOWABLE CHECKERBOARD STORAGE PATTERNS (See Notes 4 and 5)

Pattern "C"
See Note 1

Pattern "D"
See Note 2

Pattern "E"
See Note 3

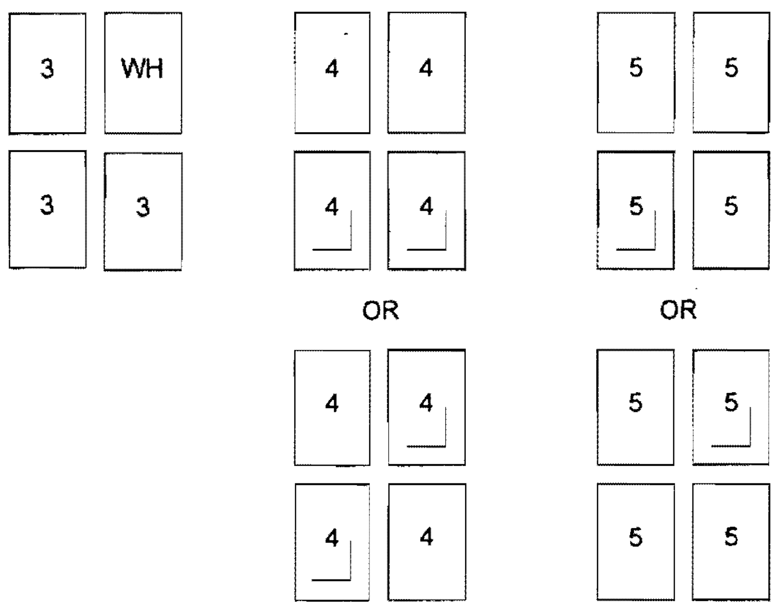


FIGURE 5.6-2 (Sheet 1 of 2)
Allowable Region 2 Storage Patterns and Fuel Alignments

NOTES to Figure 5.6-2

NOTES:

1. Numbering denotes fuel assembly type. WH represents an empty cell. Minimum burnup for fuel assembly type 3 is defined in Table 5.6-1. Allowable pattern is at least one empty cell in any of the 2x2 array locations. Diagram is for illustration only.

2. Numbering denotes fuel assembly type. Minimum burnup for fuel assembly type 4 is defined in Table 5.6-1. Allowable pattern is at least two inserts, (either Metamic or full-length, full-strength CEA) in the 2x2 array. Diagrams are for illustration only.

3. Numbering denotes fuel assembly type. Minimum burnup for fuel assembly type 5 is defined in Table 5.6-1. Allowable pattern is one insert, (either Metamic or full-length, full-strength CEA) in the 2x2 array. Diagrams are for illustration only.

4. The storage arrangements of fuel within a rack module may contain more than one pattern. Each cell is a part of up to four 2x2 arrays, and each cell must simultaneously meet the requirements of all those arrays of which it is a part.

5. Empty cells within any pattern are acceptable.

FIGURE 5.6-2 (Sheet 2 of 2)
Allowable Region 2 Storage Patterns and Fuel Arrangements

Allowed Region 2 to Region 1 Fuel Alignments
(See Notes 4 and 5)

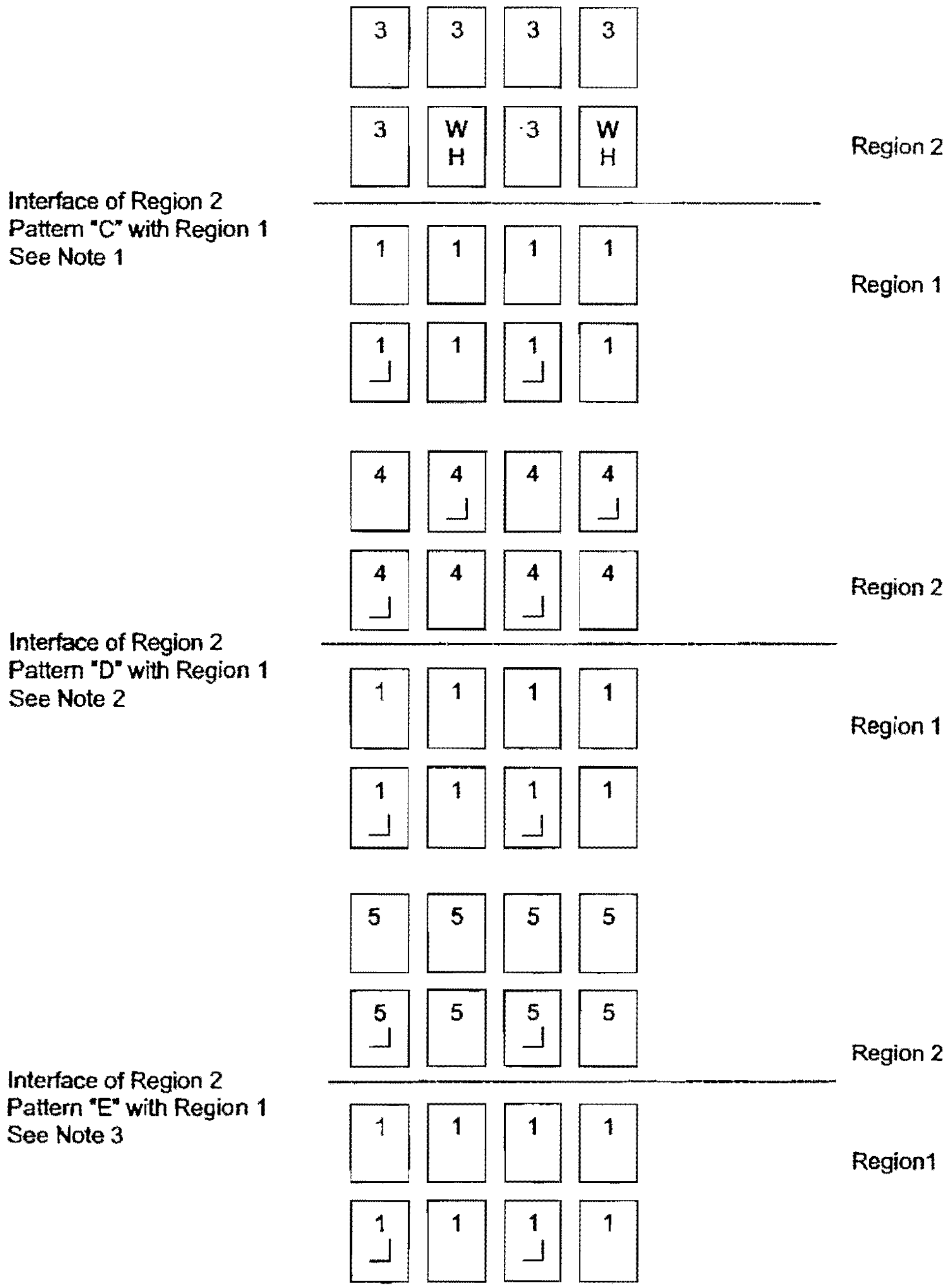


FIGURE 5.6-3 (Sheet 1 of 2)
Region 2 Interface Requirements with Region 1

NOTES to Figure 5.6-3

NOTES:

1. WH represents an empty cell. For the interface of Pattern "C" with Region 1, the empty cell must be on the rack periphery facing Region 1 racks. Diagrams are for illustration only.

2. For the interface of pattern "D" with Region 1, at least one cell on the rack periphery facing Region 1 rack must contain an insert (either Metamic of full-length full-strength CEA) in the 2x2 array. If the insert is Metamic, the insert must be oriented so that the corner of the L-shape is located closest to the Region 1 rack. Diagram is for illustration only.

3. For the interface of Pattern "E" with Region 1, the insert must be on the rack periphery facing the Region 1 rack. The insert may be either a Metamic of full-length full strength CEA. If the insert is Metamic, the insert must be oriented so that the corner of the L-shape is located closest to the Region 1 rack. Diagram is for illustration only.

4. Empty cells with any pattern are acceptable.

5. There are no interface requirements within Region 1. Any Pattern within Region 1 may be used for the interface. Pattern "B" was used only as an illustration.

FIGURE 5.6-3 (Sheet 2 of 2)
Region 2 Interface requirements with Region 1

TABLE 5.6-1
Minimum Burnup as a Function of Enrichment

Fuel Type	Cooling Time (Years)	Coefficients		
		A	B	C
1	0	-36.6860	22.4942	-1.4413
2	0	-36.1742	16.6000	-0.8958
3	0	-34.7091	23.1361	-1.6204
4	0	-24.5145	21.3404	-1.2444
	2.5	-26.8311	22.5246	-1.5029
	5	-24.7233	20.9763	-1.3246
	10	-23.6285	19.9541	-1.2505
	15	-23.5458	19.9336	-1.3180
	20	-22.4382	19.2460	-1.2629
5	0	-8.1856	14.5275	-0.0719
	2.5	-11.8506	16.1475	-0.3969
	5	-16.5196	18.5309	-0.7837
	10	-13.6831	16.3475	-0.5844
	15	-12.5819	15.6175	-0.5656
	20	-12.6469	16.4575	-0.5906

NOTES:

1. To qualify in a "fuel type," the burnup of a fuel assembly must exceed the minimum burnup "BU" calculated by inserting the "coefficients" for the associated "fuel type" and "cooling time" into the polynomial function:

   $BU = A + B*E + C*E^2$, where:

   BU = Minimum Burnup (GWD/MTU)

   E = Initial Maximum Planar Average Enrichment (weight percent uranium-235)

   A, B, C = Coefficients

2. Interpolation between values of cooling time is not permitted.

(2) conform to the guidance of Appendix I to 10 CFR Part 50, and
(3) include the following:

1) Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters in the ODCM.

2) A Land Use Census to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the monitoring program are made if required by the results of this census, and

3) Participation in a Interlaboratory Comparison Program to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring.

h. Containment Leakage Rate Testing Program

A program 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 is in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," as modified by the following exception(s):

a) Bechtel Topical Report, BN-TOP-1 or ANS 56.8-1994 (as recommended by R.G. 1.163) will be used for type A testing.

b) The first Type A test performed after the May 1993 Type A test shall be no later than May 2008.

The peak calculated containment internal pressure for the design basis loss of coolant accident P_a, is 42.8 psig. The containment design pressure is 44 psig.

The maximum allowed containment leakage rate, L_a, at P_a, shall be 0.50% of containment air weight per day.

Leakage rate acceptance criteria are:

a. 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 < 0.60 L_a for the Type B and C tests, $\leq$ 0.75 L_a for Type A tests, and $\leq$ 0.096 L_a for secondary containment bypass leakage paths.

b. Air lock testing acceptance criteria are:

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

2) For the personnel air lock door seal, leakage rate is < 0.01 L_a when pressurized to $\geq$ 1.0 P_a.

3) For the emergency air lock door seal, leakage rate is < 0.01 L_a when pressurized to $\geq$ 10 psig.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT (continued)

6.9.1.9 At least once every 5 years, an estimate of the actual population within 10 miles of the plant shall be prepared and submitted to the NRC.

6.9.1.10 At least once every 10 years, an estimate of the actual population within 50 miles of the plant shall be prepared and submitted to the NRC.

6.9.1.11 CORE OPERATING LIMITS REPORT (COLR)

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

Specification 3.1.1.1	Shutdown Margin – T_{avg} Greater Than 200°F
Specification 3.1.1.2	Shutdown Margin – T_{avg} Less Than or Equal to 200°F
Specification 3.1.1.4	Moderator Temperature Coefficient
Specification 3.1.3.1	Full Length CEA Position – Misalignment > 15 inches
Specification 3.1.3.6	Regulating CEA Insertion Limits
Specification 3.2.1	Linear Heat Rate
Specification 3.2.3	Total Integrated Radial Peaking Factor – F_r^T
Specification 3.2.5	DNB Parameters
Specification 3.9.1	Refueling Operations – Boron Concentration

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, as described in the following documents, approved Revisions and Supplements as specified in the COLR.

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

2. NF-TR-95-01, "Nuclear Physics Methodology for Reload Design of Turkey Point & St. Lucie Nuclear Plants," Florida Power & Light Company, January 1995.

3. XN-75-27(A) [also issued as XN-NF-75-27(A)], "Exxon Nuclear Neutronic(s) Design Methods for Pressurized Water Reactors"

4. DELETED

5. XN-NF-82-21(P)(A), "Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations"

CORE OPERATING LIMITS REPORT (continued)

6. DELETED

7. XN-75-32(P)(A), "Computational Procedure for Evaluating Fuel Rod Bowing"

8. DELETED

9. XN-NF-78-44(NP)(A), "A Generic Analysis of the Control Rod Ejection Transient for Pressurized Water Reactors"

10. XN-NF-621(P)(A), " Exxon Nuclear DNB Correlation for PWR Fuel Designs"

11. DELETED

12. XN-NF-82-06(P)(A), "Qualification of Exxon Nuclear Fuel for Extended Burnup"

13. ANF-88-133(P)(A), "Qualification of Advanced Nuclear Fuels' PWR Design Methodology for Rod Burnups of 62 GWd/MTU"

14. XN-NF-85-92 (P)(A), "Exxon Nuclear Uranium Dioxide/Gadolinia Irradiation Examination and Thermal Conductivity Results"

15. DELETED

16. DELETED

17. EMF-92-116(P)(A), "Generic Mechanical Design Criteria for PWR Fuel Design"

18. EMF-92-153(P)(A), "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel"

19. EMF-96-029(P)(A), Volumes 1 and 2, "Reactor Analysis System for PWRs Volume 1 – Methodology Description, Volume 2 – Benchmarking Results"

CORE OPERATING LIMITS REPORT (continued)

20. EMF-1961(P)(A), "Statistical Setpoint/Transient Methodology for Combustion Engineering Type Reactors"

21. EMF-2310(P)(A), "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors," Revision 1, as supplemented by ANP-3000(P), "St. Lucie Unit 1 EPU - Information to Support License Amendment Request," Revision 0.

22. EMF-2328(P)(A), "PWR Small Break LOCA Evaluation Model, S-RELAP5 Based," Revision 0, as supplemented by ANP-3000(P), "St. Lucie Unit 1 EPU - Information to Support License Amendment Request," Revision 0.

23. EMF-2103(P)(A), "Realistic Large Break LOCA Methodology for Pressurized Water Reactors," Revision 0, as supplemented by ANP-2903(P), "St. Lucie Nuclear Plant Unit 1 EPU Cycle Realistic Large Break LOCA summary Report with Zr-4 Fuel Cladding," Revision 1.