

**ATTACHMENT 3
LICENSE AMENDMENT REQUEST
EXTENDED POWER UPRATE**

**RENEWED FACILITY OPERATING LICENSE
AND
TECHNICAL SPECIFICATIONS
MARKUPS AND CLEAN PAGES**

**FLORIDA POWER & LIGHT
ST. LUCIE UNIT 2**

This coversheet plus 132 pages

LIST OF AFFECTED PAGES

Renewed Facility Operating License

-3-

Technical Specifications

I

XXI

XXII

INSERT 1

XXIII

1-3

INSERT 2

1-4

1-5

2-3 (replaced by INSERT 3)

INSERT 3

2-4

2-5

2-6

3/4 1-1

3/4 1-3

3/4 1-8

3/4 1-13

3/4 1-14

3/4 1-15 (replaced by INSERT 4)

INSERT 4

3/4 1-24

3/4 2-14

3/4 2-15

3/4 4-8

3/4 4-25

INSERT 5

3/4 4-27

3/4 4-28

3/4 4-31a

3/4 4-31b

3/4 4-37a

3/4 5-1

3/4 5-3

3/4 5-5

3/4 5-8

3/4 7-3

3/4 8-1

Technical Specifications (continued)

3/4 8-6

INSERT 6

3/4 8-7

INSERT 7

3/4 8-9

3/4 9-1

3/4 9-12

3/4 10-1

3/4 11-15

5-4

INSERT 8

5-4a

5-4b

5-4C (DELETED)

5-4D (DELETED)

5-4E (DELETED)

5-4F (DELETED)

5-4G (DELETED)

5-4h

5-4i

5-4j

5-4k

5-4l

5-4m

5-4n

5-4o

6-15b

6-20

6-20a

6-20c

6-20d

6-20e

INSERT 9

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 ~~2700~~ megawatts (thermal).

3020

~~Commencing with the startup for Cycle 16 and until the Combustion Engineering Model 3410 Steam Generators are replaced, the maximum reactor core power shall not exceed 89 percent of 2700 megawatts (thermal) if:~~

- a. ~~The Reactor Coolant System Flow Rate is less than 335,000 gpm but greater than or equal to 300,000 gpm, or~~
- b. ~~The Reactor Coolant System Flow Rate is greater than or equal to 300,000 gpm AND the percentage of steam generator tubes plugged is greater than 30 percent (2520 tubes/SG) but less than or equal to 42 percent (3532 tubes/SG).~~

~~This restriction in maximum reactor core power is based on analyses provided by FPL in submittals dated October 21, 2005 and February 28, 2006, and approved by the NRC in Amendment No. 145, which limits the percent of steam generator tubes plugged to a maximum of 42 percent (3532 tubes) in either steam generator and limits the plugging asymmetry between steam generators to a maximum of 600 tubes.~~

B. Technical Specifications

XXX

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. ~~157~~ are hereby incorporated in the renewed license. FPL shall operate the facility in accordance with the Technical Specifications.

XXXXXXX XX, XXXX

INDEX

DEFINITIONS

<u>SECTION</u>	<u>PAGE</u>
1.0 <u>DEFINITIONS</u>	
1.1 ACTION.....	1-1
1.2 AXIAL SHAPE INDEX	1-1
1.3 AZIMUTHAL POWER TILT	1-1
1.4 CHANNEL CALIBRATION	1-1
1.5 CHANNEL CHECK.....	1-1
1.6 CHANNEL FUNCTIONAL TEST	1-2
1.7 CONTAINMENT VESSEL INTEGRITY.....	1-2
1.8 CONTROLLED LEAKAGE	1-2
1.9 CORE ALTERATION.....	1-2
1.9a CORE OPERATING LIMITS REPORT (COLR).....	1-2
1.10 DOSE EQUIVALENT I-131	1-3
1.11 DOSE EQUIVALENT AVERAGE DISINTEGRATION ENERGY	1-3
1.12 ENGINEERED SAFETY FEATURES RESPONSE TIME	1-3
1.13 FREQUENCY NOTATION	1-3
1.14 GASEOUS RADWASTE TREATMENT SYSTEM.....	1-3
1.15 IDENTIFIED LEAKAGE	1-3
1.16 LOW TEMPERATURE RCS OVERPRESSURE PROTECTION RANGE	1-4
1.17 MEMBER(S) OF THE PUBLIC.....	1-4
1.18 OFFSITE DOSE CALCULATION MANUAL (ODCM).....	1-4
1.19 OPERABLE – OPERABILITY	1-4
1.20 OPERATIONAL MODE – MODE.....	1-4
1.21 PHYSICS TESTS	1-4
1.22 PRESSURE BOUNDARY LEAKAGE	1-5
1.23 PROCESS CONTROL PROGRAM.....	1-5
1.24 PURGE – PURGING	1-5
1.25 RATED THERMAL POWER	1-5
1.26 REACTOR TRIP SYSTEM RESPONSE TIME.....	1-5
1.27 REPORTABLE EVENT	1-5
1.28 SHIELD BUILDING INTEGRITY.....	1-5
1.29 SHUTDOWN MARGIN	1-6
1.30 SITE BOUNDARY	1-6

INDEX

LIST OF FIGURES

<u>FIGURE</u>	<u>PAGE</u>
2.1-1 REACTOR CORE THERMAL MARGIN SAFETY LIMIT LINES FOUR REACTOR COOLANT PUMPS OPERATING	2-3
2.2-1 LOCAL POWER DENSITY – HIGH TRIP SETPOINT PART 1 (FRACTION OF RATED THERMAL POWER VERSUS QR_2)	2-7
2.2-2 LOCAL POWER DENSITY – HIGH TRIP SETPOINT PART 2 (QR_2 VERSUS Y_1).....	2-8
2.2-3 THERMAL MARGIN/LOW PRESSURE TRIP SETPOINT PART 1 (Y_1 , VERSUS A_1).....	2-9
2.2-4 THERMAL MARGIN/LOW PRESSURE TRIP SETPOINT PART 2 (FRACTION OF RATED THERMAL POWER VERSUS QR_1)	2-10
3.1-1 MINIMUM BORIC ACID STORAGE TANK VOLUME AS A FUNCTION OF STORED BORIC ACID CONCENTRATION.....	3/4 1-15
3.1-1a DELETED	
3.1-2 DELETED	
3.2-1 DELETED	
3.2-2 DELETED	
3.2-3 DELETED	
4.2-1 DELETED	
3.2-4 DELETED	
3.4-1 DOSE EQUIVALENT I-131 PRIMARY COOLANT SPECIFIC ACTIVITY LIMITS VERSUS PERCENT OF RATED THERMAL POWER WITH THE PRIMARY COOLANT SPECIFIC ACTIVITY > 1 μCi/GRAM DOSE EQUIVALENT I-131	3/4 4-28
3.4-2 ST. LUCIE UNIT 2 REACTOR COOLANT SYSTEM PRESSURE- TEMPERATURE LIMITS FOR 55 EFY, HEATUP, CORE CRITICAL, AND INSERVICE TEST	3/4 4-31a

INDEX

LIST OF FIGURES (continued)

<u>FIGURE</u>		<u>PAGE</u>
3.4-3	ST. LUCIE UNIT 2 REACTOR COOLANT SYSTEM PRESSURE-TEMPERATURE LIMITS FOR 55 EFY, COOLDOWN AND INSERVICE TEST	3/4 4-31b
3.4-4	DELETED	3/4 4-32
4.7-1	SAMPLING PLAN FOR SNUBBER FUNCTIONAL TEST	3/4 7-25
5.1-1	SITE AREA MAP	5-2
5.6-1a	REQUIRED FUEL ASSEMBLY BURNUP vs INITIAL ENRICHMENT and DECAY TIME, REGION II, 1.3 w/o	5-4B
5.6-1b	REQUIRED FUEL ASSEMBLY BURNUP vs INITIAL ENRICHMENT and DECAY TIME, REGION II, 1.5 w/o	5-4C
5.6-1c	REQUIRED FUEL ASSEMBLY BURNUP vs INITIAL ENRICHMENT and DECAY TIME, REGION I, 1.4 w/o	5-4D
5.6-1d	REQUIRED FUEL ASSEMBLY BURNUP vs INITIAL ENRICHMENT and DECAY TIME, REGION I, 1.82 w/o	5-4E
5.6-1e	REQUIRED FUEL ASSEMBLY BURNUP vs INITIAL ENRICHMENT, REGION I, 2.82 w/o	5-4F
5.6-1f	REQUIRED FUEL ASSEMBLY BURNUP vs INITIAL ENRICHMENT, REGION II CASK PIT STORAGE RACK	5-4G
6.2-1	DELETED	6-3
6.2-2	DELETED	6-4

47

DELETED

INSERT 1



5.6-1	ALLOWABLE REGION 1 STORAGE PATTERNS AND FUEL ARRANGEMENTS.....	5-4h
5.6-2	ALLOWABLE REGION 2 STORAGE PATTERNS AND FUEL ARRANGEMENTS (Sheet 1 of 3)	5-4i
5.6-2	ALLOWABLE REGION 2 STORAGE PATTERNS AND FUEL ARRANGEMENTS (Sheet 2 of 3)	5-4j
5.6-2	ALLOWABLE REGION 2 STORAGE PATTERNS AND FUEL ARRANGEMENTS (Sheet 3 of 3)	5-4k
5.6-3	INTERFACE REQUIREMENTS BETWEEN REGION 1 AND REGION 2 (Sheet 1 of 2).....	5-4l
5.6-3	INTERFACE REQUIREMENTS BETWEEN REGION 1 AND REGION 2 (Sheet 2 of 2).....	5-4m
5.6-4	ALLOWABLE CASK PIT STORAGE RACK PATTERNS.....	5-4n

INDEX

LIST OF TABLES

<u>TABLE</u>	<u>PAGE</u>
1.1 FREQUENCY NOTATION	1-8
1.2 OPERATIONAL MODES	1-9
2.2-1 REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS.....	2-4
3.1-1 DELETED	
3.2-1 DELETED	3/4 2-11
3.2-2 DNB MARGIN LIMITS	3/4 2-15
3.3-1 REACTOR PROTECTIVE INSTRUMENTATION	3/4 3-2
3.3-2 DELETED	
4.3-1 REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS	3/4 3-8
3.3-3 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION	3/4 3-12
3.3-4 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES	3/4 3-17
3.3-5 DELETED	
4.3-2 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS	3/4 3-22
3.3-6 RADIATION MONITORING INSTRUMENTATION	3/4 3-25
4.3-3 RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS	3/4 3-28
3.3-8 DELETED	
4.3-5 DELETED	

5.6-1 MINIMUM BURNUP COEFFICIENTS.....	5-4o
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DEFINITIONS

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."

~~E~~ AVERAGE DISINTEGRATION ENERGY

- 1.11 ~~E shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.~~

INSERT 2

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:
- 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
 - 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
 - Reactor Coolant System leakage through a steam generator to the secondary system (primary-to-secondary leakage).

INSERT 2

DOSE EQUIVALENT XE-133

- 1.11 DOSE EQUIVALENT XE-133 shall be that concentration of Xe-133 ($\mu\text{Ci}/\text{gram}$) 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."

DEFINITIONS

LOW TEMPERATURE OVERPRESSURE PROTECTION RANGE – RCS

Deleted

1.16 ~~The LOW TEMPERATURE OVERPRESSURE PROTECTION RANGE is that operating condition when (1) the RCS cold leg temperature is less than or equal to that specified in Table 3.4-3, and (2) the Reactor Coolant System is not vented to containment by an opening of at least 3.58 square inches.~~

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.

DEFINITIONS

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 ~~2700~~ MWt.

REACTOR TRIP SYSTEM RESPONSE TIME

3020

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

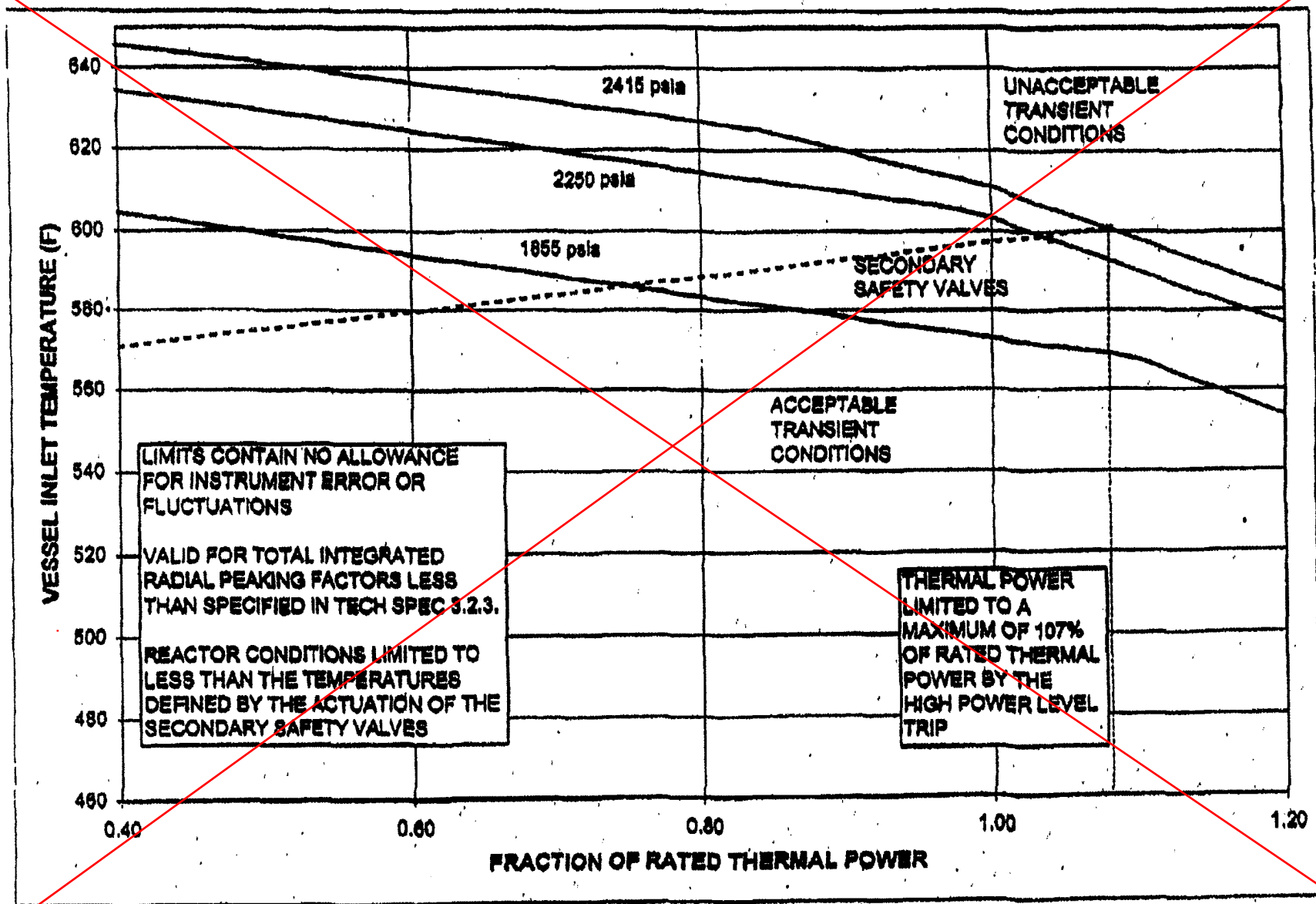
1.27 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

SHIELD BUILDING INTEGRITY

1.28 SHIELD BUILDING INTEGRITY shall exist when:

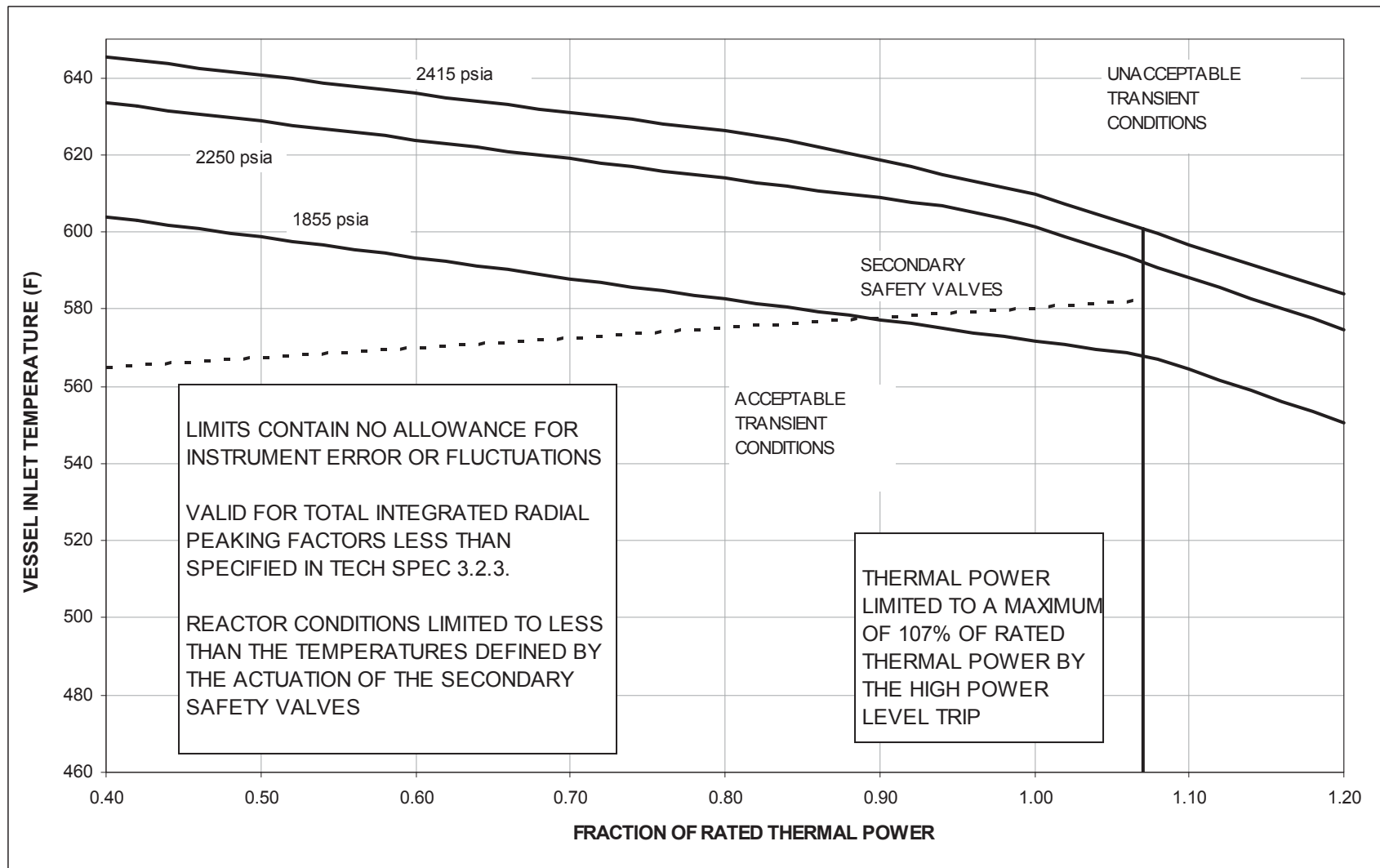
- a. Each door is closed except when the access opening is being used for normal transit entry and exit;
- b. The shield building ventilation system is in compliance with Specification 3.6.6.1, and
- c. The sealing mechanism associated with each penetration (e.g., welds, bellows or O-rings) is OPERABLE.

See INSERT 3 - New Figure 2.1-1



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

INSERT 3



**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

<u>FUNCTIONAL UNIT</u>	<u>TRIP SETPOINT</u>	<u>ALLOWABLE VALUES</u>
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 ⁽¹⁾ Difference – High (Logic in TM/LP Trip Unit)	≤ 120.0 psid	≤ 132.0 psid
8. Steam Generator Level – Low	≥ 20.5% (3)	≥ 19.5% (3)

psia (2)

Low (6), (7)

35.0% (3)

34.1% (3)

TABLE 2.2-1 (Continued)
REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

<u>FUNCTIONAL UNIT</u>	<u>TRIP SETPOINT</u>	<u>ALLOWABLE VALUES</u>
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 design Reactor Coolant flow with four pumps operating*	≥ 94.9% of design Reactor Coolant flow with four pumps operating*
15. Loss of Load (Turbine) Hydraulic Fluid Pressure – Low ⁽⁵⁾	≥ 800 psig	≥ 800 psig

minimum

For minimum

, refer to

* ~~Design~~ reactor coolant flow with four pumps operating is the minimum RCS flow specified in the COLR Table 3.2-2.

** 10-minute time delay after relay actuation.

TABLE 2.2-1 (Continued)
REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

TABLE NOTATION

- (1) Trip may be manually bypassed below 0.5% of RATED THERMAL POWER during testing pursuant to Special Test Exception 3.10.3; bypass shall be automatically removed when Wide Range Logarithmic Neutron Flux power is greater than or equal to 0.5% of RATED THERMAL POWER.
- (2) Trip may be manually bypassed below 705 psig; bypass shall be automatically removed at or above 705 psig.
- (3) % of the narrow range steam generator level indication.
- (4) Trip may be bypassed below $10^{-4}\%$ and above 15% of RATED THERMAL POWER; bypass shall be automatically removed when Wide Range Logarithmic Neutron Flux power is $\geq 10^{-4}\%$ and Power Range Neutron Flux power $\leq 15\%$ of RATED THERMAL POWER.
- (5) Trip may be bypassed below 15% of RATED THERMAL POWER; bypass shall be automatically removed when Power Range Neutron Flux power is greater than or equal to 15% of RATED THERMAL POWER.

- (6) 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.
- (7) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance of the Trip Setpoint or a value that is more conservative than the Trip Setpoint, otherwise that channel shall not be returned to OPERABLE status. The Trip Setpoint and the methodology used to determine the Trip Setpoint, the as-found acceptance criteria band, and the as-left acceptance criteria are specified in the UFSAR.

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 ~~1720~~ 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.

* See Special Test Exception 3.10.1.

REACTIVITY CONTROL SYSTEMS

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

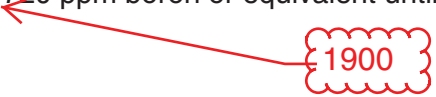
LIMITING CONDITION FOR OPERATION

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 ~~1720~~ ppm boron or equivalent until the required SHUTDOWN MARGIN is restored. |



1900

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.

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:

-  a. 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.
-  b. 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.
-  c. 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 2.5 to 3.5 weight percent boric acid (~~4371~~ to 6119 ppm boron). 3.1
- b. The refueling water tank with:
 - 1. A minimum contained borated water volume of 125,000 gallons, 5420
 - 2. A minimum boron concentration of ~~4720~~ ppm, and 1900
 - 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 ~~1720 and 2100~~ ppm of boron, and
 3. A solution temperature of between 55°F and 100°F.

1900 and 2200

APPLICABILITY: MODES 1, 2, 3 and 4.

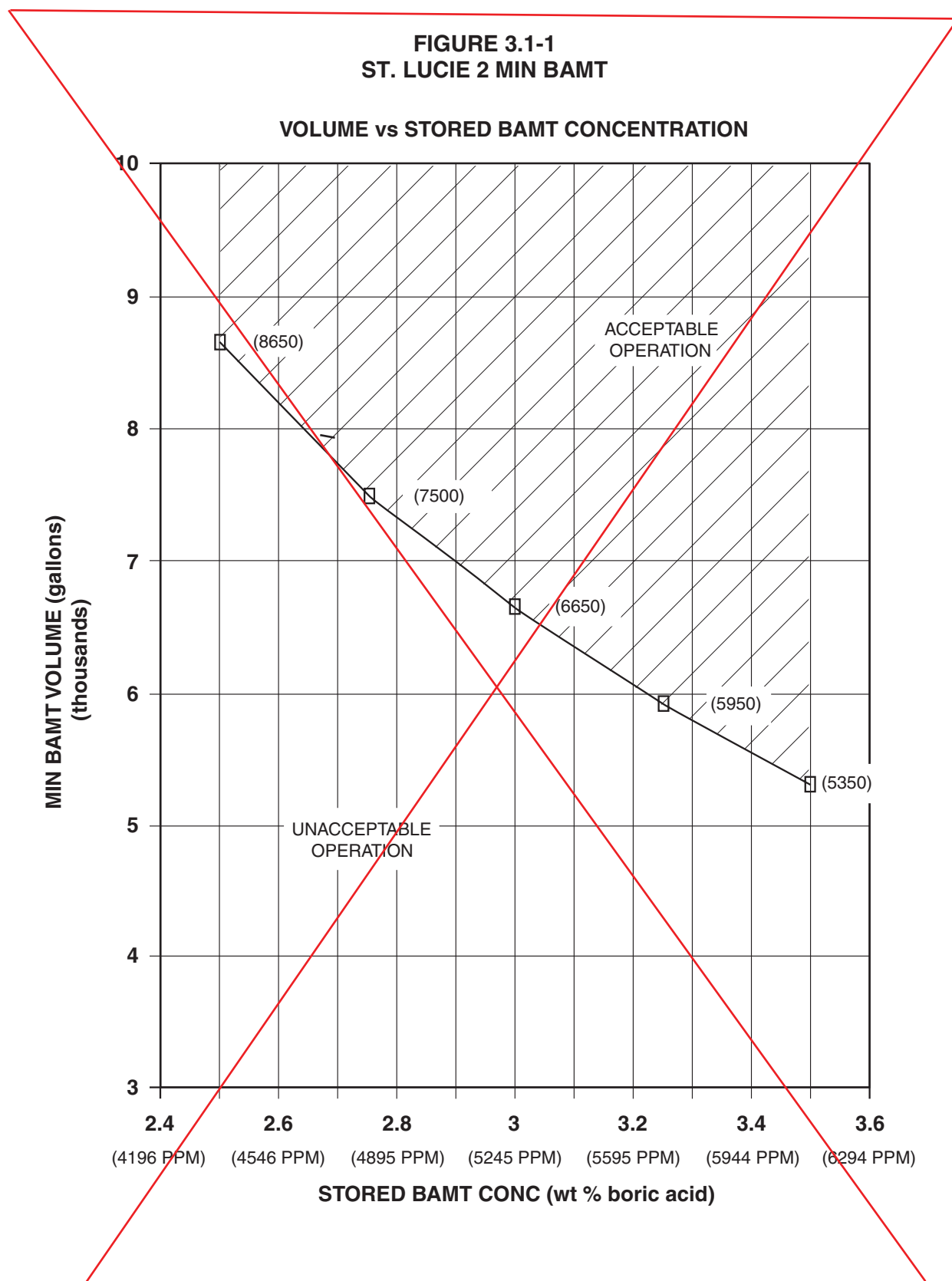
ACTION:

- 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.

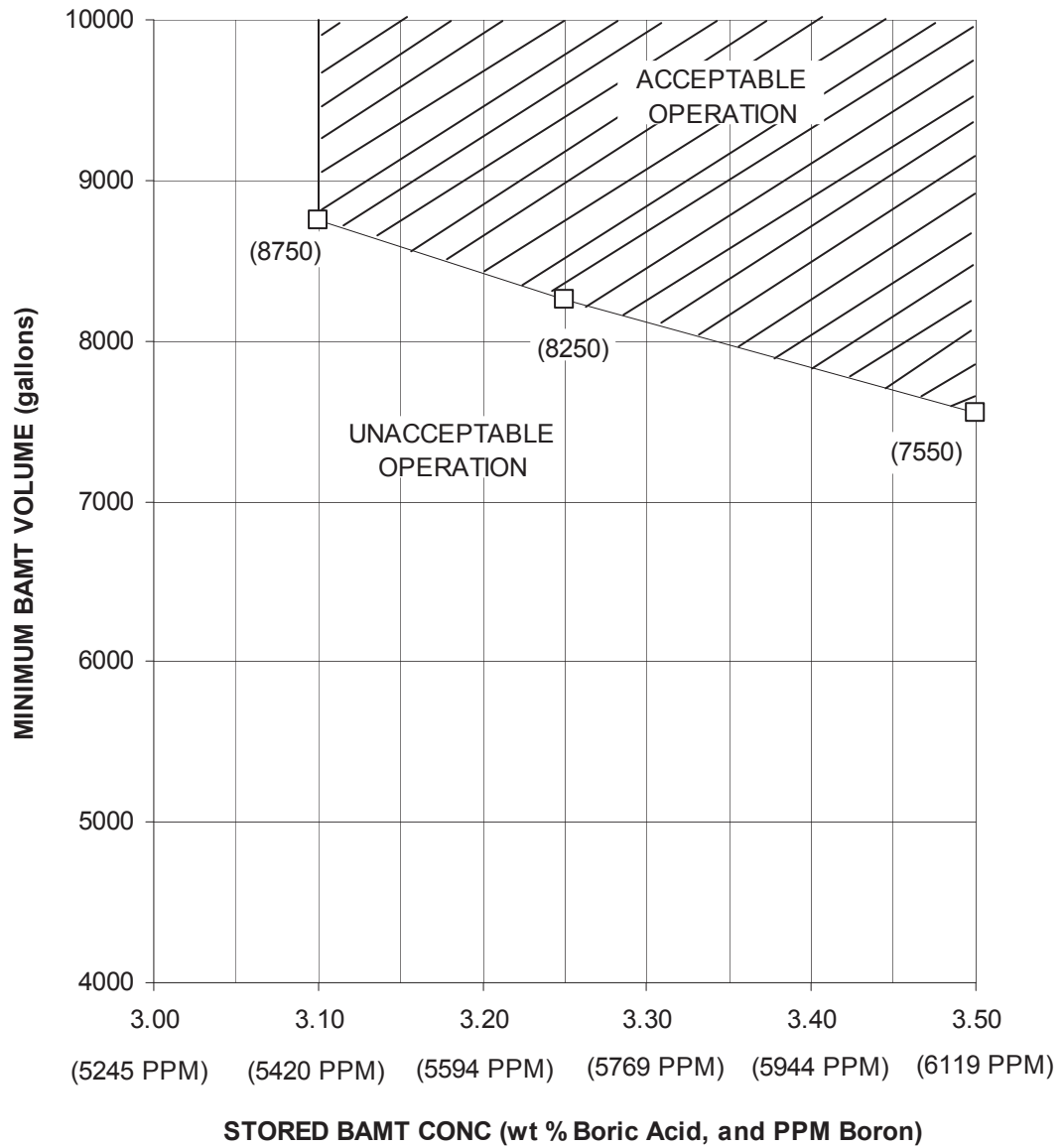
SURVEILLANCE REQUIREMENTS

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

3.25

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 ~~3.2~~ 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 ~~individuals~~ 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.

individual

POWER DISTRIBUTION LIMITS

DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB-related parameters shall be maintained within the limits shown on Table 3.2-2:

- a. Cold Leg Temperature
- b. Pressurizer Pressure
- c. Reactor Coolant System Total Flow Rate
- d. AXIAL SHAPE INDEX

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.

DNB-related

SURVEILLANCE REQUIREMENTS

4.2.5.1 Each of the parameters of Table 3.2-2 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 $\geq 80\%$ of RATED THERMAL POWER.

90%

DELETED

TABLE 3.2-2

DNB MARGIN

LIMITS

PARAMETER

**FOUR REACTOR
COOLANT PUMPS
OPERATING**

Cold Leg Temperature (Narrow Range)

Within the limits specified in the
COLR Table 3.2-2

Pressurizer Pressure*

Within the limits specified in the
COLR Table 3.2-2

Reactor Coolant Flow Rate**

$\geq 335,000$ gpm and $\geq$ the limit specified
in the COLR Table 3.2-2

AXIAL SHAPE INDEX

COLR Figure 3.2-4

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

** Commencing with the startup for Cycle 16 and until the Combustion Engineering Model 3410 Steam Generators are replaced, Reactor Coolant Flow Rate will also be limited in accordance with Renewed Operating License Paragraph 3.A.

REACTOR COOLANT SYSTEM

OPERATING

2410.3

2560.3

LIMITING CONDITION FOR OPERATION

3.4.2.2 All pressurizer code safety valves shall be OPERABLE with a lift setting of ≥ 2435.3 psig and ≤ 2535.3 psig.*

APPLICABILITY: MODES 1, 2, 3, and 4 with all RCS cold leg temperatures $> 230^{\circ}\text{F}$.

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 $\leq 230^{\circ}\text{F}$ 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 $\pm 1\%$ of 2500 psia.

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

REACTOR COOLANT SYSTEM

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 $\frac{100}{E}$ microcuries/gram.

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

ACTION:

MODES 1, 2 and 3*:

- a. ~~With the specific activity of the primary coolant greater than 1.0 microcurie/gram DOSE EQUIVALENT I 131 for more than 48 hours during one continuous time interval or exceeding the limit line shown on Figure 3.4-1, be in at least HOT STANDBY with T_{avg} less than 500°F within 6 hours.~~
- b. ~~With the specific activity of the primary coolant greater than $\frac{100}{E}$ microcuries/gram, be in at least HOT STANDBY with T_{avg} less than 500°F within 6 hours.~~

MODES 1, 2, 3, 4 and 5:

~~With the specific activity of the primary coolant greater than 1 microcurie/gram DOSE EQUIVALENT I 131 or greater than $\frac{100}{E}$ microcuries/gram, perform the sampling and analysis requirements of item 4 a) of Table 4.4-4 until the specific activity of the primary coolant is restored to within its limits.~~

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.

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

INSERT 5

- a. With the specific activity of the primary coolant $> 1.0 \mu\text{Ci/gram DOSE EQUIVALENT I-131}$, verify DOSE EQUIVALENT I-131 is $\leq 60.0 \mu\text{Ci/gram}$ once per four hours.
- b. With the specific activity of the primary coolant $> 1.0 \mu\text{Ci/gram DOSE EQUIVALENT I-131}$, but $\leq 60.0 \mu\text{Ci/gram DOSE EQUIVALENT I-131}$, operation may continue for up to 48 hours while efforts are made to restore DOSE EQUIVALENT I-131 to within the $1.0 \mu\text{Ci/gram}$ limit. Specification 3.0.4 is not applicable.
- c. With the specific activity of the primary coolant $> 1.0 \mu\text{Ci/gram DOSE EQUIVALENT I-131}$ for greater than 48 hours during one continuous time interval, or $> 60.0 \mu\text{Ci/gram DOSE EQUIVALENT I-131}$, be in HOT STANDBY within 6 hours and COLD SHUTDOWN within the following 30 hours.
- d. With the specific activity of the primary coolant $> 518.9 \mu\text{Ci/gram DOSE EQUIVALENT XE-133}$, operation may continue for up to 48 hours while efforts are made to restore DOSE EQUIVALENT XE-133 to within the $518.9 \mu\text{Ci/gram DOSE EQUIVALENT XE-133}$ limit. Specification 3.0.4 is not applicable.
- e. With the specific activity of the primary coolant $> 518.9 \mu\text{Ci/gram DOSE EQUIVALENT XE-133}$ 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.

TABLE 4.4-4

**PRIMARY COOLANT SPECIFIC ACTIVITY SAMPLE
AND ANALYSIS PROGRAM**

DOSE EQUIVALENT XE-133	TYPE OF MEASUREMENT AND ANALYSIS	MINIMUM	SAMPLE AND ANALYSIS FREQUENCY	MODES IN WHICH SAMPLE AND ANALYSIS REQUIRED
	1. Gross Activity Determination		At least once per 72 hours	1, 2, 3, and 4
	2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration		1 per 14 days	1
	3. Radiochemical for E Determination		1 per 6 months*	4
	4. Isotopic Analysis for Iodine Including I-131, I-133, and I-135		a) Once per 4 hours, whenever the specific activity exceeds 1 micro-Ci/gram, DOSE EQUIVALENT I-131 or 100/E micro-Ci/gram, and	1#, 2#, 3#, 4#, 5#
3.				and
I-132,				
I-134,				
			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.

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

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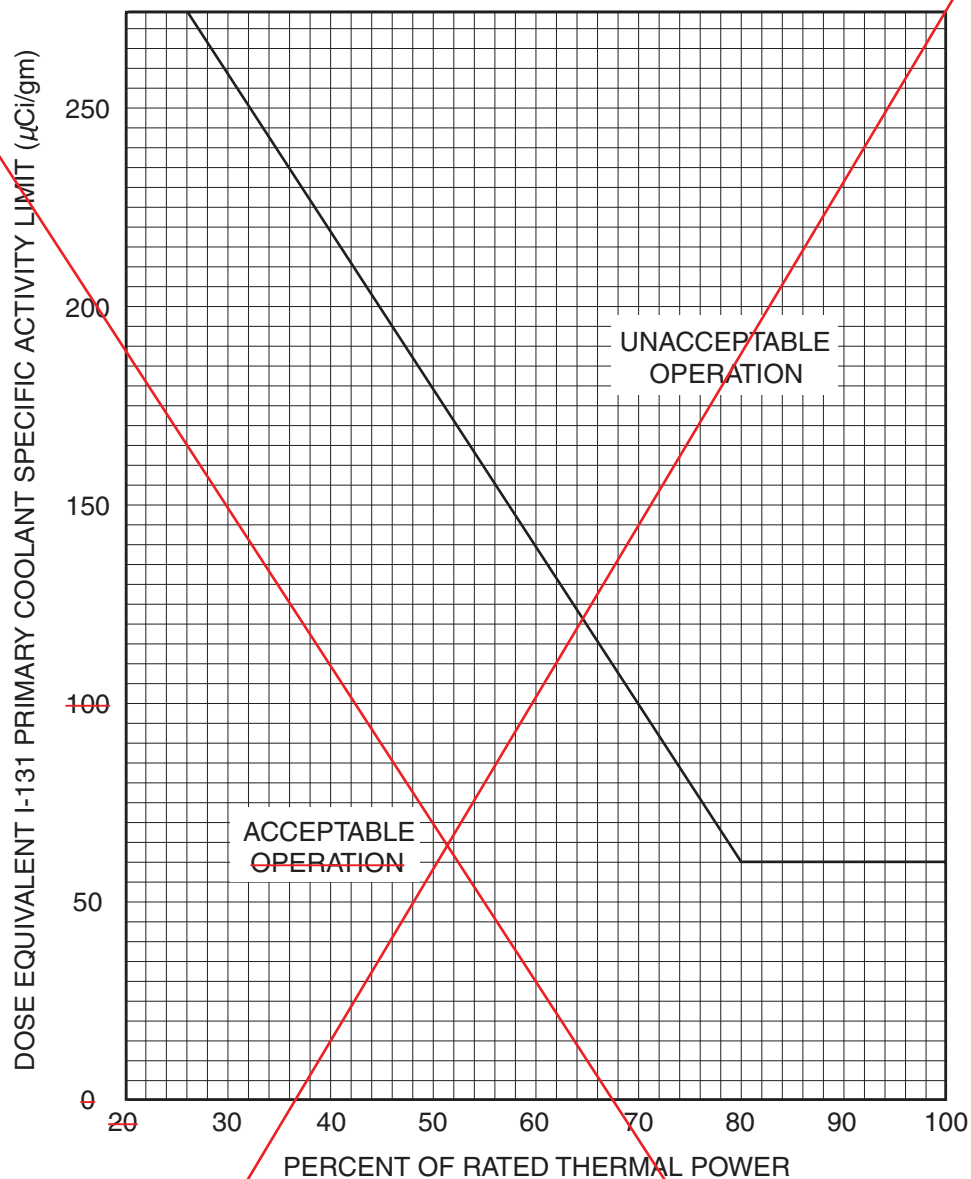
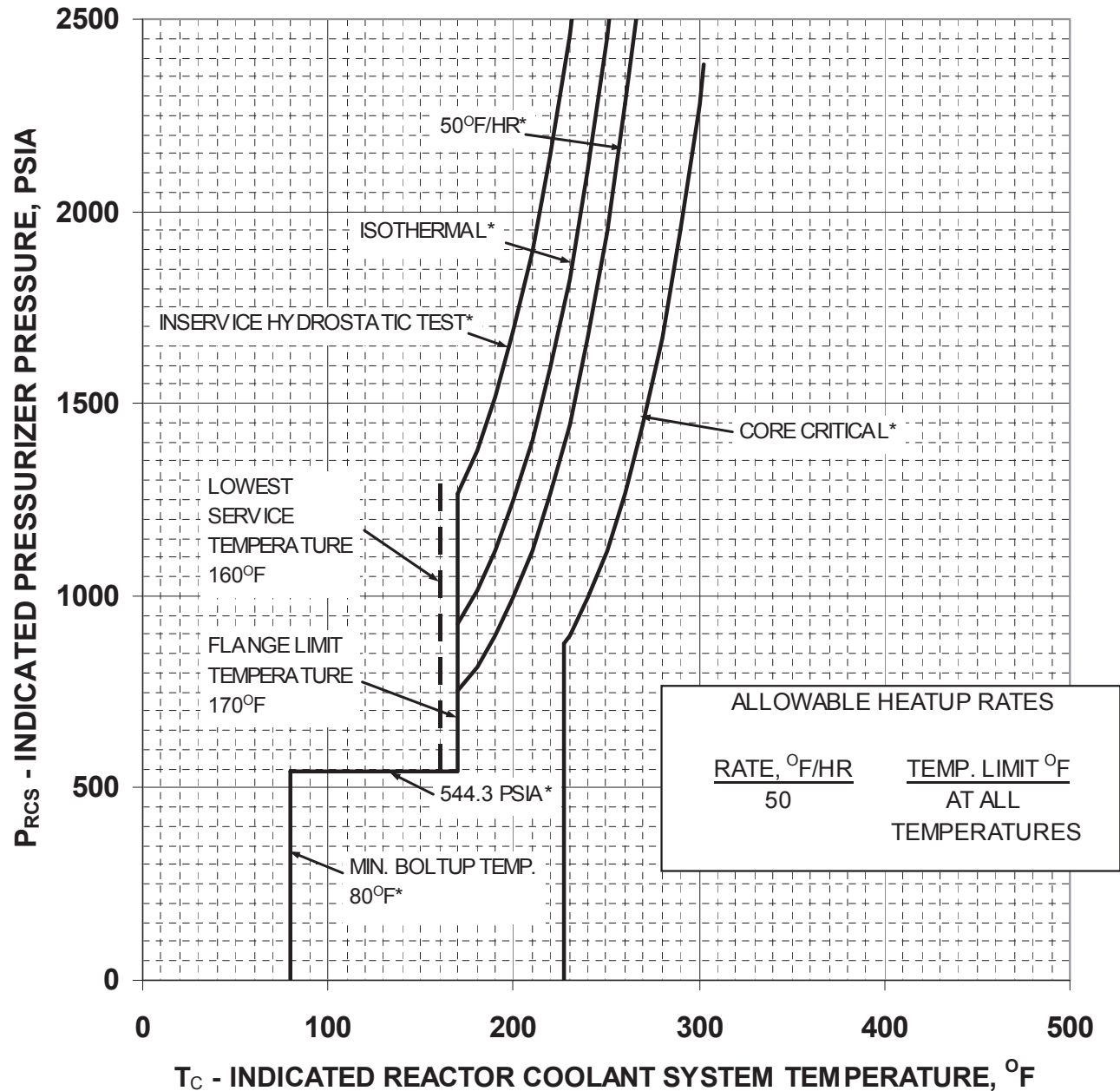


FIGURE 3.4-1

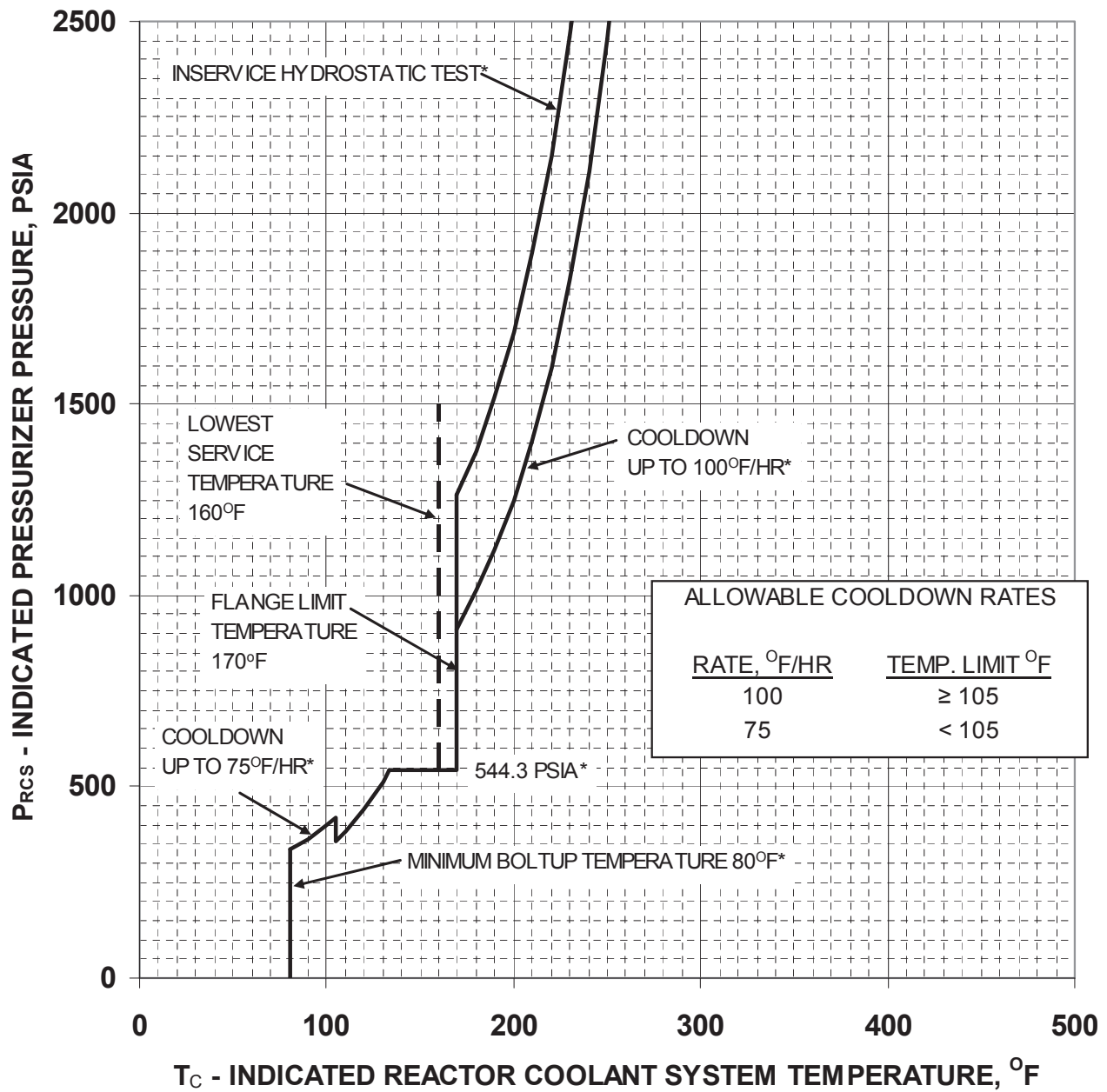
DOSE EQUIVALENT I-131 Primary Coolant Specific Activity Limit Versus Percent of RATED THERMAL POWER with the Primary Coolant Specific Activity > 1.0 μCi/gram Dose Equivalent I-131

**FIGURE 3.4-2
ST. LUCIE UNIT 2 REACTOR COOLANT SYSTEM PRESSURE-
TEMPERATURE LIMITS FOR 55 EFY, HEATUP, CORE CRITICAL, AND
INSERVICE TEST**



* - Includes Instrument Uncertainty

**FIGURE 3.4-3
ST. LUCIE UNIT 2 REACTOR COOLANT SYSTEM PRESSURE-
TEMPERATURE LIMITS FOR 55-EFPY, COOLDOWN AND INSERVICE
TEST**



* - Includes Instrument Uncertainty

TABLE 3.4-3

LOW TEMPERATURE RCS OVERPRESSURE PROTECTION RANGE

<u>Operating Period, EFY</u>	<u>Cold Leg Temperature, F°</u>	
	<u>During Heatup</u>	<u>During Cooldown</u>
≤ 55	≤ 246	≤ 224

°F

TABLE 3.4-4

MINIMUM COLD LEG TEMPERATURE FOR PORV USE FOR LTOP

<u>Operating Period EFY</u>	<u>Cold Leg Temperature °F</u>	
	<u>T_{cold}, F° During Heatup</u>	<u>T_{cold}, F° During Cooldown</u>
≤ 55	80	132

47

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 ~~1720 and 2100~~ 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

* 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 ~~1720 and 2100~~ 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 ~~1720 and 2100~~ ppm of boron.

EMERGENCY CORE COOLING SYSTEMS

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.

With pressurizer pressure greater than or equal to 1750 psia.

* 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.

EMERGENCY CORE COOLING SYSTEMS

SURVEILLANCE REQUIREMENTS (continued)

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 ± 0.5 grams of TSP from a TSP storage basket is submerged, without agitation, in 10.0 ± 0.1 gallons of $120 \pm 10^\circ\text{F}$ 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 ~~path~~ actuates to its correct position on SIAS and/or RAS test signals. paths
 2. Verifying that each of the following pumps start automatically upon receipt of a Safety Injection Actuation Test Signal:
 - a. High-Pressure Safety Injection ~~pump~~. pumps
 - b. Low-Pressure Safety Injection ~~pump~~. 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:~~ Delete underline " _ "
 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 minimum contained borated water volume 477,360 gallons,
 - A boron concentration of between ~~1720 and 2100~~ ppm of boron, and
 - A solution temperature of between 55°F and 100°F.

1900 and 2200

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:
- At least once per 7 days by:
 - Verifying the contained borated water volume in the tank, and
 - Verifying the boron concentration of the water.
 - At least once per 24 hours by verifying the RWT temperature when the outside air temperature is less than 55°F or greater than 100°F.

TABLE 3.7-2
STEAM LINE SAFETY VALVES PER LOOP

<u>VALVE NUMBER</u>			<u>LIFT SETTING (+ 1% to - 3%)</u>
	<u>Header A</u>	<u>Header B</u>	
a.	8201	8205	≥ 955.3 psig and ≤ 995.3 psig
b.	8202	8206	≥ 955.3 psig and ≤ 995.3 psig
c.	8203	8207	≥ 955.3 psig and ≤ 995.3 psig
d.	8204	8208	≥ 955.3 psig and ≤ 995.3 psig
e.	8209	8213	≥ 994.1 psig and ≤ 1035.7 psig
f.	8210	8214	≥ 994.1 psig and ≤ 1035.7 psig
g.	8211	8215	≥ 994.1 psig and ≤ 1035.7 psig
h.	8212	8216	≥ 994.1 psig and ≤ 1035.7 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 ~~40,000~~ gallons of fuel, and
 3. A separate fuel transfer pump.

42,500

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.

* 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.

ELECTRICAL POWER SYSTEMS

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 ± 420 volts and 60 ± 1.2 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 operates on standby for greater than or equal to 5 minutes. The steady state generator voltage and frequency shall be 4160 ± 420 volts and 60 ± 1.2 Hz within 10 seconds after the auto-start signal; the generator voltage and frequency shall be maintained within these limits during 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 ± 420 volts and 60 ± 1.2 Hz during this test.

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



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.

ELECTRICAL POWER SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

- 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 actuation signal.
7. ~~Verifying the diesel generator operates for at least 24 hours. **** During the first 2 hours of this test, the diesel generator shall be loaded within a load band of 3800 to 3985 kW[#] and during the remaining 22 hours of this test, the diesel generator shall be loaded within a load band of 3450 to 3685 kW[#]. The generator voltage and frequency shall be 4160 $\pm$ 420 volts and 60 $\pm$ 1.2 Hz within 10 seconds after the start signal; the steady-state generator voltage and frequency shall be maintained within these limits during this test.~~
8. Verifying that the auto-connected loads to each diesel generator do not exceed the 2000-hour rating of 3935 kW.
9. Verifying the diesel generator's capability to:
- a) Synchronize with the offsite power source while the generator is loaded with its emergency loads upon a simulated restoration of offsite power.
 - b) Transfer its load to the offsite power source, and
 - c) Be restored to its standby status.
10. Verifying that with the diesel generator operating in a test mode (connected to its bus), a simulated safety injection signal overrides the test mode by (1) returning the diesel generator to standby operation and (2) automatically energizes the emergency loads with offsite power.
11. Verifying that the fuel transfer pump transfers fuel from each fuel storage tank to the engine-mounted tanks of each diesel via the installed cross connection lines.

INSERT 7

This band is meant as guidance to avoid routine overloading of the engine. Variations in load in excess of this band due to changing bus loads shall not invalidate this test.

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



7. Verifying that the diesel operates for at least 24 hours****.
- a) Within 10 seconds, generator voltage and frequency shall be 4160 ± 420 volts and 60 ± 1.2 Hz.
 - b) Steady-state generator voltage and frequency shall be 4160 ± 210 volts and 60 ± 0.6 Hz and shall be maintained throughout this test.
 - c) During the first 2 hours of this test, the diesel generator shall be loaded within a load band of 3800 to 3985 kW[#], and
 - d) During the remaining 22 hours of this test, the diesel generator shall be loaded within a load band of 3450 to 3685 kW[#].

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 ~~40,000~~ gallons of fuel, and
 3. A fuel transfer pump.

42,500

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 1720 ppm boron or greater to restore boron concentration to within limits.

1900

SURVEILLANCE REQUIREMENTS

- 4.9.1.1 The boron concentration limit shall be determined prior to:
- Removing or unbolting the reactor vessel head, and
 - 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.

* 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.

REFUELING OPERATIONS

3/4.9.11 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 ~~1720~~ 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 fuel storage pool boron concentration to within the required limit.
- c. The provisions of Specification 3.0.3 are not applicable.

1900

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.

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 ~~1720~~ 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 ~~1720~~ ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

1900

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.

* 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 ~~285,000~~ curies noble gases (considered as Xe-133).

202,500

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 ~~100~~ $\frac{\mu\text{Ci}}{\text{g}}$.

518.9 $\mu\text{Ci}/\text{gram}$ DOSE EQUIVALENT XE-133.

DESIGN FEATURES

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1-1.

5.6 FUEL STORAGE

CRITICALITY

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

1. A k_{eff} equivalent to less than 1.0 when flooded with unborated water, including a conservative allowance for biases and uncertainties as described in Section 9.1 of the Updated Final Safety Analysis Report.

500

2. A k_{eff} equivalent to less than or equal to 0.95 when flooded with water containing 520 ppm boron, including a conservative allowance for biases and uncertainties as described in Section 9.1 of the Updated Final Safety Analysis Report.

8.965

3. A nominal 8.96 inch center-to-center distance between fuel assemblies placed in the spent fuel pool storage racks and a nominal 8.80 inch center-to-center distance between fuel assemblies placed in the cask pit storage rack.

INSERT 8

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

b. ~~Fuel placed in Region I of the spent fuel storage racks shall be stored in a configuration that will assure compliance with 5.6.1 a.1 and 5.6.1 a.2, above, with the following considerations:~~

1. ~~Fresh fuel shall have a nominal average U-235 enrichment of less than or equal to 4.5 weight percent.~~

2. ~~The reactivity effect of CEAs placed in fuel assemblies may be considered.~~

3. ~~The reactivity equivalencing effects of burnable absorbers may be considered.~~

4. ~~The reactivity effects of fuel assembly burnup and decay time may be considered as specified in Figures 5.6-1c through 5.6-1e.~~

c. ~~Fuel placed in Region II of the spent fuel storage racks shall be placed in a configuration that will assure compliance with 5.6.1 a.1 and 5.6.1 a.2, above, with the following considerations:~~

1. ~~Fuel placed in the Region II spent fuel pool storage racks shall meet the burnup and decay time requirements specified in Figure 5.6-1a or 5.6-1b. Fuel placed in the Region II cask pit storage rack shall meet the burnup requirements specified in Figure 5.6-1f.~~

2. ~~The reactivity effect of CEAs placed in fuel assemblies may be considered.~~

3. ~~The reactivity equivalencing effects of burnable absorbers may be considered.~~

4. For storage of enriched fuel assemblies, requirements of Specification 5.6.1.a.1 and 5.6.1.a.2 shall be met by positioning fuel in the spent fuel pool storage racks consistent with the requirements of Specification 5.6.1.c or in configurations that have been shown to comply with Specifications 5.6.1.a.1 and 5.6.1.a.2 using the methodology as described in Section 9.1 of the Updated Final Safety Analysis Report.
 5. Fissile material, not contained in a fuel assembly lattice, shall be stored in accordance with the requirements of Specifications 5.6.1.a.1 and 5.6.1.a.2.
- b. The 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 pool storage racks shall be controlled as described below.
1. The maximum initial planar average U-235 enrichment of any fuel assembly inserted in a spent fuel pool 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 pattern definitions of Figure 5.6-1 and the minimum burnup requirements as defined in Table 5.6-1. (See Specification 5.6.1.c.7 for exceptions)
 3. Fuel placed in Region 2 of the spent fuel pool storage racks shall comply with the storage pattern definitions or allowed special arrangement definitions of Figure 5.6-2 and the minimum burnup requirements as defined in Table 5.6-1. (See Specification 5.6.1.c.7 for exceptions)
 4. The 2x2 array of fuel assemblies that span the interface between Region 1 and Region 2 of the spent fuel pool storage racks shall comply with the storage pattern definitions of Figure 5.6-3 and the minimum burnup requirements as defined in Table 5.6-1. The allowed special arrangements in Region 2 as shown in Figure 5.6-2 shall not be placed adjacent to Region 1. (See Specification 5.6.1.c.7 for exceptions)
 5. Fuel placed in the cask pit storage rack shall comply with the storage pattern definitions of Figure 5.6-4 and the minimum burnup requirements as defined in Table 5.6-1. (See Specification 5.6.1.c.7 for exceptions)
 6. The same directional orientation for Metamic inserts is required for contiguous groups of 2x2 arrays where Metamic inserts are required.
 7. Fresh or spent fuel in any allowed configuration may be replaced with non-fuel hardware, and fresh fuel in any allowed configuration may be replaced with a fuel rod storage basket containing fuel rod(s). Also, storage of Metamic inserts or control rods, without any fissile material, is acceptable in locations designated as completely water-filled cells.

DESIGN FEATURES (continued)

CRITICALITY (continued)

maximum planar average

4.6

- 5.6.1 d. The new fuel storage racks are designed for dry storage of unirradiated fuel assemblies having a U-235 enrichment less than or equal to 4.5 weight percent, while maintaining a k_{eff} of less than or equal to 0.98 under the most reactive condition.

DRAINAGE

- 5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 56 feet.

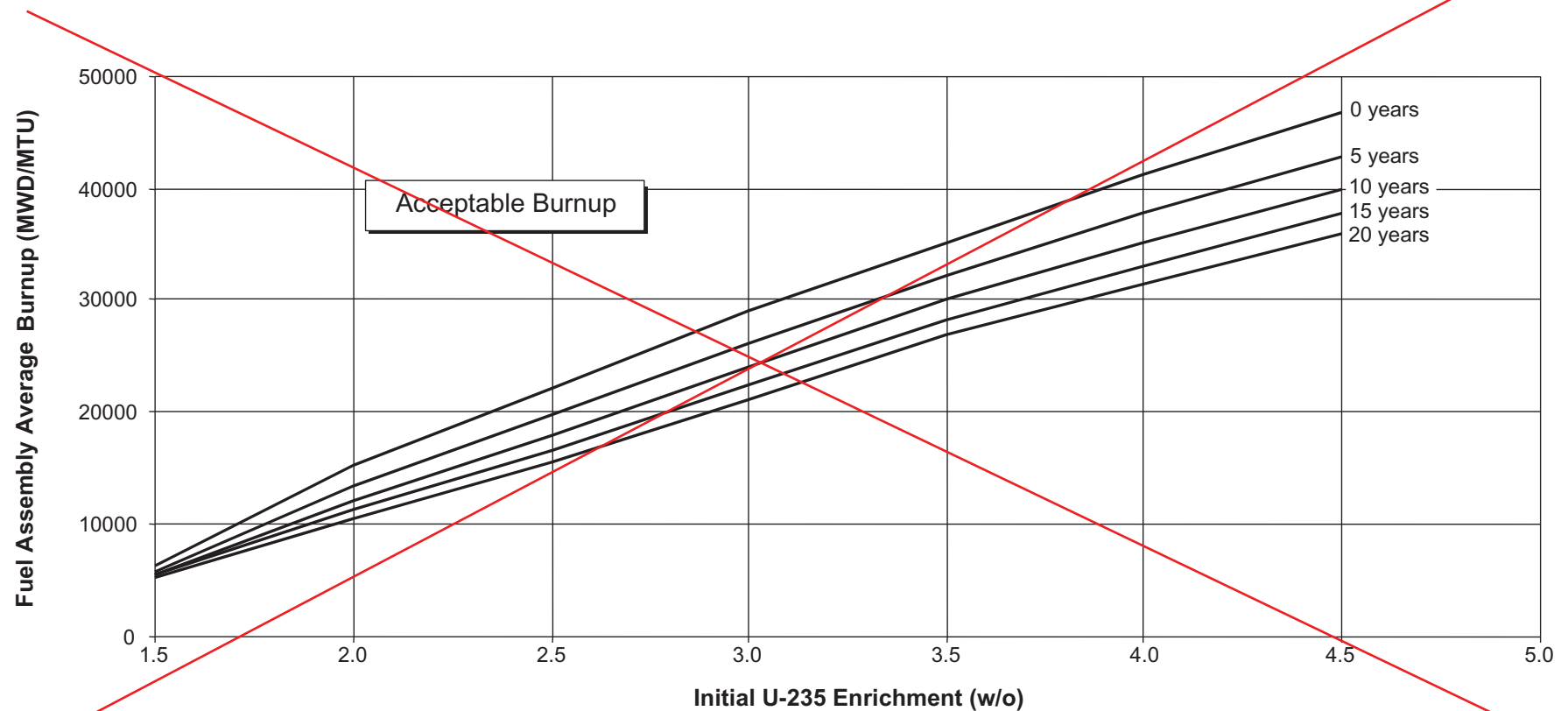
CAPACITY

- 5.6.3 The spent fuel pool storage racks are designed and shall be maintained with a storage capacity limited to no more than 1360 fuel assemblies, and the cask pit storage rack is designed and shall be maintained with a storage capacity limited to no more than 225 fuel assemblies. The total Unit 2 spent fuel pool and cask pit storage capacity is limited to no more than 1585 fuel assemblies.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMITS

- 5.7.1 The components identified in Table 5.7-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7-1.

FIGURE 5-6-1a
Required Fuel Assembly Burnup vs Initial Enrichment and Decay Time
Region II, 1.3 w/o



Pages 5-4C through 5-4F (Amendment 101) and page 5-4G (Amendment 135) have been deleted from the Technical Specifications. The next page is 5-4h.

b

FIGURE 5-6-1b
Required Fuel Assembly Burnup vs Initial Enrichment and Decay Time
Region II, 1.5 w/o

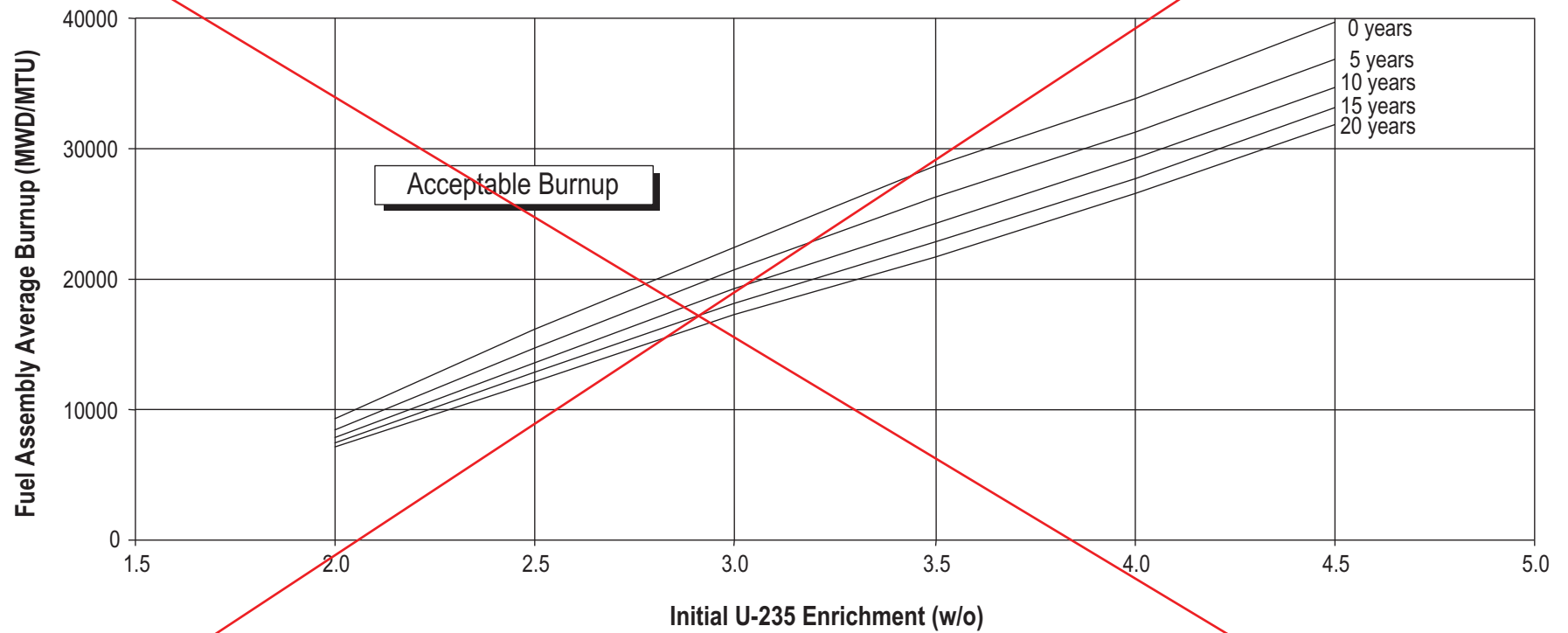


FIGURE 5.6-1c
Required Fuel Assembly Burnup vs Initial Enrichment and Decay Time
Region I, 1.4 w/o

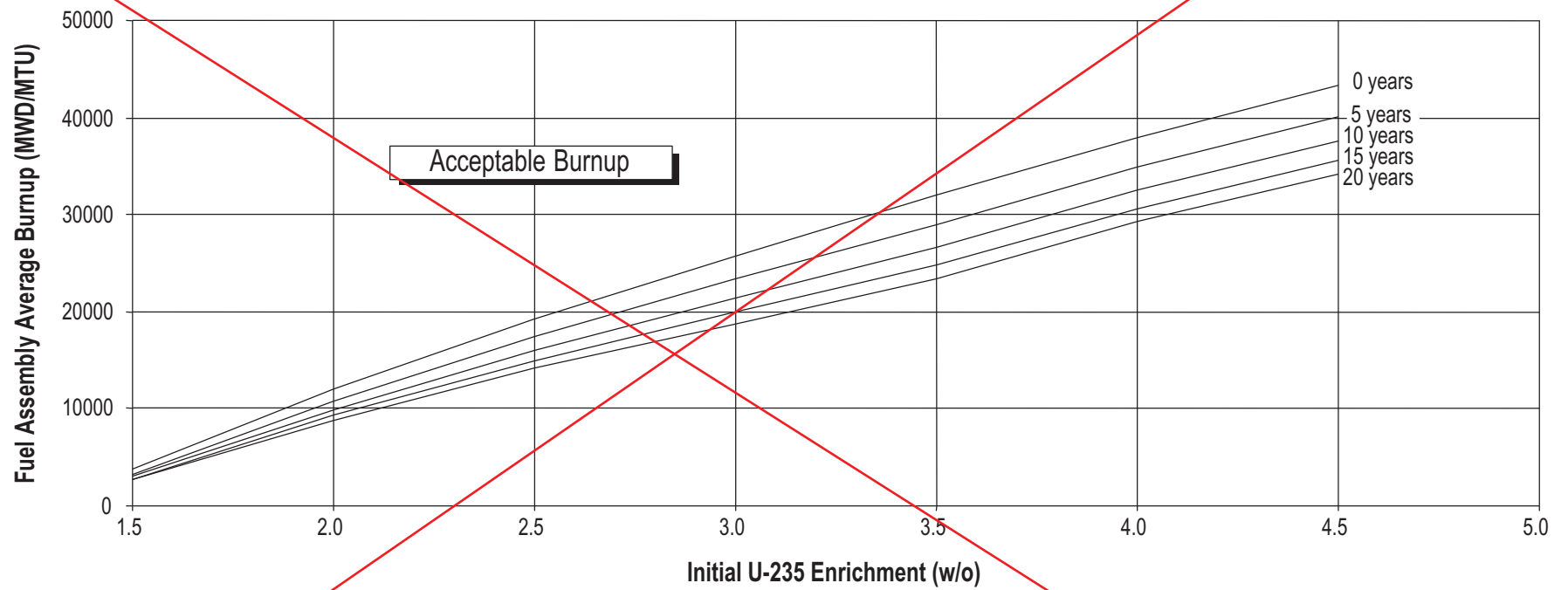


FIGURE 5.6-1d
Required Fuel Assembly Burnup vs Initial Enrichment and Decay Time
Region I, 1.82 w/o

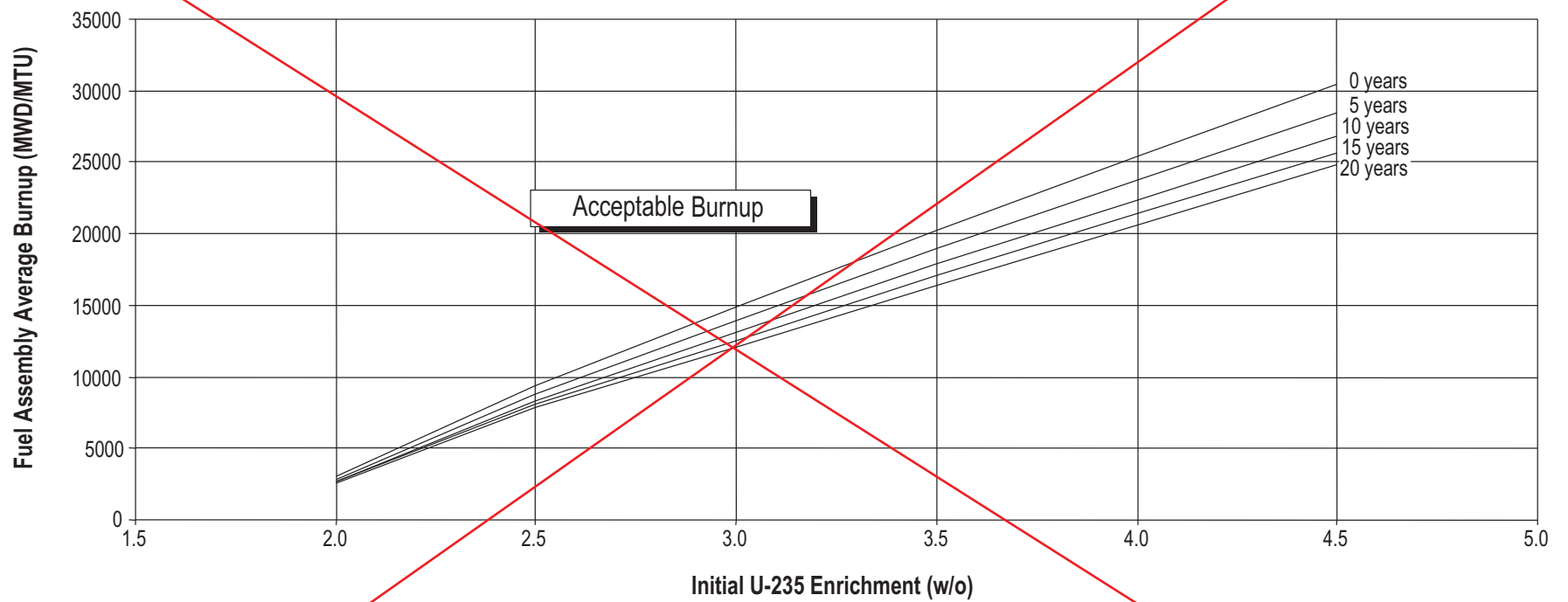
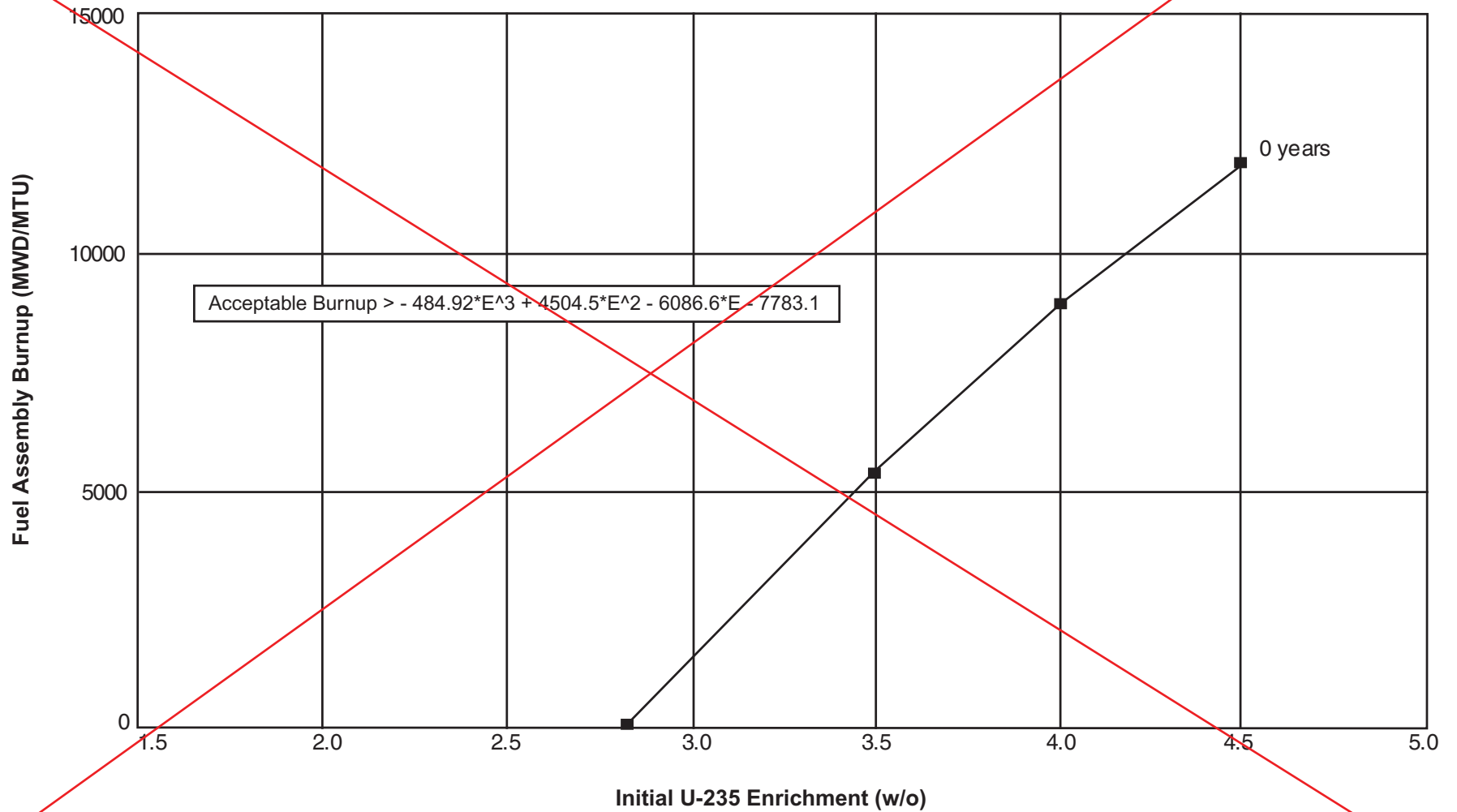


FIGURE 5.6-1e
Required Fuel Assembly Burnup vs Initial Enrichment
Region I, 2.82 w/o



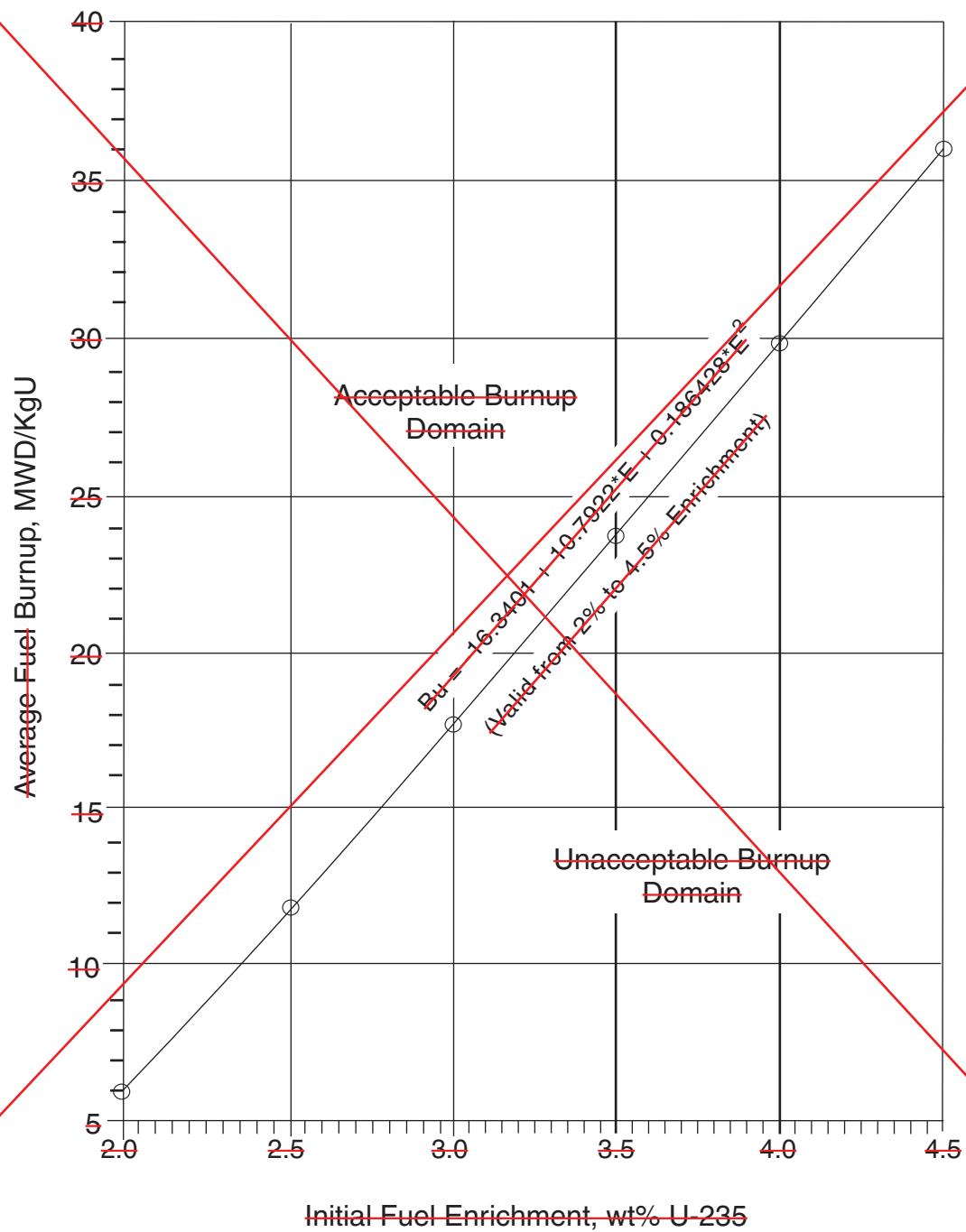


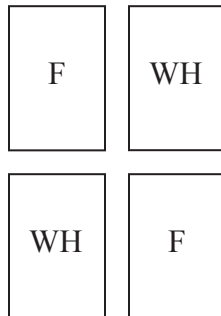
FIGURE 5.6 1f

**REQUIRED FUEL ASSEMBLY BURNUP vs INITIAL ENRICHMENT
REGION II CASK PIT STORAGE RACK**

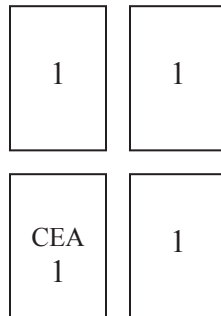
NEW FIGURE 5.6-1

Allowable Storage Patterns
(See Notes 1 and 2)

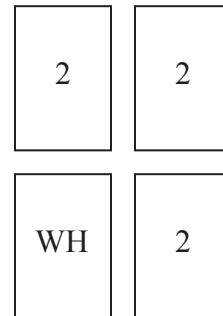
Pattern "A"
See Definition 1



Pattern "B"
See Definition 2



Pattern "C"
See Definition 3



DEFINITIONS:

1. Allowable pattern is Fresh Fuel or fuel of lower reactivity checkerboarded with completely water-filled cells. Diagram is for illustration only, where F represents Fresh Fuel and WH represents a completely water-filled cell.
2. Allowable pattern is placement of fuel assembly type 1 or fuel of lower reactivity in each 2x2 array location with at least one full-length full-strength CEA or equivalent (5 absorber rods) placed in any cell. Minimum burnup for fuel assembly type 1 is defined in Table 5.6-1 as a function of maximum initial planar average enrichment. Diagram is for illustration only.
3. Allowable pattern is placement of fuel assembly type 2 or fuel of lower reactivity in three of the 2x2 array locations in combination with one completely water-filled cell. Minimum burnup for fuel assembly type 2 is defined in Table 5.6-1 as a function of maximum initial planar average enrichment. Diagram is for illustration only.

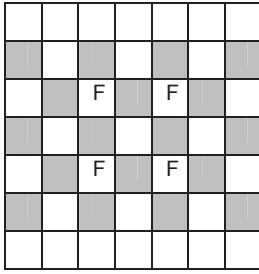
NOTES:

1. The storage arrangements of fuel within a rack module may contain more than one pattern. There are no interface limitations within Region 1 between rack modules or within racks; however, each assembly must meet the burnup requirements of each 2x2 array that it resides within.
2. Open cells within any pattern are acceptable.

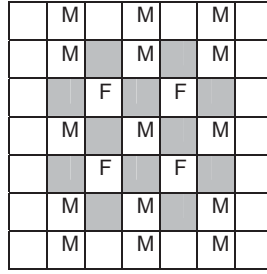
FIGURE 5.6-1
Allowable Region 1 Storage Patterns and Fuel Arrangements

ALLOWED SPECIAL ARRANGEMENTS (See Notes 1 and 2)
Fresh Fuel Assemblies in Region 2 Racks

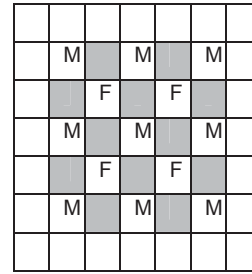
See Definition SR1



See Definition SR2



See Definition SR3



=

Fresh Fuel Assembly



= Fuel Assembly
with Metamic Insert



=

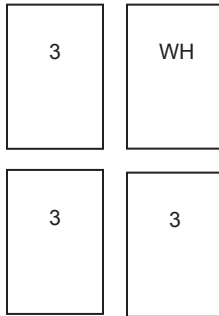
Empty Cell



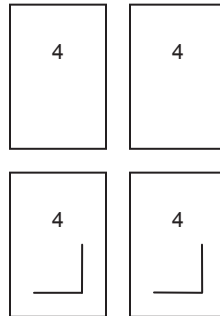
= Fuel Assembly per
Definition

ALLOWABLE STORAGE PATTERNS (See Notes 1 and 2)

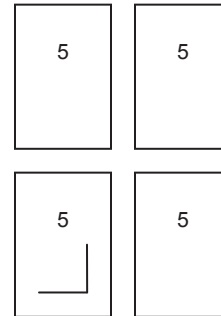
Pattern "D"
See Definition 1



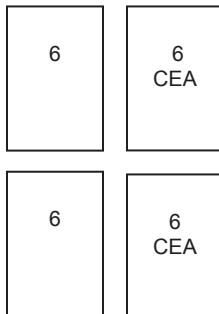
Pattern "E"
See Definition 2



Pattern "F"
See Definition 3



Pattern "G"
See Definition 4



Pattern "H"
See Definition 5

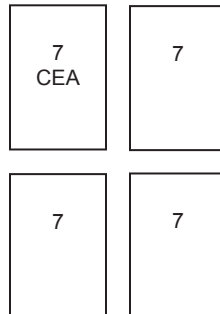


FIGURE 5.6-2 (Sheet 1 of 3)
Allowable Region 2 Storage Patterns and Fuel Arrangements

DEFINITIONS for Figure 5.6-2

1. Allowable pattern is fuel assembly type 3 or fuel of lower reactivity in three of the 2x2 array locations in combination with one completely water-filled cell. Minimum burnup for fuel assembly type 3 is defined in Table 5.6-1 as a function of maximum initial planar average enrichment. Diagram is for illustration only.
2. Allowable pattern is fuel assembly type 4 or fuel of lower reactivity in each of the 2x2 array locations with at least two Metamic inserts placed anywhere in the 2x2 array. Minimum burnup for fuel assembly type 4 is defined in Table 5.6-1 as a function of maximum initial planar average enrichment and cooling time. Diagram is for illustration only.
3. Allowable pattern is fuel assembly type 5 or fuel of lower reactivity in each of the 2x2 array locations with at least one Metamic insert placed anywhere in the 2x2 array. Minimum burnup for fuel assembly type 5 is defined in Table 5.6-1 as a function of maximum initial planar average enrichment and cooling time. Diagram is for illustration only.
4. Allowable pattern is fuel assembly type 6 or fuel of lower reactivity in each of the 2x2 array locations with at least two full-length, full strength 5 finger CEAs or equivalent placed anywhere in the 2x2 array. Minimum burnup for fuel assembly type 6 is defined in Table 5.6-1 as a function of maximum initial planar average enrichment and cooling time. Diagram is for illustration only.
5. Allowable pattern is fuel assembly type 7 in each of the 2x2 array locations with at least one full-length, full strength 5 finger CEA or equivalent placed anywhere in the 2x2 array. Minimum burnup for fuel assembly type 7 is defined in Table 5.6-1 as a function of maximum initial planar average enrichment and cooling time. Diagram is for illustration only.

SR1. Allowable pattern is up to four fresh fuel assemblies or fuel of lower reactivity placed in a 3x3 array in combination with Pattern "D" placed outside the 3x3 array. Fresh fuel shall be placed in the corners of the 3x3 array with completely water-filled cells placed face-adjacent on all sides. A fuel assembly of type 3 or fuel of lower reactivity shall be placed in the center of the 3x3 array. Minimum burnup for fuel assembly type 3 is defined in Table 5.6-1 as a function of maximum initial planar average enrichment. Diagram is for illustration only.

SR2. Allowable pattern is up to four fresh fuel assemblies or fuel of lower reactivity placed in a 3x3 array in combination with Pattern "E" placed outside the 3x3 array. Fresh fuel shall be placed in the corners of the 3x3 array with completely water-filled cells placed face-adjacent on all sides. A fuel assembly of type 4 or fuel of lower reactivity with a Metamic insert shall be placed in the center of the 3x3 array. Minimum burnup for fuel assembly type 4 is defined in Table 5.6-1 as a function of maximum initial planar average enrichment and cooling time. Diagram is for illustration only.

FIGURE 5.6-2 (Sheet 2 of 3)
Allowable Region 2 Storage Patterns and Fuel Arrangements

SR3. Allowable pattern is up to four fresh fuel assemblies or fuel of lower reactivity placed in a 3x3 array in combination with Pattern “F” placed outside the 3x3 array. Fresh fuel shall be placed in the corners of the 3x3 array with completely water-filled cells placed face-adjacent on all sides. A fuel assembly of type 5 or fuel of lower reactivity with a Metamic insert shall be placed in the center of the 3x3 array. Minimum burnup for fuel assembly type 5 is defined in Table 5.6-1 as a function of maximum initial planar average enrichment and cooling time. Diagram is for illustration only.

NOTES

1. The storage arrangements of fuel within a rack module may contain more than one pattern. There are no interface limitations within Region 2 between rack modules or within racks. However, each assembly must meet the burnup requirements of each 2x2 array or allowed special arrangement that it resides within.
2. Open cells within any pattern are acceptable.

FIGURE 5.6-2 (Sheet 3 of 3)
Allowable Region 2 Storage Patterns and Fuel Arrangements

Allowed Region 2 to Region 1 Fuel Alignments
(See Note 1)

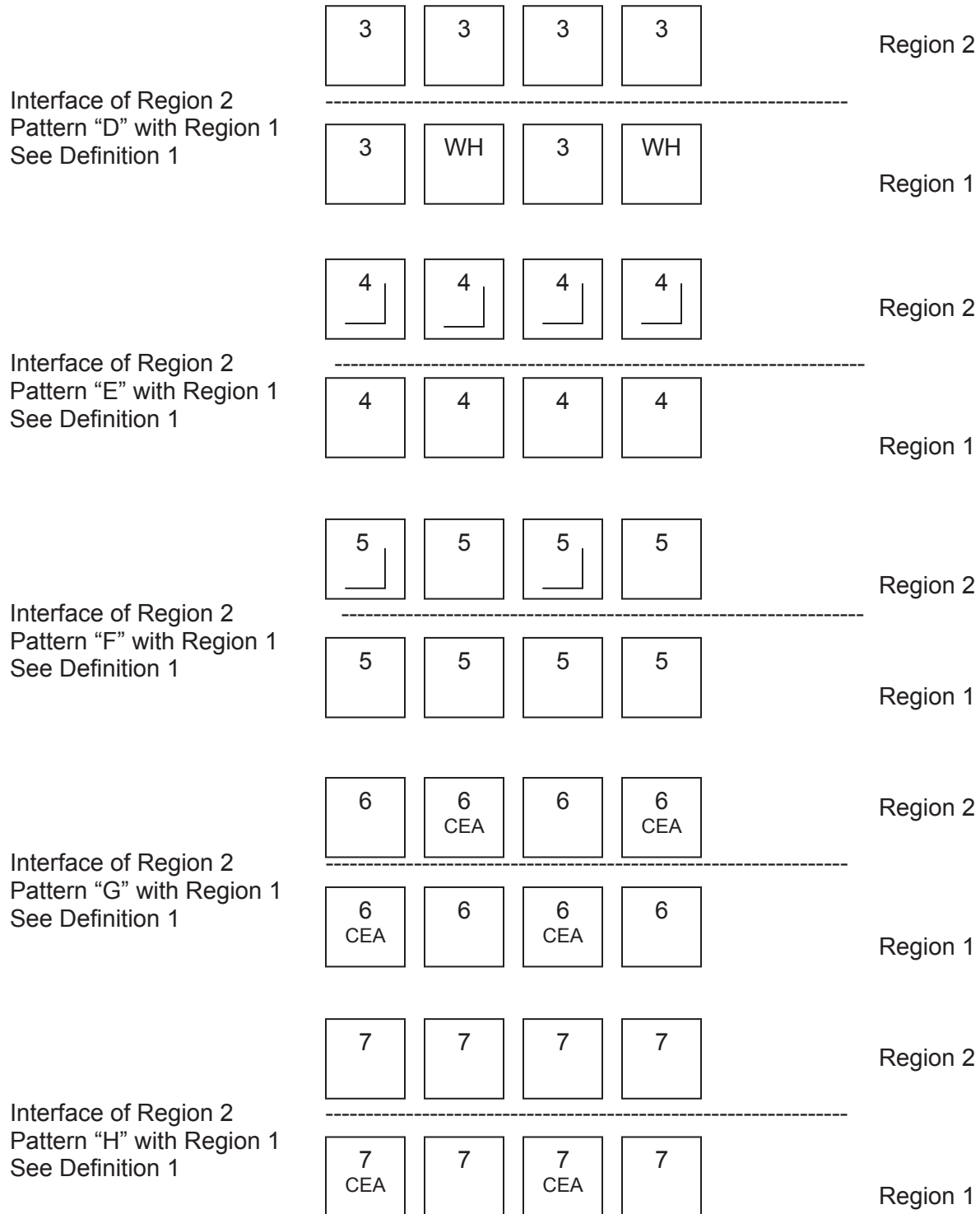


FIGURE 5.6-3 (Sheet 1 of 2)
Interface Requirements between Region 1 and Region 2

DEFINITION:

1. Each 2x2 array that spans Region 1 and Region 2 shall match one of the Region 2 allowable storage patterns as defined by Specification 5.6.1.c.3. Any required Metamic inserts must be placed into the fuel assemblies in Region 2. Locations of water-filled cells or CEAs may be in either Region 1 or Region 2. For interface assemblies, the requirements of Specifications 5.6.1.c.2 and Specification 5.6.1.c.3 shall be followed within Region 1 and Region 2, respectively. The Diagrams are for illustration only.

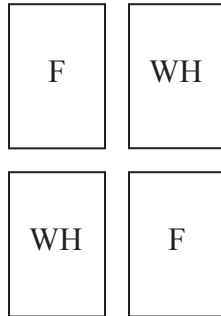
NOTES:

1. Open cells within any pattern are acceptable.

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

Allowable Storage Patterns (See Notes 1 and 2)

Pattern "A"
See Definition 1



Pattern "B"
See Definition 2



DEFINITIONS:

1. Allowable pattern is Fresh Fuel or lower reactivity fuel checkerboarded with completely water-filled cells. Diagram is for illustration only, where F represents Fresh Fuel and WH represents a completely water-filled cell.
2. Allowable pattern is placement of fuel assembly type 8 or fuel of lower reactivity in three of the 2x2 array locations in combination with one completely water-filled cell in any location. Minimum burnup for fuel assembly type 8 is defined in Table 5.6-1 as a function of maximum initial planar average enrichment. Diagram is for illustration only.

NOTES:

1. The storage arrangements of fuel within a rack module may contain more than one pattern. There are no interface limitations within the cask pit storage rack, however, each assembly must meet the burnup requirements of each 2x2 array that it resides within.
2. Open cells within any pattern are acceptable.

FIGURE 5.6-4
Allowable Cask Pit Storage Rack Patterns

TABLE 5.6-1
Minimum Burnup Coefficients

Fuel Type	Cooling Time (Years)	Coefficients		
		A	B	C
1	0	-33.4237	25.6742	-1.6478
2	0	-25.3198	14.3200	-0.4042
3	0	-23.4150	16.2050	-0.5500
4	0	-33.6414	25.0670	-1.5551
	2.5	-32.3764	23.9988	-1.5075
	5	-30.9234	22.9382	-1.4372
	10	-28.4951	21.1511	-1.3029
	15	-27.2024	20.2802	-1.2479
	20	-25.2009	18.6218	-1.0364
5	0	-24.8402	23.5991	-1.2082
	2.5	-23.0170	21.6493	-1.0298
	5	-21.9293	20.6257	-0.9730
	10	-20.0813	19.0808	-0.9022
	15	-19.5503	18.5429	-0.9129
	20	-18.7485	17.7308	-0.8390
6	0	-32.4900	25.3077	-1.5518
	2.5	-31.1598	23.9185	-1.4435
	5	-29.2169	22.5424	-1.3274
	10	-26.8886	20.6662	-1.1425
	15	-25.5703	19.7629	-1.1129
	20	-24.5754	18.9056	-1.0147
7	0	-24.6989	24.1660	-1.2578
	2.5	-23.0399	22.3047	-1.0965
	5	-21.3290	20.7413	-0.9613
	10	-20.0836	19.4780	-0.8949
	15	-19.2480	18.5880	-0.8685
	20	-18.6424	18.1241	-0.8950
8	0	-47.5000	12.5000	0.0000

NOTES:

1. To qualify in a fuel type, the calculated burnup of a fuel assembly must exceed the “minimum burnup” determined for the “cooling time” and “maximum initial planar enrichment” of the fuel assembly. The “minimum burnup” for any fuel type is determined from the following polynomial function:

$$BU = A + B \cdot E + C \cdot E^2, \text{ where:}$$

BU = Minimum Burnup (GWD/MTU)

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

A, B, C = Coefficients for each fuel type

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

ADMINISTRATIVE CONTROLS

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 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. which
- b) ~~The first Type A test performed after the June 1992 Type A test shall be prior to startup following the SL 2-17 refueling outage.~~

43.48

The peak calculated containment internal pressure for the design basis loss of coolant accident P_a is 41.8 psig. The containment design pressure is 44 psig.

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

ADMINISTRATIVE CONTROLS

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. ~~CENPD-199 P, Rev. 1 P-A, "C-E Setpoint Methodology: CE Local Power Density and DNB LSSS and LCO Setpoint Methodology for Analog Protection Systems," January 1986.~~
4. ~~CENPD-266 P-A, "The ROCS and DIT Computer Code for Nuclear Design," April 1983.~~
5. CENPD-275-P, Revision 1-P-A, "C-E Methodology for Core Designs Containing Gadolinia-Urania Burnable Absorbers," May 1988.
6. ~~CENPD-188 A, "HERMITE: A Multi-Dimensional Space-Time Kinetics Code for PWR Transients," July 1976.~~

DELETED

, & Revision 1-P Supplement 1-P-A, April 1999.

ADMINISTRATIVE CONTROLS (Continued)

CORE OPERATING LIMITS REPORT (COLR) (Continued)

b. (Continued)

7. ~~CENPD-153-P, Rev. 1-P-A, "Evaluation of Uncertainty in the Nuclear Power Peaking Measured by the Self-Powered, Fixed Incore Detector System," May 1980.~~
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. ~~CEN-123(F)-P, "Statistical Combination of Uncertainties Methodology Part 2: Combination of System Parameter Uncertainties in Thermal Margin Analyses for St. Lucie Unit 1," January 1980.~~
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. ~~CEN-371(F)-P, "Extended Statistical Combination of Uncertainties," July 1989.~~
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. ~~CENPD-161-P-A, "TORC Code, A Computer Code for Determining the Thermal Margin of a Reactor Core," April 1986.~~
16. ~~CENPD-162-P-A, "Critical Heat Flux Correlation for C-E Fuel Assemblies with Standard Spacer Grids Part 1, Uniform Axial Power Distribution," April 1975.~~
17. ~~CENPD-207-P-A, "Critical Heat Flux Correlation for C-E Fuel Assemblies with Standard Spacer Grids Part 2, Non-uniform Axial Power Distribution," December 1984.~~
18. ~~CENPD-206-P-A, "TORC Code, Verification and Simplified Modeling Methods," June 1981.~~

DELETED

ADMINISTRATIVE CONTROLS (continued)

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. ~~Letter, A.E. Scherer Enclosure 1 P to LD-82-001, "CESEC Digital Simulation of a Combustion Engineering Nuclear Steam Supply System," December 1981.~~
38. ~~Safety Evaluation Report, "CESEC Digital Simulation of a Combustion Engineering Steam Supply System (TAC No.: 01142)," October 27, 1983.~~
39. ~~CENPD-282-P-A, Volumes 1, 2 and 3, and Supplement 1, "Technical Manual for the CENTS Code," February 1991, February 1991, October 1991, and June 1993, respectively.~~
40. ~~CEN-121(B)-P, "CEAW, Method of Analyzing Sequential Control Element Assembly Group Withdrawal Event for Analog Protected Systems," November 1979 (NRC SER dated December 21, 1999, Letter K. N. Jabbour (NRC) to T.F. Plunkett (FPL), TAC No. MA4523).~~
41. ~~CEN-133(B), "FIESTA, A One Dimensional, Two Group Space Time Kinetics Code for Calculating PWR Scram Reactivities," November 1979 (NRC SER dated December 21, 1999, Letter K. N. Jabbour (NRC) to T.F. Plunkett (FPL), TAC No. MA4523).~~
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. ~~CENPD-183-A, "C-E Methods for Loss of Flow Analysis," June 1984.~~
45. ~~CENPD-190-A, "C-E Method for Control Element Assembly Ejection Analysis," July 1976.~~

DELETED

ADMINISTRATIVE CONTROLS (continued)

CORE OPERATING LIMITS REPORT (COLR) (continued)

b. (continued)

DELETED

46. ~~CENPD-199-P, Rev. 1-P-A, Supplement 2-P-A, "CE Setpoint Methodology," June 1998.~~

47. ~~CENPD-382-P-A, "Methodology for Core Designs Containing Erbium Burnable Absorbers," August 1993.~~

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. ~~CEN-365(L), "Boric Acid Concentration Reduction Effort, Technical Bases and Operational Analysis," June 1988 (NRC SER dated March 13, 1989, Letter J.A. Norris (NRC) to W.F. Conway (FPL), TAC No. 69325).~~

DELETED

54. ~~DP-456, F.M. Stern (CE) to E. Case (NRC), dated August 19, 1974, Appendix 6B to CESSAR System 80 PSAR (NRC SER, NUREG-75/112, Docket No. STN 50-470, "NRC SER - Standard Reference System, CESSAR System 80," December 1975).~~

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.

ADMINISTRATIVE CONTROLS (continued)

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.

INSERT 9



~~64. 30% SCTP PLA Submittal and the SER.~~

- 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.

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.



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.

**ATTACHMENT 3
LICENSE AMENDMENT REQUEST
EXTENDED POWER UPRATE**

**RENEWED FACILITY OPERATING LICENSE
AND
TECHNICAL SPECIFICATIONS
CLEAN PAGES**

**FLORIDA POWER & LIGHT
ST. LUCIE UNIT 2**

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. XXX are hereby incorporated in the renewed license. FPL shall operate the facility in accordance with the Technical Specifications. |

INDEX

DEFINITIONS

<u>SECTION</u>	<u>PAGE</u>
<u>1.0 DEFINITIONS</u>	
1.1 ACTION	1-1
1.2 AXIAL SHAPE INDEX	1-1
1.3 AZIMUTHAL POWER TILT	1-1
1.4 CHANNEL CALIBRATION	1-1
1.5 CHANNEL CHECK	1-1
1.6 CHANNEL FUNCTIONAL TEST	1-2
1.7 CONTAINMENT VESSEL INTEGRITY	1-2
1.8 CONTROLLED LEAKAGE	1-2
1.9 CORE ALTERATION	1-2
1.9a CORE OPERATING LIMITS REPORT (COLR)	1-2
1.10 DOSE EQUIVALENT I-131	1-3
1.11 DOSE EQUIVALENT XE-133	1-3
1.12 ENGINEERED SAFETY FEATURES RESPONSE TIME	1-3
1.13 FREQUENCY NOTATION	1-3
1.14 GASEOUS RADWASTE TREATMENT SYSTEM	1-3
1.15 IDENTIFIED LEAKAGE	1-3
1.16 DELETED	1-4
1.17 MEMBER(S) OF THE PUBLIC	1-4
1.18 OFFSITE DOSE CALCULATION MANUAL (ODCM)	1-4
1.19 OPERABLE – OPERABILITY	1-4
1.20 OPERATIONAL MODE – MODE	1-4
1.21 PHYSICS TESTS	1-4
1.22 PRESSURE BOUNDARY LEAKAGE	1-5
1.23 PROCESS CONTROL PROGRAM	1-5
1.24 PURGE – PURGING	1-5
1.25 RATED THERMAL POWER	1-5
1.26 REACTOR TRIP SYSTEM RESPONSE TIME	1-5
1.27 REPORTABLE EVENT	1-5
1.28 SHIELD BUILDING INTEGRITY	1-5
1.29 SHUTDOWN MARGIN	1-6
1.30 SITE BOUNDARY	1-6

INDEX

LIST OF FIGURES

<u>FIGURE</u>	<u>PAGE</u>
2.1-1 REACTOR CORE THERMAL MARGIN SAFETY LIMIT LINES FOUR REACTOR COOLANT PUMPS OPERATING	2-3
2.2-1 LOCAL POWER DENSITY – HIGH TRIP SETPOINT PART 1 (FRACTION OF RATED THERMAL POWER VERSUS QR_2)	2-7
2.2-2 LOCAL POWER DENSITY – HIGH TRIP SETPOINT PART 2 (QR_2 VERSUS Y_1).....	2-8
2.2-3 THERMAL MARGIN/LOW PRESSURE TRIP SETPOINT PART 1 (Y_1 , VERSUS A_1).....	2-9
2.2-4 THERMAL MARGIN/LOW PRESSURE TRIP SETPOINT PART 2 (FRACTION OF RATED THERMAL POWER VERSUS QR_1)	2-10
3.1-1 MINIMUM BMT VOLUME VS STORED BORIC ACID CONCENTRATION ...3/4	1-15
3.1-1a DELETED	
3.1-2 DELETED	
3.2-1 DELETED	
3.2-2 DELETED	
3.2-3 DELETED	
4.2-1 DELETED	
3.2-4 DELETED	
3.4-1 DELETED	
3.4-2 ST. LUCIE UNIT 2 REACTOR COOLANT SYSTEM PRESSURE- TEMPERATURE LIMITS FOR 47 EFY, HEATUP, CORE CRITICAL, AND INSERVICE TEST	3/4 4-31a

INDEX

LIST OF FIGURES (continued)

<u>FIGURE</u>	<u>PAGE</u>
3.4-3 ST. LUCIE UNIT 2 REACTOR COOLANT SYSTEM PRESSURE- TEMPERATURE LIMITS FOR 47 EFPY, COOLDOWN AND INSERVICE TEST	3/4 4-31b
3.4-4 DELETED	3/4 4-32
4.7-1 SAMPLING PLAN FOR SNUBBER FUNCTIONAL TEST	3/4 7-25
5.1-1 SITE AREA MAP	5-2
5.6-1a DELETED	
5.6-1b DELETED	
5.6-1c DELETED	
5.6-1d DELETED	
5.6-1e DELETED	
5.6-1f DELETED	
5.6-1 ALLOWABLE REGION 1 STORAGE PATTERNS AND FUEL ARRANGEMENTS.....	5-4h
5.6-2 ALLOWABLE REGION 2 STORAGE PATTERNS AND FUEL ARRANGEMENTS (Sheet 1 of 3).....	5-4i
5.6-2 ALLOWABLE REGION 2 STORAGE PATTERNS AND FUEL ARRANGEMENTS (Sheet 2 of 3).....	5-4j
5.6-2 ALLOWABLE REGION 2 STORAGE PATTERNS AND FUEL ARRANGEMENTS (Sheet 3 of 3).....	5-4k
5.6-3 INTERFACE REQUIREMENTS BETWEEN REGION 1 AND REGION 2 (Sheet 1 of 2)	5-4l
5.6-3 INTERFACE REQUIREMENTS BETWEEN REGION 1 AND REGION 2 (Sheet 2 of 2)	5-4m
5.6-4 ALLOWABLE CASK PIT STORAGE RACK PATTERNS	5-4n
6.2-1 DELETED	6-3
6.2-2 DELETED	6-4

INDEX

LIST OF TABLES

<u>TABLE</u>	<u>PAGE</u>
1.1 FREQUENCY NOTATION	1-8
1.2 OPERATIONAL MODES	1-9
2.2-1 REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS	2-4
3.1-1 DELETED	
3.2-1 DELETED	3/4 2-11
3.2-2 DELETED	3/4 2-15
3.3-1 REACTOR PROTECTIVE INSTRUMENTATION	3/4 3-2
3.3-2 DELETED	
4.3-1 REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS	3/4 3-8
3.3-3 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION	3/4 3-12
3.3-4 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES	3/4 3-17
3.3-5 DELETED	
4.3-2 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS	3/4 3-22
3.3-6 RADIATION MONITORING INSTRUMENTATION.....	3/4 3-25
4.3-3 RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS	3/4 3-28
3.3-8 DELETED	
4.3-5 DELETED	
5.6-1 MINIMUM BURNUP COEFFICIENTS	5-40

DEFINITIONS

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 ($\mu\text{Ci}/\text{gram}$) 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:
- 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
 - 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
 - Reactor Coolant System leakage through a steam generator to the secondary system (primary-to-secondary leakage).

DEFINITIONS

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.

DEFINITIONS

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.

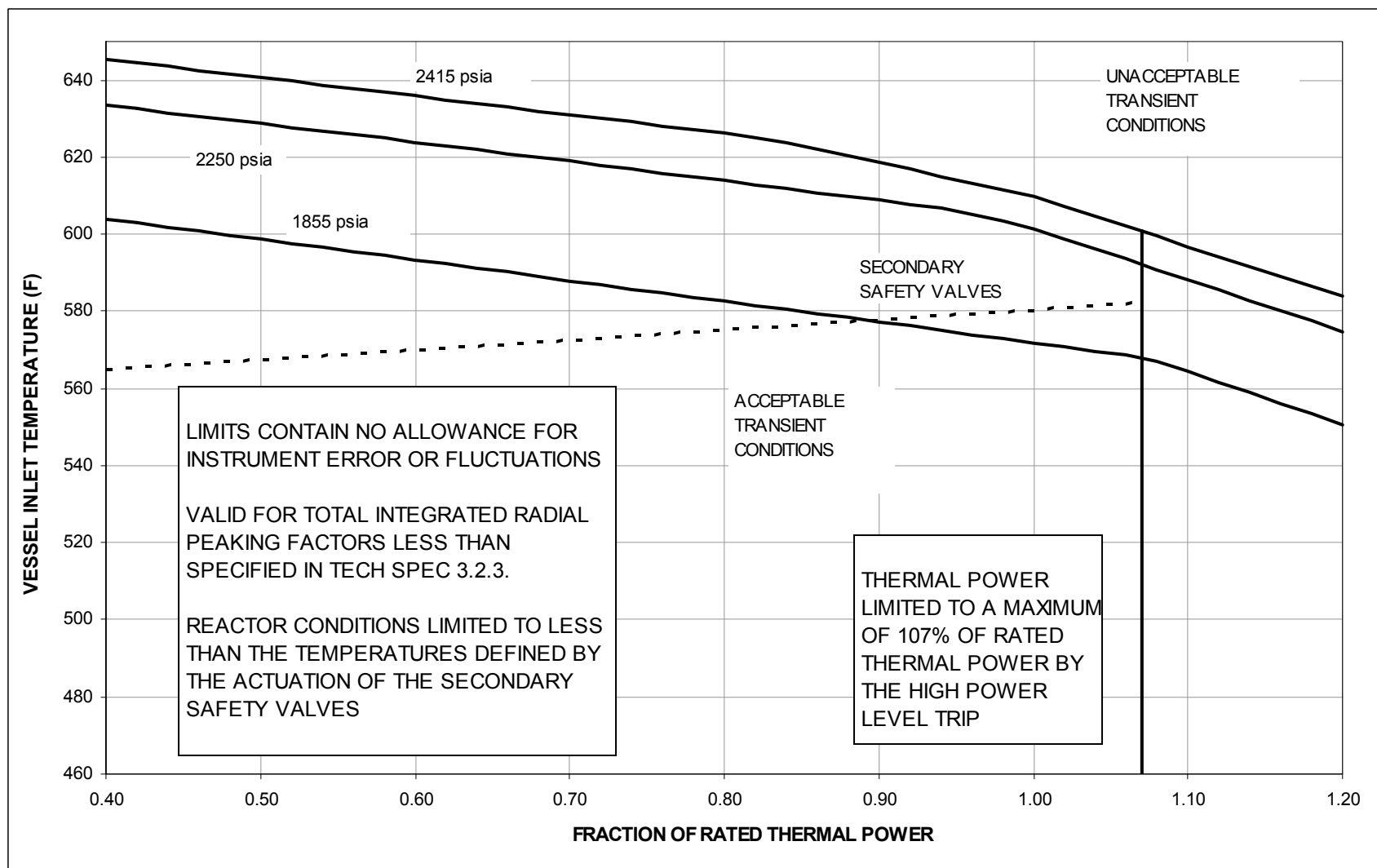
REPORTABLE EVENT

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

<u>FUNCTIONAL UNIT</u>	<u>TRIP SETPOINT</u>	<u>ALLOWABLE VALUES</u>
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 ⁽¹⁾ Difference – High (Logic in TM/LP Trip Unit)	≤ 120.0 psid	≤ 132.0 psid
8. Steam Generator Level – Low ^{(6), (7)}	≥ 35.0% ⁽³⁾	≥ 34.1% ⁽³⁾

TABLE 2.2-1 (Continued)
REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

<u>FUNCTIONAL UNIT</u>	<u>TRIP SETPOINT</u>	<u>ALLOWABLE VALUES</u>
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⁾	$\geq 95.4\%$ of minimum Reactor Coolant flow with four pumps operating*	$\geq 94.9\%$ 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 COLR Table 3.2-2.

** 10-minute time delay after relay actuation.

TABLE 2.2-1 (Continued)
REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

TABLE NOTATION

- (1) Trip may be manually bypassed below 0.5% of RATED THERMAL POWER during testing pursuant to Special Test Exception 3.10.3; bypass shall be automatically removed when Wide Range Logarithmic Neutron Flux power is greater than or equal to 0.5% of RATED THERMAL POWER.
- (2) Trip may be manually bypassed below 705 psig; bypass shall be automatically removed at or above 705 psig.
- (3) % of the narrow range steam generator level indication.
- (4) Trip may be bypassed below $10^{-4}\%$ and above 15% of RATED THERMAL POWER; bypass shall be automatically removed when Wide Range Logarithmic Neutron Flux power is $\geq 10^{-4}\%$ and Power Range Neutron Flux power $\leq 15\%$ of RATED THERMAL POWER.
- (5) Trip may be bypassed below 15% of RATED THERMAL POWER; bypass shall be automatically removed when Power Range Neutron Flux power is greater than or equal to 15% of RATED THERMAL POWER.
- (6) 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.
- (7) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance of the Trip Setpoint or a value that is more conservative than the Trip Setpoint, otherwise that channel shall not be returned to OPERABLE status. The Trip Setpoint and the methodology used to determine the Trip Setpoint, the as-found acceptance criteria band, and the as-left acceptance criteria are specified in the UFSAR.

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 K_{eff} 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 K_{eff} 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.

* 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.

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 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.

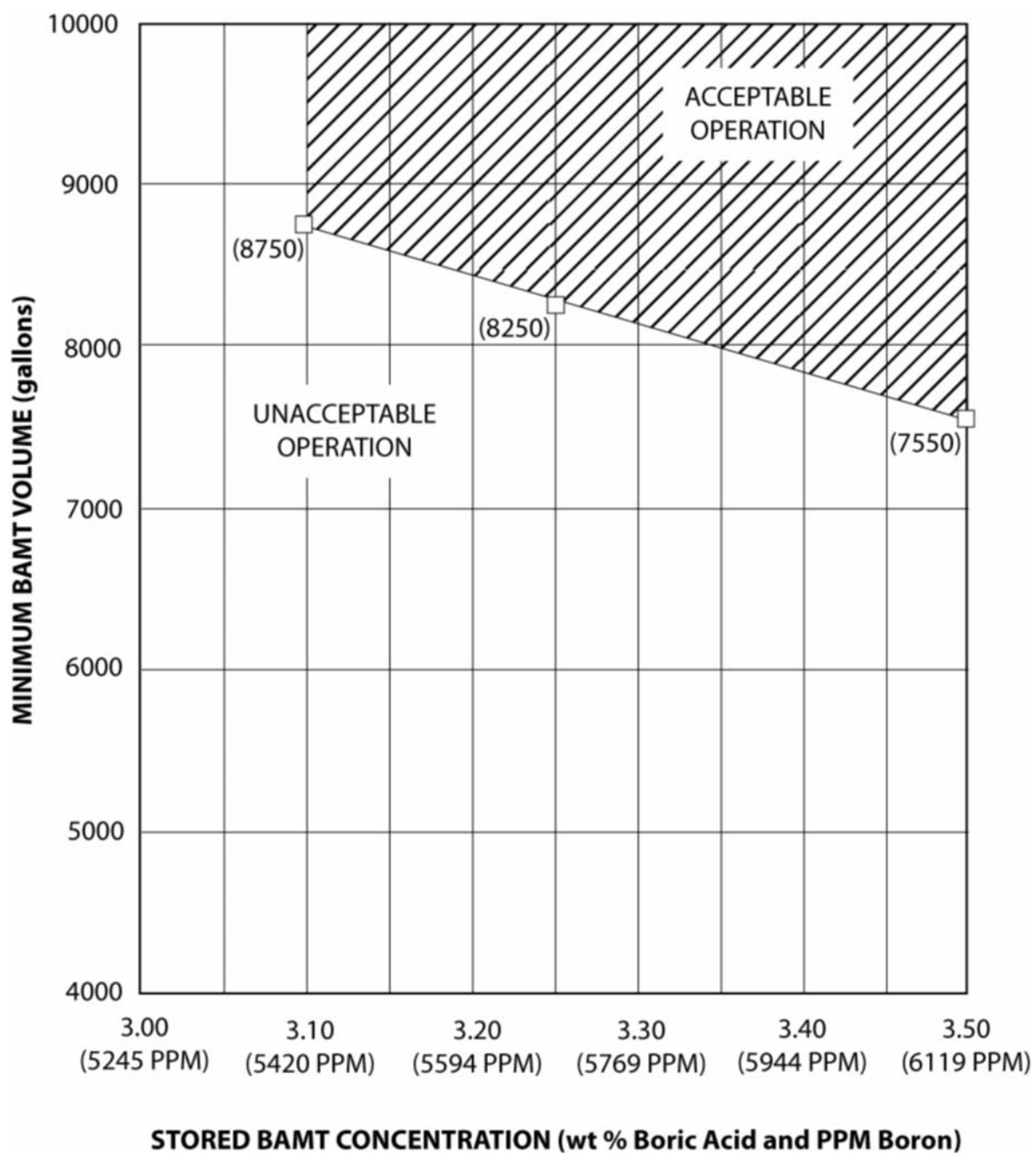
ACTION:

- 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 shown on Table 3.2-2 of the COLR:

- a. Cold Leg Temperature
- b. Pressurizer Pressure*
- c. Reactor Coolant System Total Flow Rate
- d. AXIAL SHAPE INDEX

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

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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 ≥ 2410.3 psig and ≤ 2560.3 psig.*

APPLICABILITY: MODES 1, 2, 3, and 4 with all RCS cold leg temperatures $> 230^{\circ}\text{F}$.

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 $\leq 230^{\circ}\text{F}$ 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 $\pm 1\%$ of 2500 psia.

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

REACTOR COOLANT SYSTEM

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 $\mu\text{Ci}/\text{gram}$ DOSE EQUIVALENT I-131, verify DOSE EQUIVALENT I-131 is $\leq 60.0 \mu\text{Ci}/\text{gram}$ once per four hours.
- b. With the specific activity of the primary coolant > 1.0 $\mu\text{Ci}/\text{gram}$ DOSE EQUIVALENT I-131, but $\leq 60.0 \mu\text{Ci}/\text{gram}$ 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 $\mu\text{Ci}/\text{gram}$ limit. Specification 3.0.4 is not applicable.
- c. With the specific activity of the primary coolant > 1.0 $\mu\text{Ci}/\text{gram}$ DOSE EQUIVALENT I-131 for greater than 48 hours during one continuous time interval, or > 60.0 $\mu\text{Ci}/\text{gram}$ 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 $\mu\text{Ci}/\text{gram}$ 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 $\mu\text{Ci}/\text{gram}$ DOSE EQUIVALENT XE-133 limit. Specification 3.0.4 is not applicable.
- e. With the specific activity of the primary coolant > 518.9 $\mu\text{Ci}/\text{gram}$ 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**

<u>TYPE OF MEASUREMENT AND ANALYSIS</u>	<u>MINIMUM FREQUENCY</u>	<u>MODES IN WHICH SAMPLE AND ANALYSIS REQUIRED</u>	
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

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

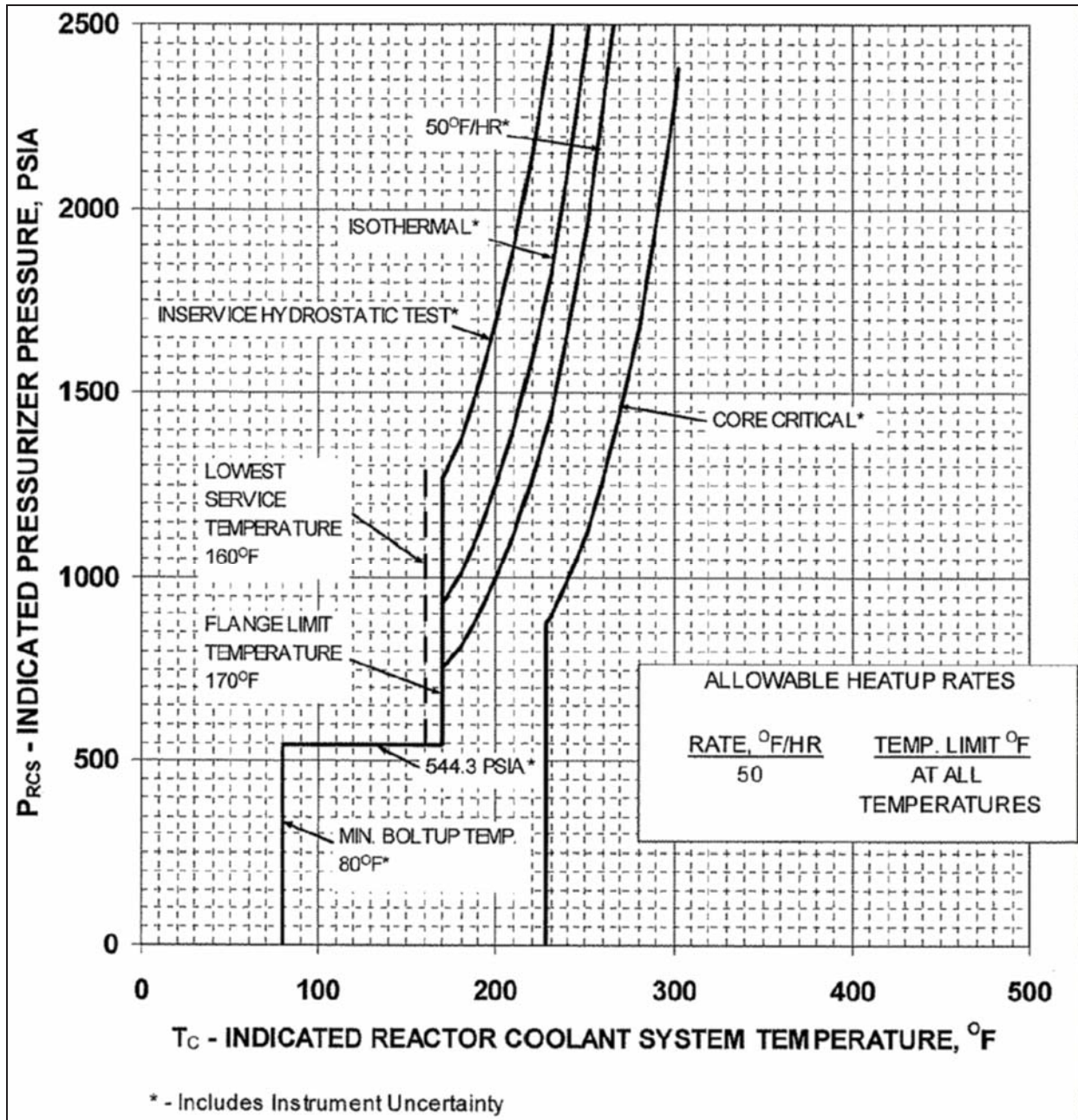


FIGURE 3.4-3
ST. LUCIE UNIT 2 REACTOR COOLANT SYSTEM PRESSURE-TEMPERATURE
LIMITS FOR 47 EFY, COOLDOWN, AND INSERVICE TEST

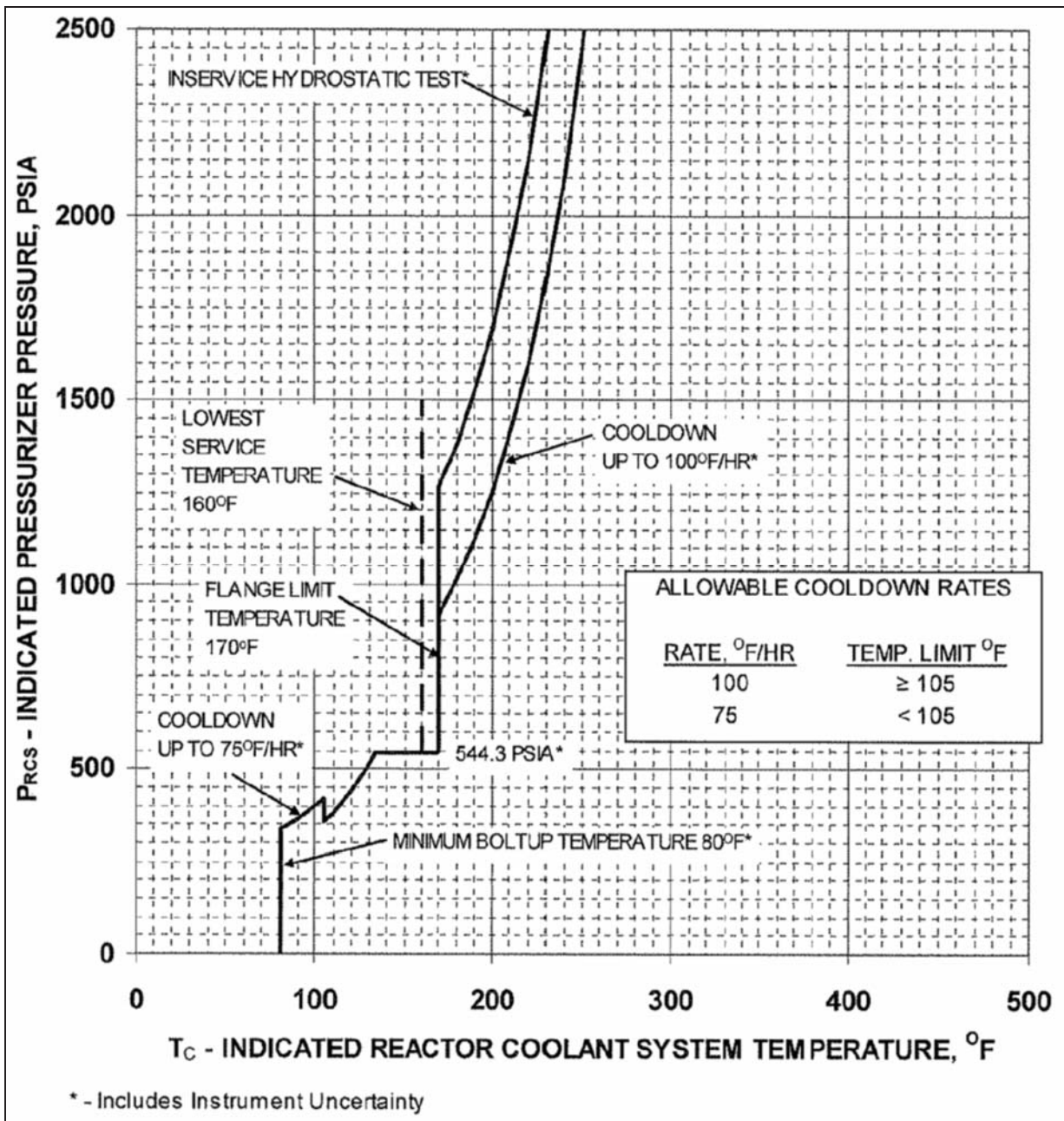


TABLE 3.4-3

LOW TEMPERATURE RCS OVERPRESSURE PROTECTION RANGE

Operating Period, <u>EFY</u>	<u>Cold Leg Temperature, °F</u>	
	<u>During Heatup</u>	<u>During Cooldown</u>
≤ 47	≤ 246	≤ 224

TABLE 3.4-4

MINIMUM COLD LEG TEMPERATURE FOR PORV USE FOR LTOP

Operating Period <u>EFY</u>	<u>Cold Leg Temperature, °F</u>	
	<u>During Heatup</u>	<u>During Cooldown</u>
≤ 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.

* 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.

EMERGENCY CORE COOLING SYSTEMS

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.

* One ECCS subsystem charging pump shall satisfy the flow path requirements of Specification 3.1.2.2.a and 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.

EMERGENCY CORE COOLING SYSTEMS

SURVEILLANCE REQUIREMENTS (continued)

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 ± 0.5 grams of TSP from a TSP storage basket is submerged, without agitation, in 10.0 ± 0.1 gallons of $120 \pm 10^\circ\text{F}$ 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 than 55°F or greater than 100°F.

TABLE 3.7-2
STEAM LINE SAFETY VALVES PER LOOP

	<u>VALVE NUMBER</u>		<u>LIFT SETTING*</u>
	<u>Header A</u>	<u>Header B</u>	
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

* $\pm 3\%$ for valves a through d and $\pm 2\%$ 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.

* 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.

ELECTRICAL POWER SYSTEMS

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 ± 210 volts and 60 ± 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 ± 210 volts and 60 ± 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

SURVEILLANCE REQUIREMENTS (Continued)

- 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 actuation signal.
- 7. Verifying the diesel generator operates for at least 24 hours**** .
 - a) Within 10 seconds, generator voltage and frequency shall be 4160 ± 420 volts and 60 ± 1.2 Hz.
 - b) Steady-state generator voltage and frequency shall be 4160 ± 210 volts and 60 ± 0.6 Hz and shall be maintained throughout this test.
 - c) During the first 2 hours of this test, the diesel generator shall be loaded within a load band of 3800 to 3985 kW[#], and
 - d) During the remaining 22 hours of this test, the diesel generator shall be loaded within a load band of 3450 to 3685 kW[#].
- 8. Verifying that the auto-connected loads to each diesel generator do not exceed the 2000-hour rating of 3935 kW.
- 9. Verifying the diesel generator's capability to:
 - a) Synchronize with the offsite power source while the generator is loaded with its emergency loads upon a simulated restoration of offsite power.
 - b) Transfer its load to the offsite power source, and
 - c) Be restored to its standby status.
- 10. Verifying that with the diesel generator operating in a test mode (connected to its bus), a simulated safety injection signal overrides the test mode by (1) returning the diesel generator to standby operation and (2) automatically energizes the emergency loads with offsite power.
- 11. Verifying that the fuel transfer pump transfers fuel from each fuel storage tank to the engine-mounted tanks of each diesel via the installed cross connection lines.

This band is meant as guidance to avoid routine overloading of the engine. Variations in load in excess of this band due to changing bus loads shall not invalidate 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.

* 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.

REFUELING OPERATIONS

3/4.9.11 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 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.

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.

* 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 202,500 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 $\mu\text{Ci}/\text{gram}$ DOSE EQUIVALENT XE-133. |

DESIGN FEATURES

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1-1.

5.6 FUEL STORAGE

CRITICALITY

- 5.6.1 a. The spent fuel storage racks are designed and shall be maintained with:
1. A k_{eff} equivalent to less than 1.0 when flooded with unborated water, including a conservative allowance for biases and uncertainties as described in Section 9.1 of the Updated Final Safety Analysis Report.
 2. A k_{eff} equivalent to less than or equal to 0.95 when flooded with water containing 500 ppm boron, including a conservative allowance for biases and uncertainties as described in Section 9.1 of the Updated Final Safety Analysis Report.
 3. A nominal 8.965 inch center-to-center distance between fuel assemblies placed in the spent fuel pool storage racks and a nominal 8.80 inch center-to-center distance between fuel assemblies placed in the cask pit storage rack.
 4. For storage of enriched fuel assemblies, requirements of Specification 5.6.1.a.1 and 5.6.1.a.2 shall be met by positioning fuel in the spent fuel pool storage racks consistent with the requirements of Specification 5.6.1.c or in configurations that have been shown to comply with Specifications 5.6.1.a.1 and 5.6.1.a.2 using the methodology as described in Section 9.1 of the Updated Final Safety Analysis Report.
 5. Fissile material, not contained in a fuel assembly lattice, shall be stored in accordance with the requirements of Specifications 5.6.1.a.1 and 5.6.1.a.2.
- b. The 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 pool storage racks shall be controlled as described below.
1. The maximum initial planar average U-235 enrichment of any fuel assembly inserted in a spent fuel pool 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 pattern definitions of Figure 5.6-1 and the minimum burnup requirements as defined in Table 5.6-1. (See Specification 5.6.1.c.7 for exceptions)
 3. Fuel placed in Region 2 of the spent fuel pool storage racks shall comply with the storage pattern definitions or allowed special arrangement definitions of Figure 5.6-2 and the minimum burnup requirements as defined in Table 5.6-1. (See Specification 5.6.1.c.7 for exceptions)

DESIGN FEATURES (continued)

CRITICALITY (continued)

5.6.1 c. (continued)

4. The 2x2 array of fuel assemblies that span the interface between Region 1 and Region 2 of the spent fuel pool storage racks shall comply with the storage pattern definitions of Figure 5.6-3 and the minimum burnup requirements as defined in Table 5.6-1. The allowed special arrangements in Region 2 as shown in Figure 5.6-2 shall not be placed adjacent to Region 1. (See Specification 5.6.1.c.7 for exceptions)
5. Fuel placed in the cask pit storage rack shall comply with the storage pattern definitions of Figure 5.6-4 and the minimum burnup requirements as defined in Table 5.6-1. (See Specification 5.6.1.c.7 for exceptions)
6. The same directional orientation for Metamic inserts is required for contiguous groups of 2x2 arrays where Metamic inserts are required.
7. Fresh or spent fuel in any allowed configuration may be replaced with non-fuel hardware, and fresh fuel in any allowed configuration may be replaced with a fuel rod storage basket containing fuel rod(s). Also, storage of Metamic inserts or control rods, without any fissile material, is acceptable in locations designated as completely water-filled cells.

- 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.

DRAINAGE

- 5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 56 feet.

CAPACITY

- 5.6.3 The spent fuel pool storage racks are designed and shall be maintained with a storage capacity limited to no more than 1491 fuel assemblies, and the cask pit storage rack is designed and shall be maintained with a storage capacity limited to no more than 225 fuel assemblies. The total Unit 2 spent fuel pool and cask pit storage capacity is limited to no more than 1716 fuel assemblies.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMITS

- 5.7.1 The components identified in Table 5.7-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7-1.

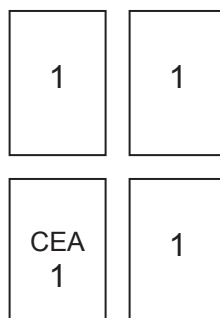
Pages 5-4C through 5-4F (Amendment 101) and page 5-4G (Amendment 135) have been deleted from the Technical Specifications.
The next page is 5-4h.

Allowable Storage Patterns (See Notes 1 and 2)

Pattern "A"
See Definition 1



Pattern "B"
See Definition 2



Pattern "C"
See Definition 3



DEFINITIONS:

1. Allowable pattern is Fresh Fuel or fuel of lower reactivity checkerboarded with completely water-filled cells. Diagram is for illustration only, where F represents Fresh Fuel and WH represents a completely water-filled cell.
2. Allowable pattern is placement of fuel assembly type 1 or fuel of lower reactivity in each 2x2 array location with at least one full-length full-strength CEA or equivalent (5 absorber rods) placed in any cell. Minimum burnup for fuel assembly type 1 is defined in Table 5.6-1 as a function of maximum initial planar average enrichment. Diagram is for illustration only.
3. Allowable pattern is placement of fuel assembly type 2 or fuel of lower reactivity in three of the 2x2 array locations in combination with one completely water-filled cell. Minimum burnup for fuel assembly type 2 is defined in Table 5.6-1 as a function of maximum initial planar average enrichment. Diagram is for illustration only.

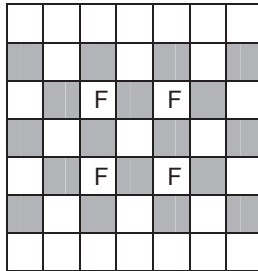
NOTES:

1. The storage arrangements of fuel within a rack module may contain more than one pattern. There are no interface limitations within Region 1 between rack modules or within racks; however, each assembly must meet the burnup requirements of each 2x2 array that it resides within.
2. Open cells within any pattern are acceptable.

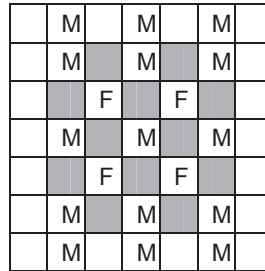
FIGURE 5.6-1
Allowable Region 1 Storage Patterns and Fuel Arrangements

ALLOWED SPECIAL ARRANGEMENTS (See Notes 1 and 2)
Fresh Fuel Assemblies in Region 2 Racks

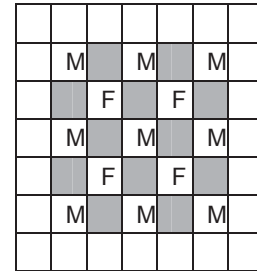
See Definition SR1



See Definition SR2



See Definition SR3



F = Fresh Fuel Assembly

= Empty Cell

M = Fresh Assembly with Metamic Insert

= Fresh Assembly per Definition

ALLOWABLE STORAGE PATTERNS (See Notes 1 and 2)

Pattern "D"
See Definition 1



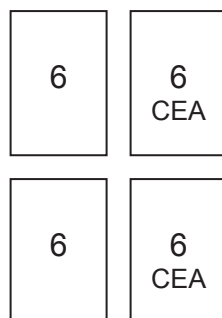
Pattern "E"
See Definition 2



Pattern "F"
See Definition 3



Pattern "G"
See Definition 4



Pattern "H"
See Definition 5



FIGURE 5.6-2 (Sheet 1 of 3)
Allowable Region 2 Storage Patterns and Fuel Arrangements

DEFINITIONS for Figure 5.6-2

1. Allowable pattern is fuel assembly type 3 or fuel of lower reactivity in three of the 2x2 array locations in combination with one completely water-filled cell. Minimum burnup for fuel assembly type 3 is defined in Table 5.6-1 as a function of maximum initial planar average enrichment. Diagram is for illustration only.
2. Allowable pattern is fuel assembly type 4 or fuel of lower reactivity in each of the 2x2 array locations with at least two Metamic inserts placed anywhere in the 2x2 array. Minimum burnup for fuel assembly type 4 is defined in Table 5.6-1 as a function of maximum initial planar average enrichment and cooling time. Diagram is for illustration only.
3. Allowable pattern is fuel assembly type 5 or fuel of lower reactivity in each of the 2x2 array locations with at least one Metamic insert placed anywhere in the 2x2 array. Minimum burnup for fuel assembly type 5 is defined in Table 5.6-1 as a function of maximum initial planar average enrichment and cooling time. Diagram is for illustration only.
4. Allowable pattern is fuel assembly type 6 or fuel of lower reactivity in each of the 2x2 array locations with at least two full-length, full strength 5 finger CEAs or equivalent placed anywhere in the 2x2 array. Minimum burnup for fuel assembly type 6 is defined in Table 5.6-1 as a function of maximum initial planar average enrichment and cooling time. Diagram is for illustration only.
5. Allowable pattern is fuel assembly type 7 in each of the 2x2 array locations with at least one full-length, full strength 5 finger CEA or equivalent placed anywhere in the 2x2 array. Minimum burnup for fuel assembly type 7 is defined in Table 5.6-1 as a function of maximum initial planar average enrichment and cooling time. Diagram is for illustration only.

SR1. Allowable pattern is up to four fresh fuel assemblies or fuel of lower reactivity placed in a 3x3 array in combination with Pattern "D" placed outside the 3x3 array. Fresh fuel shall be placed in the corners of the 3x3 array with completely water-filled cells placed face-adjacent on all sides. A fuel assembly of type 3 or fuel of lower reactivity shall be placed in the center of the 3x3 array. Minimum burnup for fuel assembly type 3 is defined in Table 5.6-1 as a function of maximum initial planar average enrichment. Diagram is for illustration only.

SR2. Allowable pattern is up to four fresh fuel assemblies or fuel of lower reactivity placed in a 3x3 array in combination with Pattern "E" placed outside the 3x3 array. Fresh fuel shall be placed in the corners of the 3x3 array with completely water-filled cells placed face-adjacent on all sides. A fuel assembly of type 4 or fuel of lower reactivity with a Metamic insert shall be placed in the center of the 3x3 array. Minimum burnup for fuel assembly type 4 is defined in Table 5.6-1 as a function of maximum initial planar average enrichment and cooling time. Diagram is for illustration only.

FIGURE 5.6-2 (Sheet 2 of 3)
Allowable Region 2 Storage Patterns and Fuel Arrangements

SR3. Allowable pattern is up to four fresh fuel assemblies or fuel of lower reactivity placed in a 3x3 array in combination with Pattern “F” placed outside the 3x3 array. Fresh fuel shall be placed in the corners of the 3x3 array with completely water-filled cells placed face-adjacent on all sides. A fuel assembly of type 5 or fuel of lower reactivity with a Metamic insert shall be placed in the center of the 3x3 array. Minimum burnup for fuel assembly type 5 is defined in Table 5.6-1 as a function of maximum initial planar average enrichment and cooling time. Diagram is for illustration only.

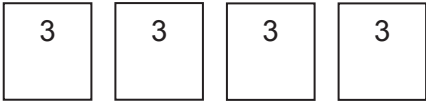
NOTES

1. The storage arrangements of fuel within a rack module may contain more than one pattern. There are no interface limitations within Region 2 between rack modules or within racks. However, each assembly must meet the burnup requirements of each 2x2 array or allowed special arrangement that it resides within.
2. Open cells within any pattern are acceptable.

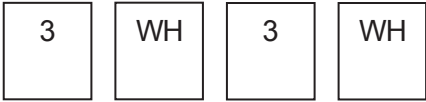
FIGURE 5.6-2 (Sheet 3 of 3)
Allowable Region 2 Storage Patterns and Fuel Arrangements

Allowed Region 2 to Region 1 Fuel Alignments
(See Note 1)

Interface of Region 2
Pattern “D” with Region 1
See Definition 1



Region 2

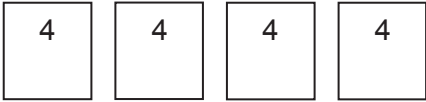


Region 1

Interface of Region 2
Pattern “E” with Region 1
See Definition 1

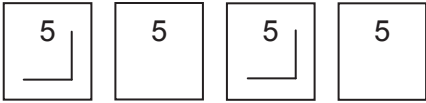


Region 2



Region 1

Interface of Region 2
Pattern “F” with Region 1
See Definition 1



Region 2

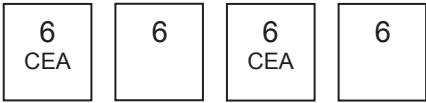


Region 1

Interface of Region 2
Pattern “G” with Region 1
See Definition 1

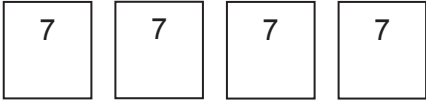


Region 2

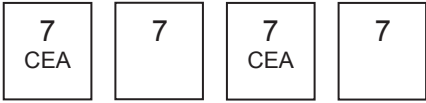


Region 1

Interface of Region 2
Pattern “H” with Region 1
See Definition 1



Region 2



Region 1

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

DEFINITION:

1. Each 2x2 array that spans Region 1 and Region 2 shall match one of the Region 2 allowable storage patterns as defined by Specification 5.6.1.c.3. Any required Metamic inserts must be placed into the fuel assemblies in Region 2. Locations of water-filled cells or CEAs may be in either Region 1 or Region 2. For interface assemblies, the requirements of Specifications 5.6.1.c.2 and Specification 5.6.1.c.3 shall be followed within Region 1 and Region 2, respectively. The Diagrams are for illustration only.

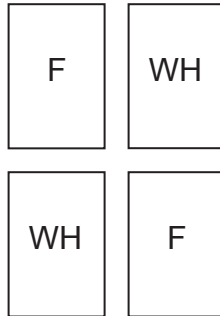
NOTES:

1. Open cells within any pattern are acceptable.

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

Allowable Storage Patterns
(See Notes 1 and 2)

Pattern "A"
See Definition 1



Pattern "B"
See Definition 2



DEFINITIONS:

1. Allowable pattern is Fresh Fuel or lower reactivity fuel checkerboarded with completely water-filled cells. Diagram is for illustration only, where F represents Fresh Fuel and WH represents a completely water-filled cell.
2. Allowable pattern is placement of fuel assembly type 8 or fuel of lower reactivity in three of the 2x2 array locations in combination with one completely water-filled cell in any location. Minimum burnup for fuel assembly type 8 is defined in Table 5.6-1 as a function of maximum initial planar average enrichment. Diagram is for illustration only.

NOTES:

1. The storage arrangements of fuel within a rack module may contain more than one pattern. There are no interface limitations within the cask pit storage rack, however, each assembly must meet the burnup requirements of each 2x2 array that it resides within.
2. Open cells within any pattern are acceptable.

FIGURE 5.6-4
Allowable Cask Pit Storage Rack Patterns

TABLE 5.6-1
Minimum Burnup Coefficients

Fuel Type	Cooling Time (Years)	Coefficients		
		A	B	C
1	0	-33.4237	25.6742	-1.6478
2	0	-25.3198	14.3200	-0.4042
3	0	-23.4150	16.2050	-0.5500
4	0	-33.6414	25.0670	-1.5551
	2.5	-32.3764	23.9988	-1.5075
	5	-30.9234	22.9382	-1.4372
	10	-28.4951	21.1511	-1.3029
	15	-27.2024	20.2802	-1.2479
	20	-25.2009	18.6218	-1.0364
5	0	-24.8402	23.5991	-1.2082
	2.5	-23.0170	21.6493	-1.0298
	5	-21.9293	20.6257	-0.9730
	10	-20.0813	19.0808	-0.9022
	15	-19.5503	18.5429	-0.9129
	20	-18.7485	17.7308	-0.8390
6	0	-32.4900	25.3077	-1.5518
	2.5	-31.1598	23.9185	-1.4435
	5	-29.2169	22.5424	-1.3274
	10	-26.8886	20.6662	-1.1425
	15	-25.5703	19.7629	-1.1129
	20	-24.5754	18.9056	-1.0147
7	0	-24.6989	24.1660	-1.2578
	2.5	-23.0399	22.3047	-1.0965
	5	-21.3290	20.7413	-0.9613
	10	-20.0836	19.4780	-0.8949
	15	-19.2480	18.5880	-0.8685
	20	-18.6424	18.1241	-0.8950
8	0	-47.5000	12.5000	-0.0000

NOTES:

1. To qualify in a fuel type, the calculated burnup of a fuel assembly must exceed the “minimum burnup” determined for the “cooling time” and “maximum initial planar enrichment” of the fuel assembly. The “minimum burnup” for any fuel type is determined from the following polynomial function:

$$BU = A + B \cdot E + C \cdot E^2, \text{ where:}$$

BU = Minimum Burnup (GWD/MTU)

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

A, B, C = Coefficients for each fuel type

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

ADMINISTRATIVE CONTROLS

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 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.

ADMINISTRATIVE CONTROLS

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 |

ADMINISTRATIVE CONTROLS (Continued)

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

ADMINISTRATIVE CONTROLS (continued)

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 |

ADMINISTRATIVE CONTROLS (continued)

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.

ADMINISTRATIVE CONTROLS (continued)

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, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses", April 1999.
66. WCAP-7908-A, "FACTRAN-A FORTRAN IV Code for Thermal Transients in a UO₂ Fuel Rod", December 1989.
67. WCAP-7979-P-A, "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.