St. Lucie Unit 1
Docket No. 50-335
Proposed License Amendment
RCS Minimum Flow and TS Update

ATTACHMENT 3 to FPL Letter L-98-283

ST. LUCIE UNIT 1 MARKED-UP TECHNICAL SPECIFICATION PAGES

Page 1-3

Page 2-2

INSERT - A (Revised Figure 2.1-1)

Page 2-4

Page B 2-5

Page 3/4 2-14

Page 5-5

Page 6-19

Page 6-19a

INSERT - B (Revised List of Analytical Methods for TS 6.9.1.11.b)

ICRP-30, Supplement to Part 1, pages 192-212, Tables entitled, "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity (Sv/Bq)."

DOSE EQUIVALENT I-131

1.10 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (µCi/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in ~~Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."~~

REPLACE

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

1.11 $\bar{E}$ 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.

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.

FREQUENCY NOTATION

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

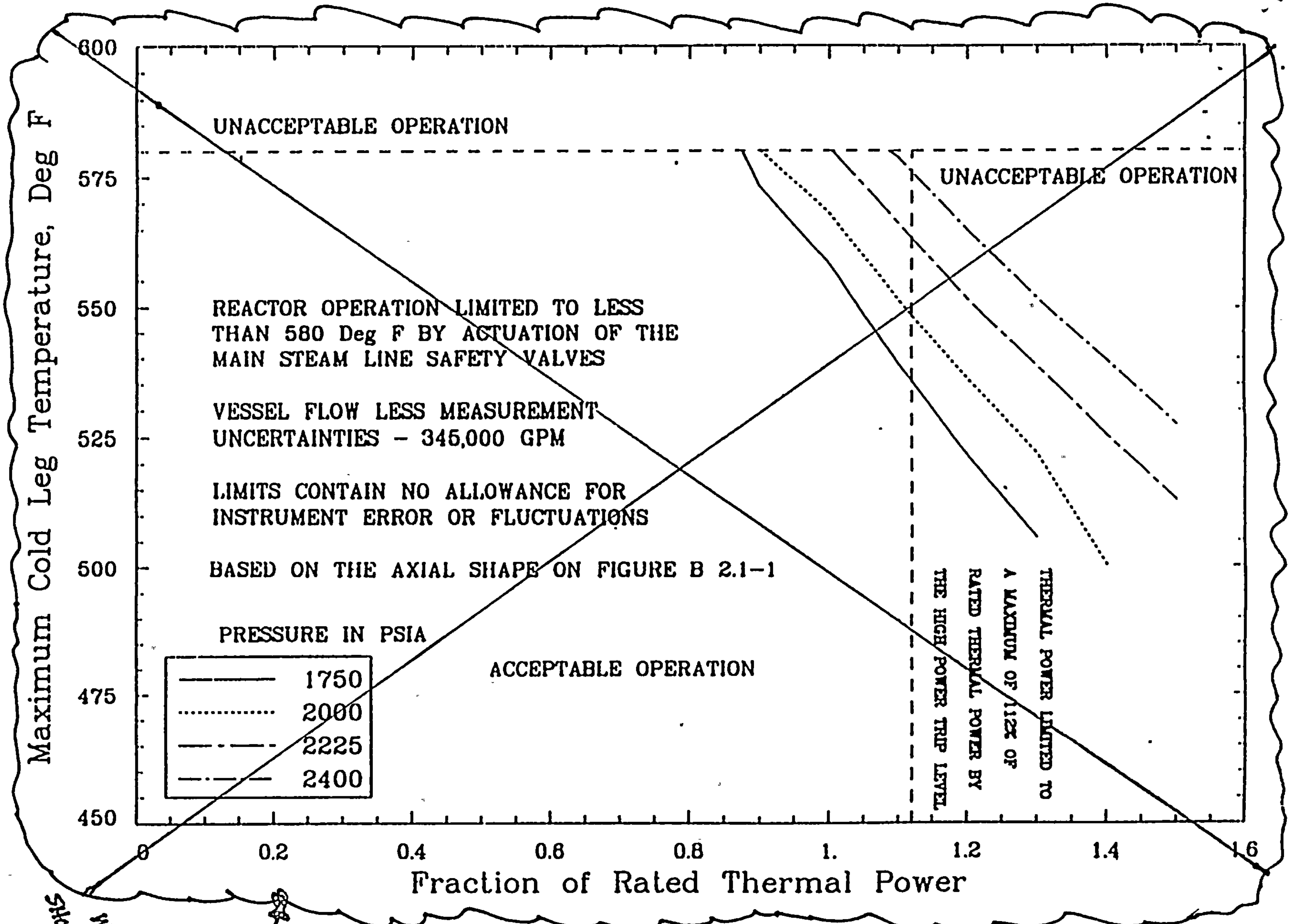
GASEOUS RADWASTE TREATMENT SYSTEM

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.

[illegible]

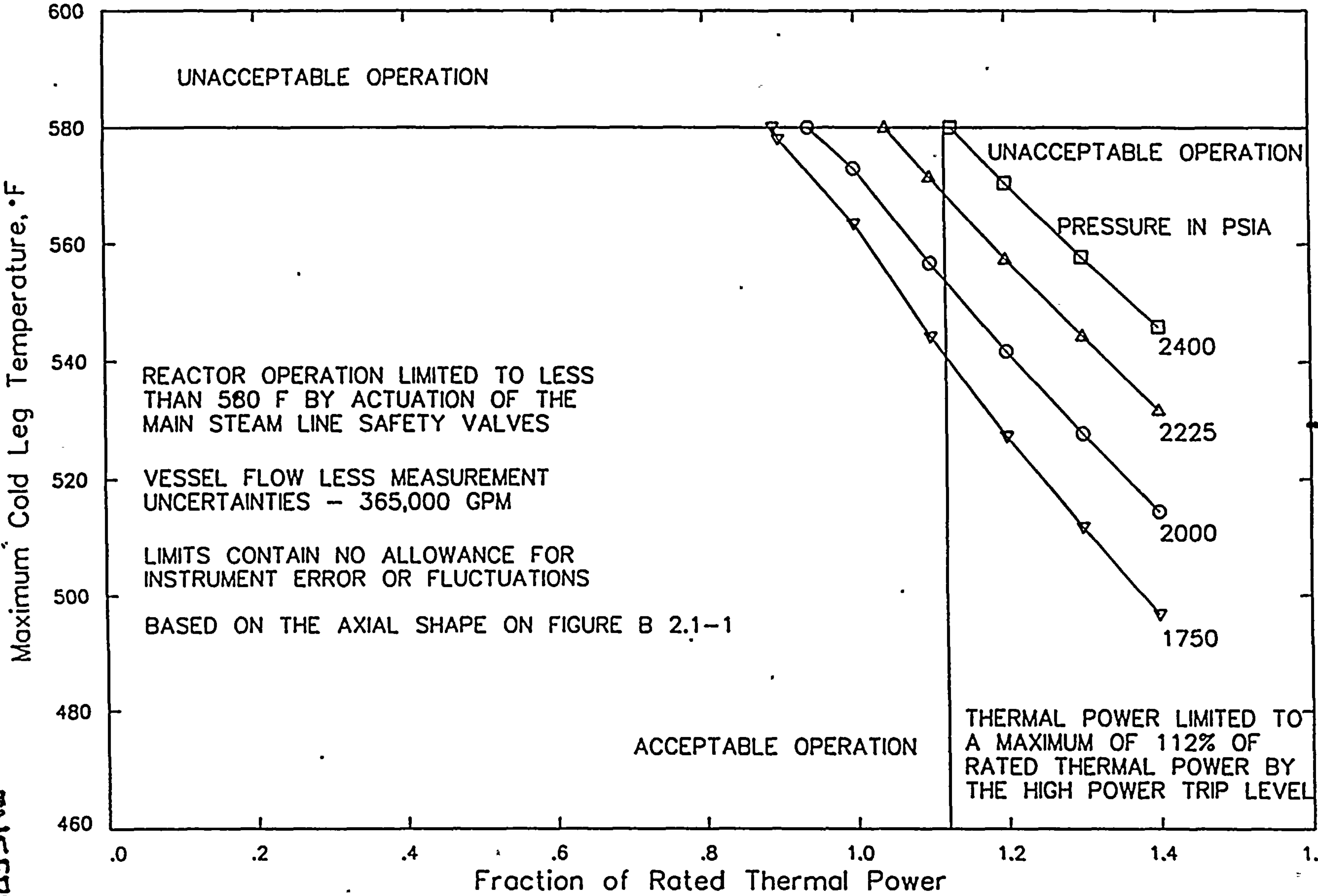
[illegible]

[illegible]



REPLACE WITH INSERT-A SHOWN ON NEXT PAGE

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



INSERT - A

INSERT - A

INSERT - A

INSERT - A

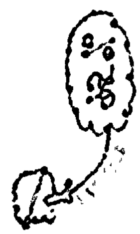
TABLE 2.2-1

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Power Level - High (1)		
Four Reactor Coolant Pumps Operating	≤ 9.61% above THERMAL POWER, with a minimum setpoint of 15% of RATED THERMAL POWER, and a maximum of ≤ 107.0% of RATED THERMAL POWER.	≤ 9.61% above THERMAL POWER, and a minimum setpoint of 15% of RATED THERMAL POWER and a maximum of ≤ 107.0% of RATED THERMAL POWER.
3. Reactor Coolant Flow - Low (1)	REPLACE 95%	REPLACE 95%
Four Reactor Coolant Pumps Operating	≥ ~~93%~~ of design reactor coolant flow with 4 pumps operating*	≥ 93% of design reactor coolant flow with 4 pumps operating*
4. Pressurizer Pressure - High	≤ 2400 psia	≤ 2400 psia
5. Containment Pressure - High	≤ 3.3 psig	≤ 3.3 psig
6. Steam Generator Pressure - Low (2)	≥ 600 psia	≥ 600 psia
7. Steam Generator Water Level -Low	≥ 20.5% Water Level - each steam generator	≥ 19.5% Water Level - each steam generator
8. Local Power Density - High (3)	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-1 and 2.2-2	Trip set point adjusted to not exceed the limit lines of Figures 2.2-1 and 2.2-2.

*Design reactor coolant flow with 4 pumps operating is ~~345,000~~ gpm.

REPLACE 365,000



2.2 LIMITING SAFETY SYSTEM SETTINGS

BASES

Reactor Coolant Flow-Low (Continued)

relationship between steam generator differential pressure, core inlet temperature, instrument errors and response times. When the calculated RCS flow falls below the trip setpoint an automatic reactor trip signal is initiated. The trip setpoint and allowable values ensure that for a degradation of RCS flow resulting from expected transients, a reactor trip occurs to prevent violation of local power density or DNBR safety limits.

Pressurizer Pressure-High

The Pressurizer Pressure-High trip, backed up by the pressurizer code safety valves and main steam line safety valves, provides reactor coolant system protection against overpressurization in the event of loss of load without reactor trip. This trip's setpoint is 100 psi below the nominal lift setting (2500 psia) of the pressurizer code safety valves and its concurrent operation with the power-operated relief valves avoids the undesirable operation of the pressurizer code safety valves.

Containment Pressure-High

The Containment Pressure High trip provides assurance that a reactor trip is initiated concurrently with a safety injection.

Steam Generator Pressure-Low

The Steam Generator Pressure-Low trip provides protection against an excessive rate of heat extraction from the steam generators and subsequent cooldown of the reactor coolant. The setting of 600 psia is sufficiently below the full-load operating point ~~of 800 psig~~ so as not

DELETE

TABLE 3.2-1

DNB MARGIN

LIMITS

Parameter	Four Reactor Coolant Pumps Operating
Cold Leg Temperature	≤ 549°F
Pressurizer Pressure	≥ 2225 psia*
Reactor Coolant Flow Rate	≥ ~~345,000~~ gpm
AXIAL SHAPE INDEX	COLR Figure 3.2-4

REPLACE 365,000

* 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.

CONTROL ELEMENT ASSEMBLIES

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

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

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

a. In accordance with the code requirements specified in Section 5.2 of the FSAR with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

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

VOLUME

5.4.2 The total water and steam volume of the reactor coolant system is 11,100 ± 180 cubic feet at a nominal T_{avg} of 567°F, when not accounting for steam generator tube plugging.

5.5 EMERGENCY CORE COOLING SYSTEMS

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

5.6 FUEL STORAGE

k_{eff} less than or equal to 0.95 if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1 of the Updated Final Safety Analysis Report

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 or equal to 0.95 when flooded with unborated water, which includes a conservative allowance of 0.0065 Δk for uncertainties~~.

REPLACE

INSERT - B (2 pages)

3. XN-75-27(A) and Supplements 1 through 5, [also issued as XN-NF-75-27(A)], "Exxon Nuclear Neutronic(s) Design Methods for Pressurized Water Reactors," Exxon Nuclear Company, Inc. / Advanced Nuclear Fuels Corporation, Report and Supplement 1 dated April 1977, Supplement 2 dated December 1980, Supplement 3 dated September 1981 (P), Supplement 4 dated December 1986 (P), and Supplement 5 dated February 1987 (P)

4. ANF-84-73(P)(A) Revision 5, Appendix B, & Supplements 1 and 2, "Advanced Nuclear Fuels Methodology for Pressurized Water Reactors: Analysis of Chapter 15 Events," Advanced Nuclear Fuels Corporation, October 1990

5. XN-NF-82-21(P)(A) Revision 1, "Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations," Exxon Nuclear Company, Inc., September 1983

6. a) ANF-84-93(P)(A) and Supplement 1, [also issued as XN-NF-84-93(P)(A)], "Steamline Break Methodology for PWRs," Advanced Nuclear Fuels Corporation, March 1989

   b) EMF-84-093(P) Revision 1, "Steam Line Break Methodology for PWRs," Siemens Power Corporation, June 1998 (This document is a Revision to ANF-84-93)

7. XN-75-32(P)(A) Supplements 1 through 4, "Computational Procedure for Evaluating Fuel Rod Bowing," Exxon Nuclear Company, Inc., October 1983

8. Siemens Power Corporation Small Break LOCA methodology as defined by:

   a) XN-NF-82-49(P)(A) Revision 1, "Exxon Nuclear Company Evaluation Model EXEM PWR Small Break Model," Advanced Nuclear Fuels Corporation, April 1989

   b) XN-NF-82-49(P)(A) Revision 1 Supplement 1, "Exxon Nuclear Company Evaluation Model Revised EXEM PWR Small Break Model," Siemens Power Corporation, December 1994

9. XN-NF-78-44(NP)(A), "A Generic Analysis of the Control Rod Ejection Transient for Pressurized Water Reactors," Exxon Nuclear Company, Inc., October 1983

10. XN-NF-621(P)(A) Revision 1, "Exxon Nuclear DNB Correlation for PWR Fuel Designs," Exxon Nuclear Company, Inc., September 1983

11. EXEM PWR Large Break LOCA Evaluation Model as defined by:

    a) 1. XN-NF-82-20(P)(A) Revision 1 Supplement 2, "Exxon Nuclear Company Evaluation Model EXEM/PWR ECCS Model Updates," Exxon Nuclear Company, Inc., February 1985

(CONT'D NEXT PAGE)

(CONT'D NEXT PAGE)

INSERT – B (Pg 1 of 2)

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 3.1.1.4	Moderator Temperature Coefficient
Specification 3.1.3.1	Full Length CEA Position - Misalignment > 15 inches
Specification 3.1.3.6	Regulating CEA Insertion Limits
Specification 3.2.1	Linear Heat Rate
Specification 3.2.3	Total Integrated Radial Peaking Factor - F_r^T
Specification 3.2.5	DNB Parameters
Specification 3.9.1	Refueling Operations - Boron Concentration

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

REPLACE WITH INSERT-B

3. ~~XN-75-27(A), Rev. 0 and Supplement 1 through 5, "Exxon Nuclear Neutronics Design Methods for Pressurized Water Reactors," Exxon Nuclear Company, Rev. 0 dated June 1975, Supplement 1 dated September 1976, Supplement 2 dated December 1980, Supplement 3 dated September 1981, Supplement 4 dated December 1986, Supplement 5 dated February 1987.~~

4. ~~ANF-84-73(P), Rev. 3, "Advanced Nuclear Fuels Methodology for Pressurized Water Reactors: Analysis of Chapter 15 Events," Advanced Nuclear Fuel Corporation, dated May 1988.~~

5. ~~XN-NF-82-21(A), Rev. 1, "Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations," Exxon Nuclear Company, dated September 1983.~~

6. ~~ANF-84-93(A), Rev. 0 and Supplement 1, "Steamline Break Methodology for PWR's," Advanced Nuclear Fuels Corporation, Rev. 0 dated March 1989, Supplement 1 dated March 1989.~~

CORE OPERATING LIMITS REPORT (continued)

DELETE

7. ~~XN-75-32(A), Supplements 1, 2, 3, and 4, "Computational Procedure for Evaluating Fuel Rod Bowing," Exxon Nuclear Company, dated October 1983.~~

8. ~~XN-NF-82-49(A), Rev. 1 and Supplement 1, "Exxon Nuclear Company Evaluation Model EXEM PWR Small Break Model," Advanced Nuclear Fuels Corporation, Rev. 1 dated April 1989, Supplement 1 dated December 1994.~~

9. ~~XN-NF-78-44(A), "A Generic Analysis of the Control Rod Ejection Transient for Pressurized Water Reactors," Exxon Nuclear Company, dated October 1983.~~

10. ~~XN-NF-621(A), Rev. 1, "Exxon Nuclear DNB Correlation of PWR Fuel Design," Exxon Nuclear Company, dated September 1983.~~

11. ~~EXEM PWR Large Break LOCA Evaluation Model as defined by:~~

    a) ~~XN-NF-82-20(A), Rev. 1 and Supplements 1 through 4, "Exxon Nuclear Company Evaluation Model EXEM/PWR ECCS Model Updates," Exxon Nuclear Company, all dated January 1990.~~

    b) ~~XN-NF-82-07(A), Rev. 1, "Exxon Nuclear Company ECCS Cladding Swelling and Rupture Model," Exxon Nuclear Company, dated November 1982.~~

    c) ~~XN-NF-81-58(A), Rev. 2 and Supplements 1 through 4, "RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model," Exxon Nuclear Company, Rev. 2 and Supplement 1 and 2 dated March 1984, Supplements 3 and 4 dated June 1990.~~

    d) ~~XN-NF-85-16(A), Volume 1 through Supplement 3; Volume 2, Rev. 1 and Supplement 1, "PWR 17x17 Fuel Cooling Tests Program," Exxon Nuclear Company, all dated February 1990.~~

    e) ~~XN-NF-85-105(A), Rev. 0 and Supplement 1, "Scaling of FCTF Based Reflood Heat Transfer Correlation for Other Bundle Designs," Exxon Nuclear Company, all dated January 1990.~~

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 to the NRC.

SPECIAL REPORTS

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

INSERT - B (Continued)

2. XN-NF-82-20(P)(A) Revision 1 and Supplements 1, 3 and 4, "Exxon Nuclear Company Evaluation Model EXEM/PWR ECCS Model Updates," Advanced Nuclear Fuels Corporation, January 1990

b) XN-NF-82-07(P)(A) Revision 1, "Exxon Nuclear Company ECCS Cladding Swelling and Rupture Model," Exxon Nuclear Company, Inc., November 1982

c) 1. XN-NF-81-58(P)(A) Revision 2, and Supplements 1 and 2, "RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model," Exxon Nuclear Company, Inc., March 1984

2. ANF-81-58(P)(A) Revision 2 Supplement 3, and Supplement 4, "RODEX2 Fuel Rod Thermal Mechanical Response Evaluation Model," Advanced Nuclear Fuels Corporation, June 1990

d) XN-NF-85-16(P)(A) Volume 1, and Supplements 1, 2 and 3; Volume 2, Revision 1 and Supplement 1, "PWR 17x17 Fuel Cooling Test Program," Advanced Nuclear Fuels Corporation, February 1990

e) XN-NF-85-105(P)(A) and Supplement 1, "Scaling of FCTF Based Reflood Heat Transfer Correlation for Other Bundle Designs," Advanced Nuclear Fuels Corporation, January 1990

f) EMF-2087(P) Revision 0, "SEM/PWR-98: ECCS Evaluation Model for PWR LBLOCA Applications," Siemens Power Corporation, August 1998

12. XN-NF-82-06(P)(A) Revision 1, and Supplements 2, 4 and 5, "Qualification of Exxon Nuclear Fuel for Extended Burnup," Exxon Nuclear Company, Inc., October 1986

13. ANF-88-133(P)(A) and Supplement 1, "Qualification of Advanced Nuclear Fuels' PWR Design Methodology for Rod Burnups of 62 GWd/MTU," Advanced Nuclear Fuels Corporation, December 1991

14. XN-NF-85-92(P)(A), "Exxon Nuclear Uranium Dioxide/Gadolinia Irradiation Examination and Thermal Conductivity Results," Exxon Nuclear Company, Inc., November 1986

15. ANF-89-151(P)(A), "ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events," Advanced Nuclear Fuels Corporation, May 1992

16. XN-NF-507(P)(A), Supplements 1 and 2, "ENC Setpoint Methodology for C. E. Reactors: Statistical Setpoint Methodology," Exxon Nuclear Company, Inc., September 1986

INSERT-B. (CONT'D).

INSERT-B (cont'd):