3/4 8-6	3/4 8-6
3/4 8-9	3/4 8-9
3/4 9-1	3/4 9-1
3/4 10-1	3/4 10-1
3/4 11-15	3/4 11-15
6-15b	6-15b
6-20	6-20
6-20a	6-20a
6-20c	6-20c
6-20d	6-20d
6-20e	6-20e
6-20f	6-20f

neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

D. Pursuant to the Act and 10 CFR Parts 30, 40, and 70, FPL 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; and

E. Pursuant to the Act and 10 CFR Parts 30, 40, and 70, FPL to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

3. This renewed license shall be deemed to contain and is subject to the conditions specified in the following Commission's regulations: 10 CFR Part 20, Section 30.34 of 10 CFR Part 30, Section 40.41 of 10 CFR Part 40, Section 50.54 and 50.59 of 10 CFR Part 50, and Section 70.32 of 10 CFR Part 70; 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; and is subject to the additional conditions specified below:

A. Maximum Power Level

FPL is authorized to operate the facility at steady state reactor core power levels not in excess of 3020 megawatts (thermal).

B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 163 are hereby incorporated in the renewed license. FPL shall operate the facility in accordance with the Technical Specifications.

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 6.15.d, shall be within 36 months in a staggered test basis, plus the 138 days allowed by SR 4.0.2, as measured from November 13, 2006, which is the date of the most recent successful pressure measurement test, or within 138 days if not performed previously.

N. FATES3B Safety Analyses

FATES3B has been specifically approved for use for St. Lucie Unit 2 licensing basis analyses based on FPL maintaining the more restrictive operational/design radial power fall-off curve limits as specified in Attachment 4 to FPL Letter L-2012-121, dated March 31, 2012 as compared to the FATES3B analysis radial power fall-off curve limits. The radial power fall-off curve limits shall be verified each cycle as part of the Reload Safety Analysis Checklist (RSAC) process.

Upon NRC approval of a new long-term fuel evaluation model and associated methods that explicitly account for thermal conductivity degradation (TCD) that is applicable to St. Lucie Unit 2 design, FPL will, within 6 months:

(a) Demonstrate that the St. Lucie Unit 2 safety analyses remain conservatively bounded in licensing basis analyses when compared to the NRC-approved new long-term fuel evaluation model that is applicable to St. Lucie Unit 2 design, or

(b) Provide a schedule for re-analysis using the NRC-approved new long-term fuel evaluation model that is applicable to St. Lucie Unit 2 design for any affected licensing basis analyses.

4. This renewed license is effective as of the date of issuance, and shall expire at midnight April 6, 2043.

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
3. Appendix C, Antitrust Conditions
4. Appendix D, Antitrust Conditions

Date of Issuance: October 2, 2003

INDEX

DEFINITIONS

INDEX

LIST OF FIGURES

LIST OF FIGURES (continued)

INDEX

LIST OF TABLES

DOSE EQUIVALENT I-131

1.10 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/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 RESPONSE 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.

IDENTIFIED 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.

OPERABLE – OPERABILITY

1.19 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

1.20 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

1.21 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and (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

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

PROCESS CONTROL PROGRAM (PCP)

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

PURGE – PURGING

1.24 PURGE or PURGING is the 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.

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.

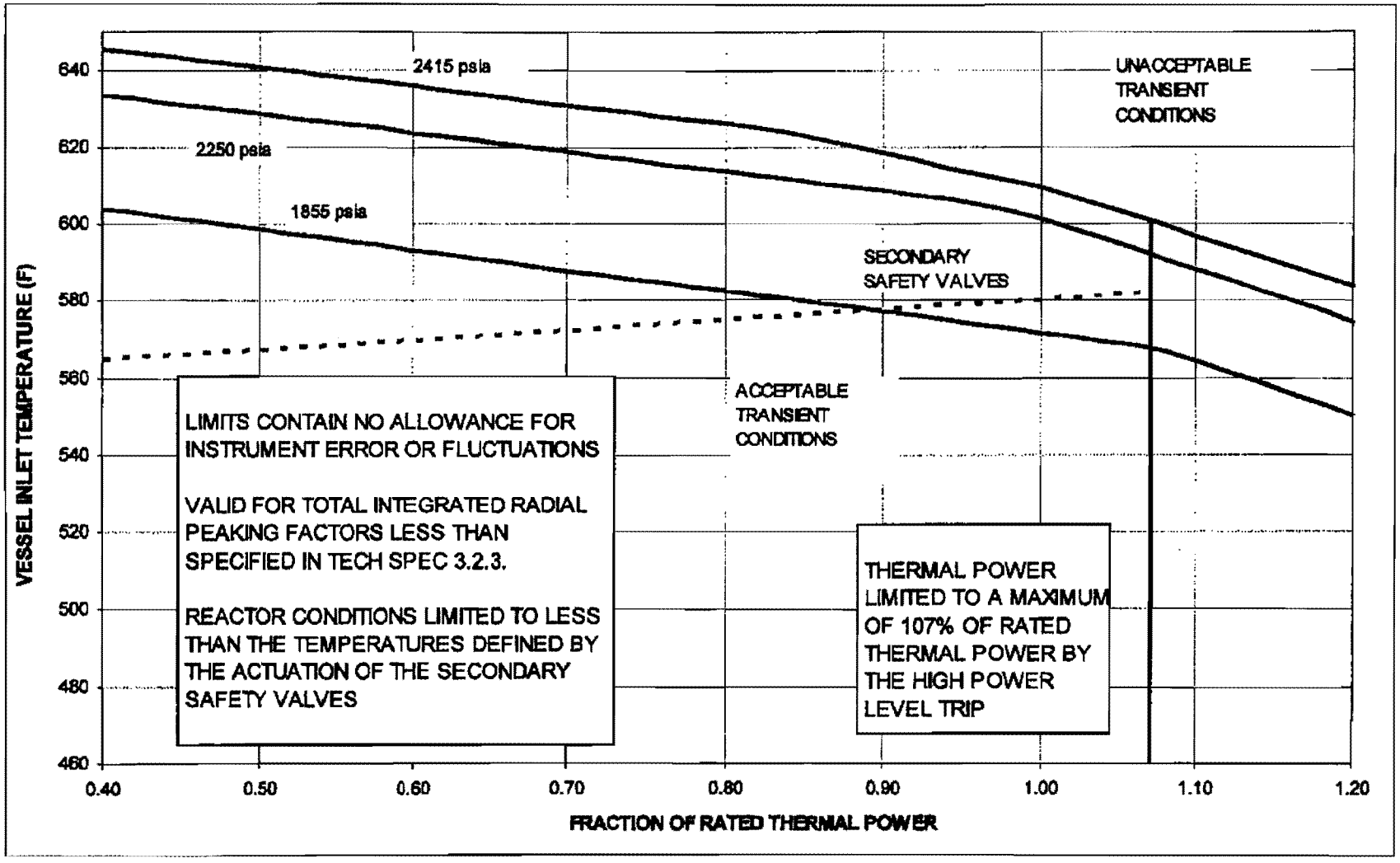

FIGURE 2.1-1: REACTOR CORE THERMAL MARGIN SAFETY LIMIT LINES FOUR REACTOR COOLANT PUMPS OPERATING

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. Variable 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. Pressurizer Pressure – High	≤ 2370 psia	≤ 2374 psia
4. Thermal Margin/Low Pressure[1]		
Four Reactor Coolant Pumps Operating	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-3 and 2.2-4. Minimum value of 1900 psia.	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-3 and 2.2-4. Minimum value of 1900 psia.
5. Containment Pressure – High	≤ 3.0 psig	≤ 3.1 psig
6. Steam Generator Pressure – Low	≥ 626.0 psia[2]	≥ 621.0 psia[2]
7. Steam Generator Pressure Difference – High (Logic in TM/LP Trip Unit)[1]	≤ 120.0 psid	≤ 132.0 psid
8. Steam Generator Level – Low	≥ 35.0%[3]	≥ 35.0%[3]

TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
9. Local Power Density – High[5] Operating	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-1 and 2.2-2	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-1 and 2.2-2.
10. Loss of Component Cooling Water to Reactor Coolant Pumps – Low	≥ 636 gpm**	≥ 636 gpm
11. Reactor Protection System Logic	Not Applicable	Not Applicable
12. Reactor Trip Breakers	Not Applicable	Not Applicable
13. Rate of Change of Power – High[4]	≤ 2.49 decades per minute	≤ 2.49 decades per minute
14. Reactor Coolant Flow – Low[1]	≥ 95.4% of minimum Reactor Coolant flow with four pumps operating*	≥ 94.9% of minimum Reactor Coolant flow with four pumps operating*
15. Loss of Load (Turbine) Hydraulic Fluid Pressure – Low[5]	≥ 800 psig	≥ 800 psig

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

** 10-minute time delay after relay actuation.

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - T_{avg} GREATER THAN 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 outside the COLR limits, immediately initiate and continue boration at greater than or equal to 40 gpm of a solution containing 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 verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable CEA(s).

b. When in MODE 1 or MODE 2 with Keff greater than or equal to 1.0, 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 with Keff less than 1.0, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical CEA position is within the limits of Specification 3.1.3.6.

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* See Special Test Exception 3.10.1.

REACTIVITY CONTROL SYSTEMS

SHUTDOWN MARGIN - T_{avg} LESS THAN OR EQUAL TO 200°F

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN outside the COLR limits, immediately initiate and continue boration at greater than or equal to 40 gpm of a solution containing 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 shall be determined to be within the COLR limits:

a. Within 1 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 4.1.1.2b and by verifying at least two charging pumps are rendered inoperable by racking out their motor circuit breakers.

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 be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to its COLR limit at 200 °F within the next 6 hours; restore at least two flow paths to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

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 3550 gallons of 3.1 to 3.5 weight percent boric acid (5420 to 6119 ppm boron).

b. The refueling water tank with:

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

2. A minimum boron concentration of 1900 ppm, and

3. A solution temperature between 40°F and 120°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

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

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 contained borated water volume 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 outside air temperature is outside the range of 40°F and 120°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.

REACTIVITY CONTROL SYSTEMS

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 2A in accordance with Figure 3.1-1.

b. Boric Acid Makeup Tank 2B in accordance with Figure 3.1-1.

c. Boric Acid Makeup Tanks 2A and 2B 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 borated water volume of 477,360 gallons,

2. A boron concentration of between 1900 and 2200 ppm of boron, and

3. A solution temperature of between 55°F and 100°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the above required boric acid makeup tank(s) inoperable, restore the tank(s) to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to its COLR limit at 200°F; restore the above required boric acid makeup tank(s) to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the refueling water tank inoperable, restore the tank to OPERABLE status 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

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

a. At least once per 7 days by:

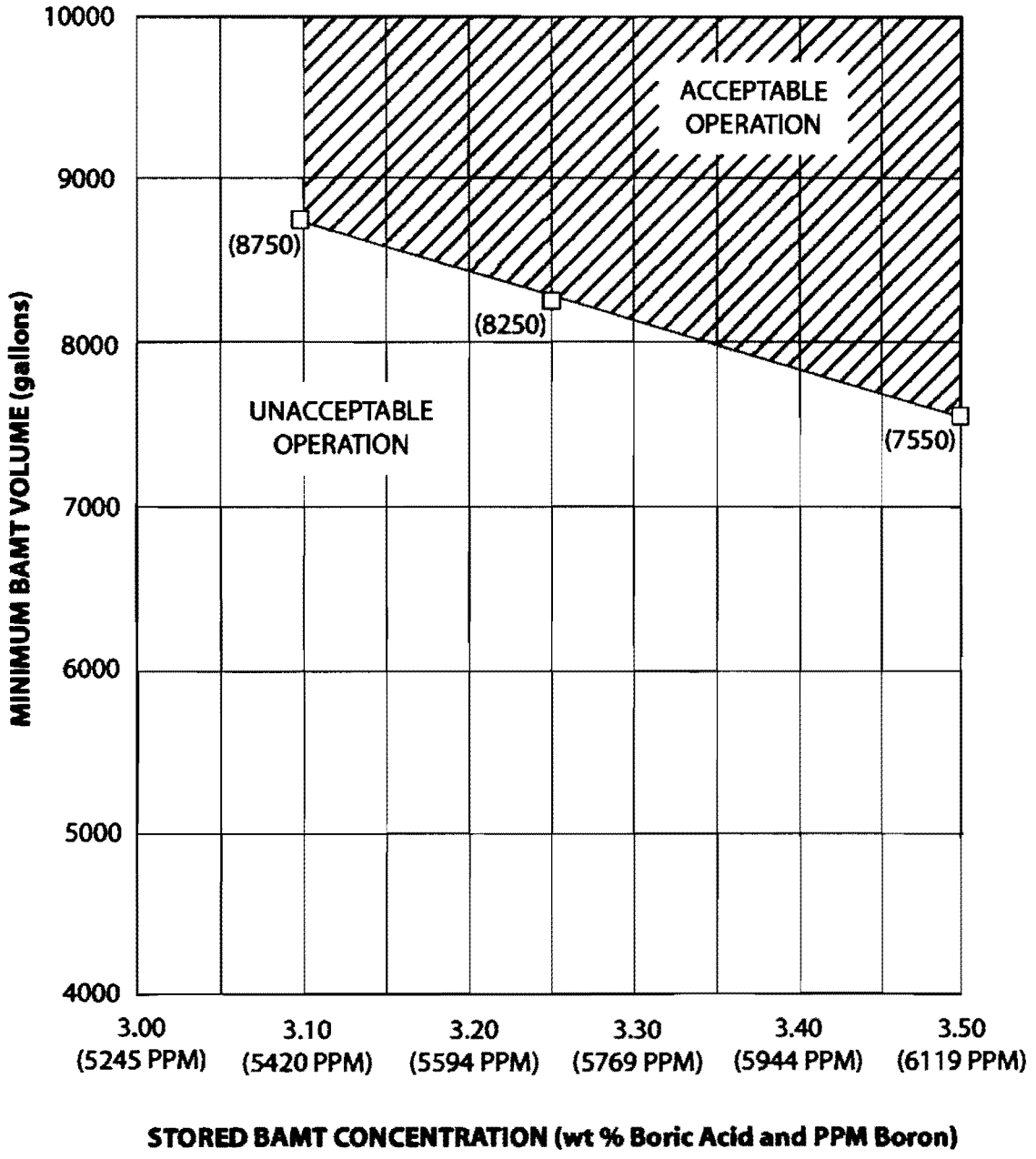
1. Verifying the boron concentration in the water and

2. Verifying the contained borated water volume of the water source.

b. At least once per 24 hours by verifying the RWT temperature when the outside air temperature is outside the range of 55°F and 100°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 is greater than 55°F.

FIGURE 3.1-1
MINIMUM BAMT VOLUME vs STORED BORIC ACID CONCENTRATION



REACTIVITY CONTROL SYSTEMS

CEA DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual full-length (shutdown and regulating) CEA drop time, from a fully withdrawn position, shall be less than or equal to 3.25 seconds from when the electrical power is interrupted to the CEA drive mechanism until the CEA reaches its 90% insertion position with:

a. T_{avg} greater than or equal to 515°F, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With the drop time of any full-length CEA determined to exceed the above limit:

1. If in MODE 1 or 2, be in at least HOT STANDBY within 6 hours, or

2. If in MODE 3, 4, or 5, restore the CEA drop time to within the above limit prior to proceeding to MODE 1 or 2.

b. With the CEA drop times within limits but determined at less than full reactor coolant flow, operation may proceed provided THERMAL POWER is restricted to less than or equal to the maximum THERMAL POWER level allowable for the reactor coolant pump combination operating at the time of CEA drop time determination.

SURVEILLANCE REQUIREMENTS

4.1.3.4 The CEA drop time of full-length CEAs shall be demonstrated through measurement prior to reactor criticality:

a. For all CEAs following each removal and installation of the reactor vessel head,

b. For specifically affected individual CEAs following any maintenance on or modification to the CEA drive system which could affect the drop time of those specific CEAs, and

c. At least once per 18 months.

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 3.2-2 of the COLR,

b. Pressurizer Pressure* as shown on Table 3.2-2 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 $\leq$ 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 $\geq$ 90% of RATED THERMAL POWER.

DELETED

TABLE 4.3-1

REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS

	FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1.	Manual Reactor Trip	N/A	N.A.	S/U(1)	1, 2, 3*, 4*, 5*
2.	Variable Power Level – High				
	a. Nuclear Power	S	D(2), M(3), Q(4)	M	1,2
	b. ΔT Power	S	D(5), Q(4)		1
3.	Pressurizer Pressure – High	S	R	M	1, 2
4.	Thermal Margin/Low Pressure	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 Pressure Difference – High	S	R	M	1, 2
8.	Steam Generator Level – Low	S	R	M(8, 9)	1, 2
9.	Local Power Density – High	S	R	M	1
10.	Loss of Component Cooling Water to Reactor Coolant Pumps	N.A.	N.A.	M	N.A.
11.	Reactor Protection System Logic	N.A.	N.A.	M(7)	1, 2, 3*, 4*, 5*

TABLE 4.3-1 (Continued)

TABLE NOTATION

* - Only if the reactor trip breakers are in the closed position and the CEA drive system is capable of CEA withdrawal.

(1) - Each startup or when required with the reactor trip breakers closed and the CEA drive system capable of rod withdrawal, if not performed in the previous 7 days.

(2) - Heat balance only (CHANNEL FUNCTIONAL TEST not included), above 15% of RATED THERMAL POWER; adjust "Nuclear Power Calibrate" potentiometer to null "Nuclear Power – ΔT Power". During PHYSICS TESTS, these daily calibrations 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) - Neutron detectors may be excluded from CHANNEL CALIBRATION.

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

(6) - At least once per 18 months and following maintenance or adjustment of the reactor trip breakers, the CHANNEL FUNCTIONAL TEST shall include verification of the independent OPERABILITY of the undervoltage and shunt trips.

(7) - The fuse circuitry in the matrix fault protection circuitry shall be determined to be OPERABLE by testing with the installed test circuitry.

(8) - If the as-found channel 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.

(9) - 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 UFSAR Section 7.2.

REACTOR COOLANT SYSTEM

OPERATING

LIMITING CONDITION FOR OPERATION

3.4.2.2 All pressurizer code safety valves shall be OPERABLE with a lift setting of $\geq$ 2410.3 psig and $\leq$ 2560.3 psig.*

APPLICABILITY: MODES 1, 2, 3, and 4 with all RCS cold leg temperatures > 230°F.

ACTION:

a. With one pressurizer code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in HOT STANDBY within 6 hours and in HOT SHUTDOWN within the next 6 hours.

b. With two or more pressurizer code safety valves inoperable, be in HOT STANDBY within 6 hours and in HOT SHUTDOWN with all RCS cold leg temperatures at $\leq$ 230°F within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.2.2 Verify each pressurizer code safety valve is OPERABLE in accordance with the Inservice Testing Program. Following testing, as-left lift settings shall be within +/- 1% of 2500 psia.

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* The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

3/4.4.8 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

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

a. Less than or equal to 1.0 microcurie/gram DOSE EQUIVALENT I-131, and

b. Less than or equal to 518.9 microcuries/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 ≤ 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 ≤ 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 specific activity exceeds 1 micro-Ci/gram, DOSE EQUIVALENT I-131, and	1#, 2#, 3#, and 4#
		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

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

DELETED

FIGURE 3.4-2
ST. LUCIE UNIT 2 REACTOR COOLANT SYSTEM PRESSURE-TEMPERATURE LIMITS FOR 47 EFPY, HEATUP, CORE CRITICAL, AND INSERVICE TEST

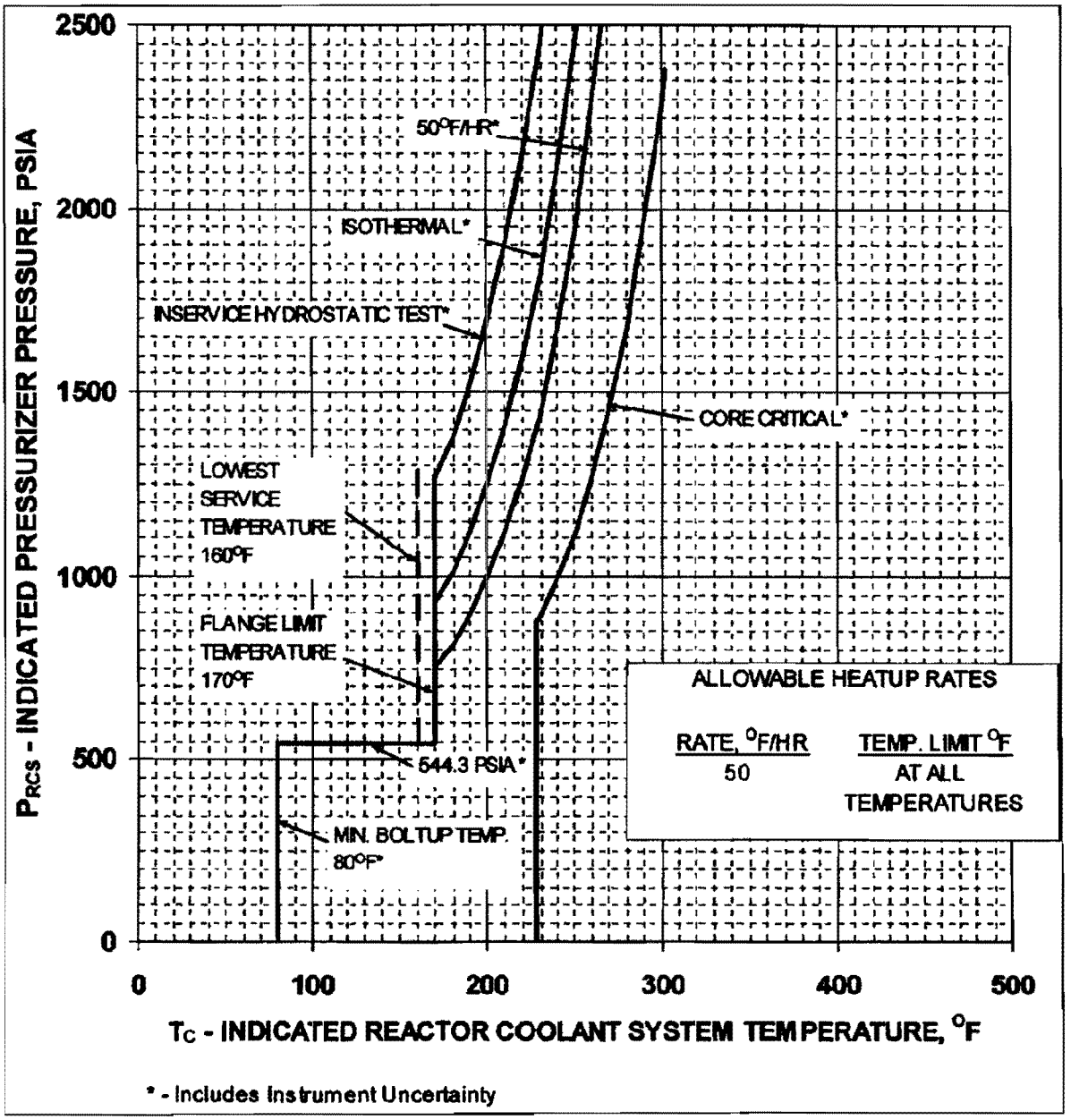

FIGURE 3.4-3
ST. LUCIE UNIT 2 REACTOR COOLANT SYSTEM PRESSURE-TEMPERATURE LIMITS FOR 47 EFPY, COOLDOWN, AND INSERVICE TEST

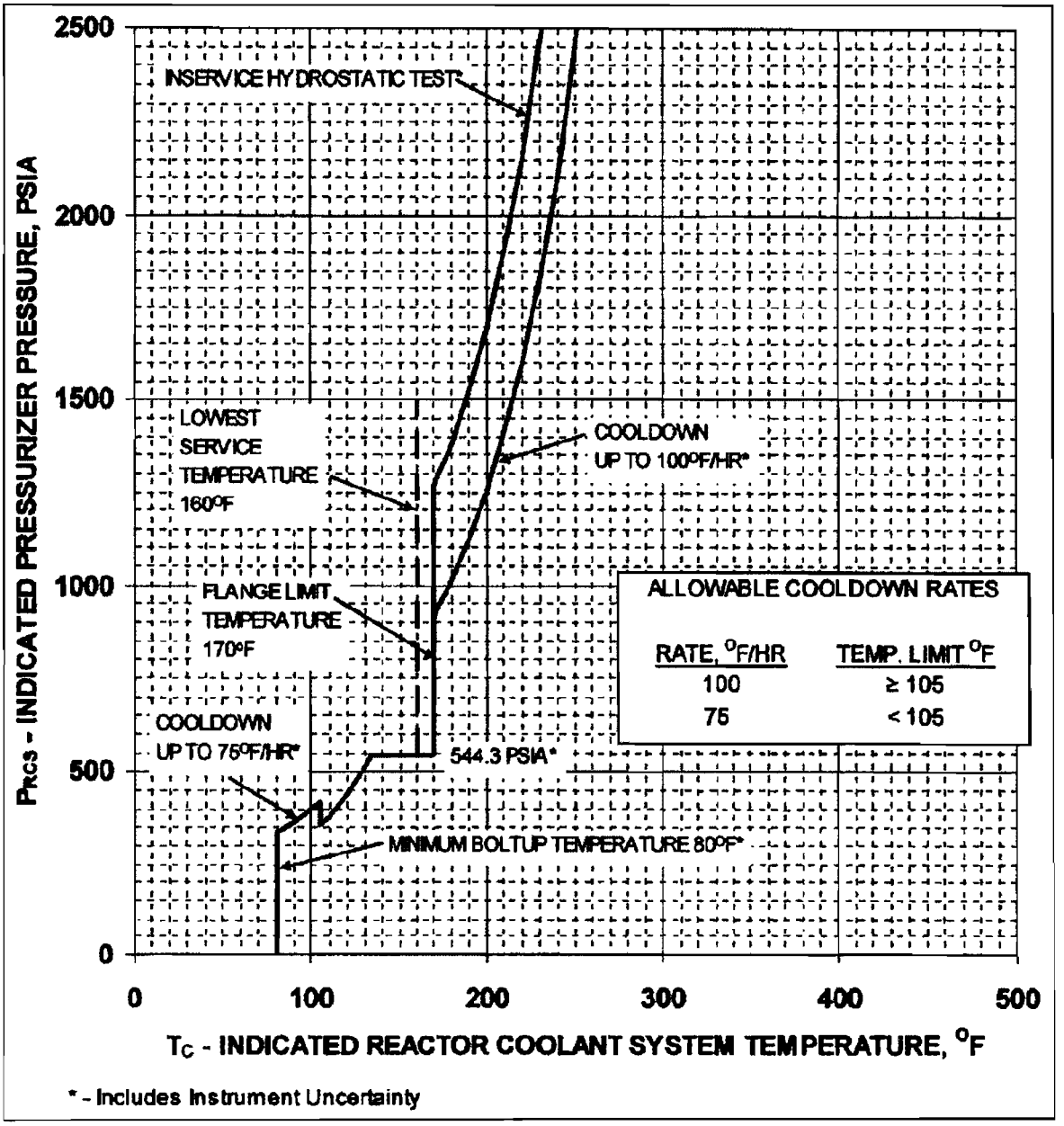

TABLE 3.4-3

LOW TEMPERATURE RCS OVERPRESSURE PROTECTION RANGE

Operating Period, EFPY	Cold Leg Temperature, °F During Heatup	Cold Leg Temperature, °F During Cooldown
≤ 47	≤ 246	≤ 224

TABLE 3.4-4

MINIMUM COLD LEG TEMPERATURE FOR PORV USE FOR LTOP

Operating Period EFPY	Cold Leg Temperature, °F During Heatup	Cold Leg Temperature, °F During Cooldown
≤ 47	80	132

3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3/4.5.1 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. A contained borated water volume of between 1420 and 1556 cubic feet,

c. A boron concentration of between 1900 and 2200 ppm of boron, and

d. A nitrogen cover-pressure of between 500 and 650 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

4.5.1.1 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.

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* With pressurizer pressure greater than or equal to 1750 psia. When pressurizer pressure is less than 1750 psia, at least three safety injection tanks shall be OPERABLE, each with a minimum pressure of 235 psig and a maximum pressure of 650 psig and a contained water volume of between 1250 and 1556 cubic feet with a boron concentration of between 1900 and 2200 ppm of boron. With all four safety injection tanks OPERABLE, each tank shall have a minimum pressure of 235 psig and a maximum pressure of 650 psig and a contained water volume of between 833 and 1556 cubic feet with a boron concentration of between 1900 and 2200 ppm of boron.

3/4.5.2 ECCS SUBSYSTEMS - OPERATING

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent Emergency Core Cooling System (ECCS) subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE high pressure safety injection pump,

b. One OPERABLE low pressure safety injection pump, and

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. The current value of the usage factor for each affected safety injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

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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 greater than or equal to 1750 psia.

2. A visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or corrosion.

3. Verifying that a minimum total of 173 cubic feet of solid granular trisodium phosphate dodecahydrate (TSP) is contained within the TSP storage baskets.

4. Verifying that when a representative sample of 70.5 $\pm$ 0.5 grams of TSP from a TSP storage basket is submerged, without agitation, in 10.0 $\pm$ 0.1 gallons of 120 $\pm$ 10°F borated water from the RWT, the pH of the mixed solution is raised to greater than or equal to 7 within 4 hours.

f. 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 SIAS and/or RAS test signals.

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

g. 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.

h. By verifying the correct position of each electrical and/or mechanical position stop for the following ECCS throttle valves:

1. During valve stroking operation or following maintenance on the valve and prior to declaring the valve OPERABLE when the ECCS subsystems are required to be OPERABLE.

EMERGENCY CORE COOLING SYSTEMS

3/4.5.4 REFUELING WATER TANK

LIMITING CONDITION FOR OPERATION

3.5.4 The refueling water tank shall be OPERABLE with:

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

b. A boron concentration of between 1900 and 2200 ppm of boron, and

c. A solution temperature of between 55°F and 100°F.

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 contained borated water volume in the tank, and

2. Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWT temperature when the outside air temperature is less then 55°F or greater than 100°F.

TABLE 3.7-2

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

3/4.8 ELECTRICAL POWER SYSTEMS

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 generators, each with:

1. Two separate engine-mounted fuel tanks containing a minimum volume of 200 gallons of fuel each,

2. A separate fuel storage system containing a minimum volume of 42,500 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 4.8.1.1.2a.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)

4. Simulating a loss-of-offsite power by itself, 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 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.

5. 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 ± 420 volts and 60 ± 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 ± 210 volts and 60 ± 0.6 Hz and shall be maintained throughout this test.

6. 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 load 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.

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

ELECTRICAL POWER SYSTEMS

A.C. SOURCES

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 with:

1. Two engine-mounted fuel tanks containing a minimum volume of 200 gallons of fuel,

2. A fuel storage system containing a minimum volume of 42,500 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, and within 8 hours, depressurize and vent the Reactor Coolant System through a greater than or equal to 3.58 square inch vent. 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 reactor vessel flange, 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).

3/4.9 REFUELING OPERATIONS

3/4.9.1 BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1 With the reactor vessel head closure bolts less than fully tensioned or with the head 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 greater than or equal to 40 gpm of a solution containing 1900 ppm boron or greater 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 within the reactor pressure vessel.

4.9.1.2 The boron concentration of the reactor coolant system and the refueling canal shall be determined by chemical analysis at least once per 72 hours.

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* The reactor shall be maintained in MODE 6 whenever fuel is in the reactor vessel with the reactor vessel head closure bolts less than fully tensioned or with the head removed.

3/4.10 SPECIAL TEST EXCEPTIONS

3/4.10.1 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, MTC, and SHUTDOWN MARGIN provided reactivity equivalent to at least the highest estimated CEA worth is available for trip insertion from OPERABLE CEA(s).

APPLICABILITY: MODES 2 and 3*.

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 greater than or equal to 40 gpm of a solution containing greater than or equal to 1900 ppm boron 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 greater than or equal to 40 gpm of a solution containing greater than or equal to 1900 ppm boron 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.

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* Operation in MODE 3 shall be limited to 6 consecutive hours.

RADIOACTIVE EFFLUENTS

GAS STORAGE TANKS

LIMITING CONDITION FOR OPERATION

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).

APPLICABILITY: At all times.

ACTION:

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.

SURVEILLANCE REQUIREMENTS

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.

than 8 days in gaseous effluents released from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50,

10) Limitations on the annual dose or dose commitment to any MEMBER OF THE PUBLIC, beyond the site boundary, due to releases of radioactivity and to radiation from uranium fuel cycle sources conforming to 40 CFR Part 190.

The provisions of Specifications 4.0.2 and 4.0.3 are applicable to the Radioactive Effluent Controls Program surveillance frequency.

g. Radiological Environmental Monitoring Program

A program shall be provided to monitor the radiation and radionuclides in the environs of the plant. The program shall provide (1) representative measurements of radioactivity in the highest potential exposure pathways, and (2) verification of the accuracy of the effluent monitoring program and modeling of the environmental exposure pathways. The program shall (1) be contained in the ODCM, (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 Bechtel Topical Report, BN-TOP-1 or ANS 56.8-1994 (as recommended by R.G. 1.163) which will be used for type A testing.

The peak calculated containment internal pressure for the design basis loss of coolant accident P_a, is 43.48 psig. The containment design pressure is 44 psig.

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

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	Movable Control Assemblies – CEA Position
Specification 3.1.3.6	Regulating CEA Insertion Limits
Specification 3.2.1	Linear Heat Rate
Specification 3.2.3	Total Integrated Radial Peaking Factors – 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 or any approved Revisions and Supplements thereto:

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. DELETED

4. DELETED

5. CENPD-275-P, Revision 1-P-A, "C-E Methodology for Core Designs Containing Gadolinia-Urania Burnable Absorbers," May 1988, & Revision 1-P Supplement 1-P-A, April 1999.

6. DELETED

CORE OPERATING LIMITS REPORT (COLR) (Continued)

b. (Continued)

7. DELETED

8. CEN-123(F)-P, "Statistical Combination of Uncertainties Methodology Part 1: CE Calculated Local Power Density and Thermal Margin/Low Pressure LSSS for St. Lucie Unit 1," December 1979.

9. DELETED

10. CEN-123(F)-P, "Statistical Combination of Uncertainties Methodology Part 3: CE Calculated Departure from Nucleate Boiling and Linear Heat Rate Limiting Conditions for Operation for St. Lucie Unit 1," February 1980.

11. CEN-191(B)-P, "CETOP-D Code Structure and Modeling Methods for Calvert Cliffs Units 1 and 2," December 1981.

12. Letter, J.W. Miller (NRC) to J.R. Williams, Jr. (FPL), Docket No. 50-389, Regarding Unit 2 Cycle 2 License Approval (Amendment No. 8 to NPF-16 and SER), November 9, 1984 (Approval of CEN-123(F)-P (three parts) and CEN-191(B)-P).

13. DELETED

14. Letter, J.A. Norris (NRC) to J.H. Goldberg (FPL), Docket No. 50-389, "St. Lucie Unit 2 – Change to Technical Specification Bases Sections '2.1.1 Reactor Core' and '3/4.2.5 DNB Parameters' (TAC No. M87722)," March 14, 1994 (Approval of CEN-371(F)-P).

15. DELETED

16. DELETED

17. DELETED

18. DELETED

CORE OPERATING LIMITS REPORT (COLR) (continued)

b. (continued)

34. Letter, J.A. Norris (NRC) to J.H. Goldberg (FPL), "St. Lucie Unit 2 – Issuance of Amendment Re: Moderator Temperature Coefficient (TAC No. M82517)," July 15, 1992.

35. Letter, J.W. Williams, Jr. (FPL) to D.G. Eisenhut (NRC), "St. Lucie Unit No. 2, Docket No. 50-389, Proposed License Amendment, Cycle 2 Reload," L-84-148, June 4, 1984.

36. Letter, J.R. Miller (NRC) to J.W. Williams, Jr. (FPL), Docket No. 50-389, Regarding Unit 2 Cycle 2 License Approval (Amendment No. 8 to NPF-16 and SER), November 9, 1984 (Approval to Methodology contained in L-84-148).

37. DELETED |

38. DELETED |

39. DELETED |

40. DELETED |

41. DELETED |

42. CEN-348(B)-P-A, Supplement 1-P-A, "Extended Statistical Combination of Uncertainties," January 1997.

43. CEN-372-P-A, "Fuel Rod Maximum Allowable Gas Pressure," May 1990.

44. DELETED |

45. DELETED |

CORE OPERATING LIMITS REPORT (COLR) (continued)

b. (continued)

46. DELETED

47. DELETED

48. CEN-396(L)-P, "Verification of the Acceptability of a 1-Pin Burnup Limit of 60 MWD/KG for St. Lucie Unit 2," November 1989 (NRC SER dated October 18, 1991, Letter J.A. Norris (NRC) to J.H. Goldberg (FPL), TAC No. 75947).

49. CENPD-269-P, Rev. 1-P, "Extended Burnup Operation of Combustion Engineering PWR Fuel," July 1984.

50. CEN-289(A)-P, "Revised Rod Bow Penalties for Arkansas Nuclear One Unit 2," December 1984 (NRC SER dated December 21, 1999, Letter K. N. Jabbour (NRC) to T.F. Plunkett (FPL), TAC No. MA4523).

51. CENPD-137, Supplement 2-P-A, "Calculative Methods for the ABB CE Small Break LOCA Evaluation Model," April 1998.

52. CENPD-140-A, "Description of the CONTRANS Digital Computer Code for Containment Pressure and Temperature Transient Analysis," June 1976.

53. DELETED

54. DELETED

55. CENPD-387-P-A, Revision 000, "ABB Critical Heat Flux Correlations for PWR Fuel," May 2000.

56. CENPD-132, Supplement 4-P-A, "Calculative Methods for the CE Nuclear Power Large Break LOCA Evaluation Model," March 2001.

57. CENPD-137, Supplement 2-P-A, "Calculative Methods for the ABB CE Small Break LOCA Evaluation Model," April 1998.

58. CENPD-404-P-A, Rev. 0, "Implementation of ZIRLO™ Cladding Material in CE Nuclear Power Fuel Assembly Designs," November 2001.

59. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985.

60. WCAP-10216-P-A, Revision 1A, "Relaxation of Constant Axial Offset Control; FQ Surveillance Technical Specification," February 1994.

CORE OPERATING LIMITS REPORT (COLR) (continued)

b. (continued)

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

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

63. WCAP-14565-P-A, Addendum 1, "Qualification of ABB Critical Heat Flux Correlations with VIPRE-01 Code," May 2003.

64. Letter, W. Jefferson Jr. (FPL) to Document Control Desk (USNRC), "St. Lucie Unit 2 Docket No. 50-389: Proposed License Amendment WCAP-9272 Reload Methodology and Implementing 30% Steam Generator Tube Plugging Limit," L-2003-276, December 2003 (NRC SER dated January 31, 2005, Letter B.T. Moroney (NRC) to J.A. Stall (FPL), TAC No. MC1566).

65. WCAP-14882-P-A, Rev. 0, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses", April 1999.

66. WCAP-7908-A, Rev. 0, "FACTRAN-A FORTRAN IV Code for Thermal Transients in a UO2 Fuel Rod", December 1989.

67. WCAP-7979-P-A, Rev. 0, "TWINKLE - A Multi-Dimensional Neutron Kinetics Computer Code", January 1975.

68. WCAP-7588, Rev. 1-A, "An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Special Kinetics Methods", January 1975.

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SHUTDOWN MARGIN, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any mid cycle revisions or supplements, shall be provided upon issuance for each reload cycle on the NRC.

SPECIAL REPORTS

STEAM GENERATOR TUBE INSPECTION REPORT

6.9.1.12 A report shall be submitted within 180 days after the initial entry into HOT SHUTDOWN following completion of an inspection of the replacement SGs performed in accordance with Specification 6.8.4.I.1. 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.

6.9.2 Special reports shall be submitted to the NRC within the time period specified for each report.

6.10 DELETED