ENCLOSURE C TO PLA-4527

TECHNICAL SPECIFICATION MARK-UPs

1.0 DEFINITIONS

The following terms are defined so that uniform interpretation of these specifications may be achieved. The defined terms appear in capitalized type and shall be applicable throughout these Technical Specifications.

ACTION

1.1 ACTION shall be that part of a Specification which prescribes remedial measures required under designated conditions.

INSERT 1

~~AVERAGE EXPOSURE~~

~~1.2 The AVERAGE BUNDLE EXPOSURE shall be equal to the sum of the axially averaged exposure of all the fuel rods in the specified bundle divided by the number of fuel rods in the fuel bundle.~~

~~The AVERAGE PLANAR EXPOSURE shall be applicable to a specific planar height and is equal to the sum of the exposure of all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle.~~

AVERAGE PLANAR LINEAR HEAT GENERATION RATE

1.3 The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) shall be applicable to a specific planar height and is equal to the sum of the LINEAR HEAT GENERATION RATES for all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle at the height.

CHANNEL CALIBRATION

1.4 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel including the sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an in-place qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel. The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is calibrated.

CHANNEL CHECK

1.5 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

CHANNEL FUNCTIONAL TEST

1.6 A CHANNEL FUNCTIONAL TEST shall be:

a. Analog channels - the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY including alarm and/or trip functions and channel failure trips.

b. Bistable channels - the injection of a simulated signal into the sensor to verify OPERABILITY including alarm and/or trip functions.

The CHANNEL FUNCTIONAL TEST may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is tested.

INSERT 1

AVERAGE EXPOSURE

1.2 The AVERAGE BUNDLE EXPOSURE shall be equal to the total energy produced by the bundle divided by the total initial weight of uranium in the fuel bundle.

The AVERAGE PLANAR EXPOSURE at a specified height shall be equal to the total energy produced per unit length at the specified height divided by the total initial weight of uranium per unit length at that height.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

THERMAL POWER, Low Pressure or Low Flow

2.1.1 THERMAL POWER shall not exceed 25% of RATED THERMAL POWER with the reactor vessel steam dome pressure less than 785 psig or core flow less than 10 million lbm/hr.

APPLICABILITY: OPERATIONAL CONDITIONS 1 AND 2.

ACTION:

With THERMAL POWER exceeding 25% of RATED THERMAL POWER and the reactor vessel steam dome pressure less than 785 psig or core flow less than 10 million lbm/hr., be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.1.

THERMAL POWER, High Pressure and High Flow

1.09

2.1.2 The MINIMUM CRITICAL POWER RATIO (MCPR) shall not be less than ~~1.08~~* with the reactor vessel steam dome pressure greater than 785 psig and core flow greater than 10 million lbm/hr.

APPLICABILITY: OPERATIONAL CONDITIONS 1 AND 2.

ACTION: 1.09

With MCPR less than ~~1.08~~* and the reactor vessel steam dome pressure greater than 785 psig and core flow greater than 10 million lbm/hr., be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.1.

REACTOR COOLANT SYSTEM PRESSURE

2.1.3 The reactor coolant system pressure, as measured in the reactor vessel steam dome, shall not exceed 1325 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3 AND 4.

ACTION:

With the reactor coolant system pressure, as measured in the reactor vessel steam dome, above 1325 psig, be in at least HOT SHUTDOWN with reactor coolant system pressure less than or equal to 1325 psig within 2 hours and comply with the requirements of Specification 6.7.1.

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* See Specification 3.4.1.1.2.a for single loop operation requirement.

NRC IMNS DIVISION TEL:301-504-5269 Oct 15 '96 12:51 No.005 P.08

2.1 SAFETY LIMITS

BASES

2.0 INTRODUCTION

The fuel cladding, reactor pressure vessel and primary system piping are the principal barriers to the release of radioactive materials to the environs. Safety Limits are established to protect the integrity of these barriers during normal plant operations and anticipated transients. The fuel cladding integrity Safety Limit is set such that no fuel damage is expected to occur if the limit is not violated. Because fuel damage is not directly observable, a step-back approach is used to establish a Safety Limit such that the MCPR is not less than the limit specified in Specification 2.1.2 for SPC fuel. MCPR greater than the specified limit represents a conservative margin relative to the conditions required to maintain fuel cladding integrity. The fuel cladding is one of the physical barriers which separate the radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions and the Limiting Safety System Settings. While fission product migration from cladding perforation is just as measurable as that from use related cracking, the thermally caused cladding perforations signal a threshold beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding Safety Limit is defined with a margin to the conditions which would produce onset of transition boiling, MCPR of 1.0. These conditions represent a significant departure from the condition intended by design for planned operation. The MCPR fuel cladding integrity Safety Limit assures that during normal operation and during anticipated operational occurrences, at least 99.9% of the fuel rods in the core do not experience transition boiling (ref. ANF-524(P)(A) Revision 2).

2.1.1 THERMAL POWER, Low Pressure or Low Flow

The use of the ANFB correlation is valid for critical power calculations at pressures greater than 585 psig and bundle mass fluxes greater than 0.1×10^6 lbs/hr-ft^2. For operation at low pressures or low flows, the fuel cladding integrity Safety Limit is established by a limiting condition on core THERMAL POWER with the following basis:

Provided that the water level in the vessel downcomer is maintained above the top of the active fuel, natural circulation is sufficient to assure a minimum bundle flow for all fuel assemblies which have a relatively high power and potentially can approach a critical heat flux condition. For the SPC 9x9-2 fuel design, the minimum bundle flow is greater than 30,000 lbs/hr. ~~For the SPC 9x9-2 design, the coolant minimum flow and maximum flow area is such that the mass flux is always greater than 0.25×10^6 lbs/hr-ft^2. Full scale critical power tests taken at pressures down to 14.7 psia indicate that the fuel assembly critical power at 0.25×10^6 lbs/hr-ft^2 is 3.35 Mwt or greater.~~ At 25% thermal power a bundle power of 3.35 Mwt corresponds to a bundle radial peaking factor of approximately 3.0 which is significantly higher than the expected peaking factor. Thus, a THERMAL POWER limit of 25% of RATED THERMAL POWER for reactor pressures below 785 psig is conservative.

INSERT 2

INSERT 2

For the SPC ATRIUM-10 design, the minimum bundle flow is greater than 28,000 lbs/hr. For both the SPC 9x9-2 and ATRIUM-10 fuel designs, the coolant minimum flow and maximum flow area is such that the mass flux is always greater than 0.25×10^6 lbs/hr-ft^2. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at 0.25×10^6 lbs/hr-ft^2 is 3.35 Mwt or greater.

2.1.2 THERMAL POWER, High Pressure and High Flow

Onset of transition boiling results in a decrease in heat transfer from the clad and, therefore, elevated clad temperature and the possibility of clad failure. However, the existence of critical power, or boiling transition, is not a directly observable parameter in an operating reactor. Therefore, the margin to boiling transition is calculated from plant operating parameters such as core power, core flow, feedwater temperature, and core power distribution. The margin for each fuel assembly is characterized by the critical power ratio (CPR), which is the ratio of the bundle power which would produce onset of transition boiling divided by the actual bundle power. The minimum value of this ratio for any bundle in the core is the minimum critical power ratio (MCPR).

The Safety Limit MCPR assures sufficient conservatism in the operating MCPR limit that in the event of an anticipated operational occurrence from the limiting condition for operation, at least 99.9% of the fuel rods in the core would be expected to avoid boiling transition. The margin between calculated boiling transition (MCPR = 1.00) and the Safety Limit MCPR is based on a detailed statistical procedure which considers the uncertainties in monitoring the core operating state. One specific uncertainty included in the safety limit is the uncertainty inherent in the ~~XN-3~~ critical power correlation. ANF-524 (P)(A), Revision 2 and PL-NF-90-001(A) and Supplement 2 describe the methodologies used in determining the Safety Limit MCPR.

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The ANFB critical power correlation is based on a significant body of practical test data, providing a high degree of assurance that the critical power as evaluated by the correlation is within a small percentage of the actual critical power being estimated. As long as the core pressure and flow are within the range of validity of the correlation (refer to Section B 2.1.1), the assumed reactor conditions used in defining the safety limit introduce conservatism into the limit because bounding high radial power factors and bounding flat local peaking distributions are used to estimate the number of rods in boiling transition. These conservatisms and the inherent accuracy of the ANFB correlation provide a reasonable degree of assurance that during sustained operation at the Safety Limit MCPR there would be no transition boiling in the core. If boiling transition were to occur, there is reason to believe that the integrity of the fuel would not necessarily be compromised. Significant test data accumulated by the U.S. Nuclear Regulatory Commission and private organizations indicate that the use of a boiling transition limitation to protect against cladding failure is a very conservative approach. Much of the data indicates that LWR fuel can survive for an extended period of time in an environment of boiling transition.

SPC fuel is monitored using the ANFB Critical Power Correlation. The effects of channel bow on MCPR are explicitly included in the calculation of the ANFB MCPR Safety Limit. Explicit treatment of channel bow in the ANFB MCPR Safety Limit addresses the concerns of NRC Bulletin No. 90-02 entitled "Loss of Thermal Margin Caused by Channel Box Bow."

3.4.1.1.2 One reactor coolant recirculation loop shall be in operation with the pump speed ≤ 80% of the rated pump speed and the reactor at a THERMAL POWER/core flow condition outside of Regions I and II of Figure 3.4.1.1.1-1, and

a. the following revised specification limits shall be followed:

1. Specification 2.1.2: the MCPR Safety Limit shall be increased to ~~1.09~~. 1.12
2. Table 2.2.1-1: the APRM Flow-Biased Scram Trip Setpoints shall be as follows:

Trip Setpoint	Allowable Value
≤ 0.58W + 54%	≤ 0.58W + 57%

3. Specification 3.2.2: the APRM Setpoints shall be as follows:

Trip Setpoint	Allowable Value
S ≤ (0.58W + 54%) T	S ≤ (0.58W + 57%) T
S_{RB} ≤ (0.58W + 45%) T	S_{RB} ≤ (0.58W + 48%) T

4. Specification 3.2.3: The MINIMUM CRITICAL POWER RATIO (MCPR) shall be greater than or equal to the applicable Single Loop Operation MCPR limit as specified in the CORE OPERATING LIMITS REPORT.
5. Specification 3.2.4: The LINEAR HEAT GENERATION RATE (LHGR) shall be less than or equal to the applicable Single Loop Operation LHGR limit as specified in the CORE OPERATING LIMITS REPORT.
6. Table 3.3.6-2: the RBM/APRM Control Rod Block Setpoints shall be as follows:

	Trip Setpoint	Allowable Value
a. RBM - Upscale	≤ 0.63w + 35%	≤ 0.63W + 37%
	Trip Setpoint	Allowable Value
b. APRM-Flow Biased	≤ 0.58W + 45%	≤ 0.58W + 48%

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*, except during two loop operation.#

ACTION:

a. In OPERATIONAL CONDITION 1:
   1. With
      a) no reactor coolant system recirculation loops in operation, or
      b) Region I of Figure 3.4.1.1.1-1 entered, or
      c) Region II of Figure 3.4.1.1.1-1 entered and core thermal hydraulic instability occurring as evidenced by:

or water channels

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 764 fuel assemblies. Each assembly consists of a matrix of Zircaloy clad fuel rods with an initial composition of non-enriched or slightly enriched uranium dioxide as fuel material and water rods. Limited substitutions of Zirconium alloy filler rods for fuel rods, in accordance with NRC-approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff-approved codes and methods, and shown by test or analyses to comply with all fuel safety design bases. A limited number of lead use assemblies that have not completed representative testing may be placed in non-limiting core regions. ~~Each fuel rod shall have a nominal active fuel length of 150 inches.~~ Reload fuel shall have a maximum average enrichment of ~~4.0~~ weight percent U-235.

4.5 lattice

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 185 cruciform shaped control rod assemblies. The control material shall be boron carbide powder (B_4C), and/or Hafnium metal. The control rod shall have a nominal axial absorber length of 143 inches. Control rod assemblies shall be limited to those control rod designs approved by the NRC for use in BWRs.

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:

1. 1250 psig on the suction side of the recirculation pumps.

2. 1500 psig from the recirculation pump discharge to the jet pumps.

c. For a temperature of 575°F.

VOLUME

5.4.2 The total water and steam volume of the reactor vessel and recirculation system is approximately 22,400 cubic feet at a nominal T_{ave} of 532°F.

CORE OPERATING LIMITS REPORT (Continued)

14. ANF-1125(P)(A) and ANF-1125(P)(A), Supplement 1, "ANFB Critical Power Correlation," April 1990.

15. NEDC-32071P, "SAFER/GESTR-LOCA Loss of Coolant Accident Analysis," GE Nuclear Energy, May 1992.

16. NE-092-001A, Revision 1, "Licensing Topical Report for Power Uprate With Increased Core Flow," Pennsylvania Power & Light Company, December 1992.

17. NRC SER on PP&L Power Uprate LTR (November 30, 1993).

18. PL-NF-90-001, Supplement 1-A, "Application of Reactor Analysis Methods for BWR Design and Analysis: Loss of Feedwater Heating Changes and Use of RETRAN MOD 5.1," September 1994.

19. PL-NF-94-005-P-A, "Technical Basis for SPC 9x9-2 Extended Fuel Exposure at Susquehanna SES," January 1995.

20. NEDE-24011-P-A-10, "General Electric Standard Application for Reactor Fuel," February 1991.

21. PL-NF-90-001, Supplement 2 -A, "Application of Reactor Analysis Methods to BWR Design and Analysis: CASMO-3G Code and ANFB Critical Power Correlation."

INSERT 3

6.9.3.3 The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, transient analysis limits and accident analysis limits) of the safety analysis are met.

6.10 RECORD RETENTION

In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

6.10.1 The following records shall be retained for at least 5 years:

a. Records and logs of unit operation covering time interval at each power level.

INSERT 3

22. ANF-89-98(P)(A) Revision 1 and Revision 1 Supplement 1, "Generic Mechanical Design Criteria for BWR Fuel Designs," Advanced Nuclear Fuels Corporation, May 1995.

23. XN-NF-81-58 (P)(A) Supplements 1 and 2 Revision 2, "RODEX 2 Fuel Rod Thermal-Mechanical Response Evaluation Model," May 1986.

24. XN-NF-85-74(P)(A), "RODEX 2A (BWR) Fuel Rod Thermal-Mechanical Response Evaluation Model," August 1986.

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

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

27. ANF-90-082(P)(A) Revision 1 and Revision 1 Supplement 1, "Application of ANF Design Methodology for Fuel Assembly Reconstitution," May 1995.

28. ANF-91-048(P)(A), "Advanced Nuclear Fuels Corporation Methodology for Boiling Water Reactors EXEM BWR Evaluation Model," January 1993.

29. ANF-CC-33(P)(A) Supplement 2, "HUXY : A Generalized Multirod Heatup Code with 10CFR50 Appendix K Heatup Option," January 1991.

30. ANF-CC-33(P)(A) Supplement 1 Revision 1, "HUXY : A Generalized Multirod Heatup Code with 10CFR50 Appendix K Heatup Option Users Manual," August 1986.

31. XN-NF-80-19(P)(A), Volumes 2, 2A, 2B, and 2C "Exxon Nuclear Methodology for Boiling Water Reactors: EXEM BWR ECCS Evaluation Model," September 1982.

32. XN-NF-80-19(P)(A), Volumes 3 Revision 2 "Exxon Nuclear Methodology for Boiling Water Reactors Thermex: Thermal Limits Methodology Summary Description," January 1987.

33. XN-NF-79-71(P)(A) Revision 2, Supplements 1, 2, and 3, "Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors," March 1986.

34. ANF-1358(P)(A), Revision 1, "The Loss of Feedwater Heating Transient in Boiling Water Reactors," September 1992.

35. ANF-913(P)(A) Volume 1 Revision 1 and Volume 1 Supplements 2, 3, and 4, "COTRANSA2 : A Computer Program for Boiling Water Reactor Transient Analyses," August 1990.

INSERT 3 (cont.)

36. XN-NF-84-105(P)(A), Volume 1 and Volume 1 Supplements 1 and 2, "XCOBRA-T : A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis," February 1987.

37. XN-NF-84-105(P)(A), Volume 1 Supplement 4, "XCOBRA-T : A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis, Void Fraction Model Comparison to Experimental Data," June 1988.