October 9, 1987

Docket No. 50-387

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Mr. Harold W. Keiser
Vice President
Nuclear Operations
Pennsylvania Power and Light Company
2 North Ninth Street
Allentown, Pennsylvania 18101

Dear Mr. Keiser:

SUBJECT: TECHNICAL SPECIFICATION CHANGES TO SUPPORT CYCLE 4 RELOAD (TAC NO. 65636)

RE: SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 1

The Commission has issued the enclosed Amendment No. 72 to Facility Operating License No. NPF-14 for the Susquehanna Steam Electric Station (SSES), Unit 1. This amendment is in response to your letter dated June 19, 1987.

This amendment changes the SSES, Unit 1 Technical Specifications in support of the fuel reload for Cycle 4 operation.

A copy of our Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's Biweekly Federal Register Notice.

Sincerely,

/s/

Walter R. Butler, Director
Project Directorate I-2
Division of Reactor Projects I/II

Enclosures:
1. Amendment No. 72 to License No. NPF-14
2. Safety Evaluation

cc w/enclosures:
See next page

PDI-2/LA
MO'Brien
9/24/87

PDI-2/PM
MThadani:ca
9/24/87

PDI-2/D
WButler
10/8/87

OGC
10/1/87

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

October 9, 1987

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See next page

Mr. Harold W. Keiser
Pennsylvania Power & Light Company

Susquehanna Steam Electric Station
Units 1 & 2

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

PENNSYLVANIA POWER & LIGHT COMPANY

ALLEGHENY ELECTRIC COOPERATIVE, INC.

DOCKET NO. 50-387

SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 72
License No. NPF-14

1. The Nuclear Regulatory Commission (the Commission or the NRC) having found that:

   A. The application for the amendment filed by the Pennsylvania Power & Light Company, dated June 19, 1987, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of the Facility Operating License No. NPF-14 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No.72 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. PP&L shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective prior to startup for Cycle 4 operation.

FOR THE NUCLEAR REGULATORY COMMISSION

/s/

Walter R. Butler, Director
Project Directorate I-2
Division of Reactor Projects I/II

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 9, 1987

PDI-2/LA	PDI-2/PM	OGC	PDI-2/D
MO'Brien	MThadani:ca		WButler
9/24/87	9/24/87	10/1/87	10/8/87

3. This license amendment is effective prior to startup for Cycle 4 operation.

FOR THE NUCLEAR REGULATORY COMMISSION

Walter R. Butler

Walter R. Butler, Director
Project Directorate I-2
Division of Reactor Projects I/II

Attachment:
Changes to the Technical Specifications

Date of Issuance: October 9, 1987

ATTACHMENT TO LICENSE AMENDMENT NO. 72

FACILITY OPERATING LICENSE NO. NPF-14

DOCKET NO. 50-387

Replace the following pages of the Appendix A Technical Specifications with enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change. The overleaf pages are provided to maintain document completeness.*

REMOVE	INSERT
i	i
ii*	ii*
iii*	iii*
iv	iv
xxi	xxi
xxii	xxii
xxv	xxv
1-1	1-1
1-2*	1-2*
1-3	1-3
1-4*	1-4*
B 2-1	B 2-1
-	B 2-1a
B 2-2	B 2-2
3/4 2-1	3/4 2-1
3/4 2-2*	3/4 2-2*
3/4 2-3*	3/4 2-3*
3/4 2-4	3/4 2-4
-	3/4 2-4a
3/4 2-5	3/4 2-5
-	3/4 2-5a
3/4 2-6*	3/4 2-6*
3/4 2-7	3/4 2-7
3/4 2-8*	3/4 2-8*

3/4 2-9	3/4 2-9
3/4 2-9a	3/4 2-9a
3/4 2-10a	3/4 2-10a
3/4 2-10b	3/4 2-10b
-	3/4 2-10c
3/4 4-1b*	3/4 4-1b*
3/4 4-1c	3/4 4-1c
B 3/4 1-1	B 3/4 1-1
B 3/4 1-2*	B 3/4 1-2*
B 3/4 2-1	B 3/4 2-1
B 3/4 2-2*	B 3/4 2-2*
B 3/4 4-1	B 3/4 4-1
B 3/4 4-2*	B 3/4 4-2*
5-5*	5-5*
5-6	5-6

DEFINITIONS

SECTION

SECTION

DEFINITIONS (Continued) PAGE

INDEX

SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

BASES

LIST OF FIGURES

INDEX

LIST OF TABLES

LIST OF TABLES (Continued)

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.

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.

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. 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.

CORE ALTERATION

1.7 CORE ALTERATION shall be the addition, removal, relocation or movement of fuel, sources, or reactivity controls within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Normal-movement of the SRMs, IRMs, TIPs or special moveable detectors is not considered a CORE ALTERATION. Suspension of CORE ALTERATIONS shall not preclude completion of the movement of a component to a safe conservative position.

CRITICAL POWER RATIO

1.8 The CRITICAL POWER RATIO (CPR) shall be the ratio of that power in the assembly which is calculated by application of the appropriate correlation(s) to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power.

DOSE EQUIVALENT I-131

1.9 DOSE EQUIVALENT I-131 shall be that concentration of I-131, microcuries per 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."

$\bar{E}$-AVERAGE DISINTEGRATION ENERGY

1.10 $\bar{E}$ 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, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME

1.11 The EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ECCS actuation setpoint at the channel sensor until the ECCS 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 any series of sequential, overlapping or total steps such that the entire response time is measured.

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME

1.12 The END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME shall be that time interval to complete supression of the electric arc between the fully open contacts of the recirculation pump circuit breaker from initial movement of the associated:

a. Turbine stop valves, and

b. Turbine control valves.

This total system response time consists of two components, the instrumentation response time and the breaker arc suppression time. These times may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

FRACTION OF LIMITING POWER DENSITY

1.13 The FRACTION OF LIMITING POWER DENSITY (FLPD) shall be the LHGR existing at a given location divided by the LHGR specified in Section 3.2.2 for that bundle type.

FRACTION OF RATED THERMAL POWER

1.14 The FRACTION OF RATED THERMAL POWER (FRTP) shall be the measured THERMAL POWER divided by the RATED THERMAL POWER.

FREQUENCY NOTATION

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

GASEOUS RADWASTE TREATMENT SYSTEM

1.16 A GASEOUS RADWASTE TREATMENT SYSTEM shall be 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

1.17 IDENTIFIED LEAKAGE shall be:

a. Leakage into collection systems, such as pump seal or valve packing leaks, that is captured and conducted to a collecting tank, or

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of the leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE.

ISOLATION SYSTEM RESPONSE TIME

1.18 The ISOLATION SYSTEM RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its isolation actuation setpoint at the channel sensor until the isolation valves travel to their required positions. Times shall include diesel generator starting and sequence loading delays where applicable. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

LIMITING CONTROL ROD PATTERN

1.19 A LIMITING CONTROL ROD PATTERN shall be a pattern which results in the core being on a thermal hydraulic limit, i.e., operating on a limiting value for APLHGR, LHGR, or MCPR.

LINEAR HEAT GENERATION RATE

1.20 LINEAR HEAT GENERATION RATE (LHGR) shall be the heat generation per unit length of fuel rod. It is the integral of the heat flux over the heat transfer area associated with the unit length.

LOGIC SYSTEM FUNCTIONAL TEST

1.21 A LOGIC SYSTEM FUNCTIONAL TEST shall be a test of all logic components, ie., all relays and contacts, all trip units, solid state logic elements, etc, of a logic circuit, from sensor through and including the actuated device, to verify OPERABILITY. The LOGIC SYSTEM FUNCTIONAL TEST may be performed by any series of sequential, overlapping or total system steps such that the entire logic system is tested.

MAXIMUM FRACTION OF LIMITING POWER DENSITY

1.22 The MAXIMUM FRACTION OF LIMITING POWER DENSITY (MFLPD) shall be the highest value of the FLPD which exists in the core.

MEMBER(S) OF THE PUBLIC

1.23 MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the utility, its contractors or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational or other purposes not associated with the plant.

MINIMUM CRITICAL POWER RATIO

1.24 The MINIMUM CRITICAL POWER RATIO (MCPR) shall be the smallest CPR which exists in the core for each class of fuel.

OFFSITE DOSE CALCULATION MANUAL

1.25 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the current methodology and parameters used in the calculation of offsite doses due to 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.

OPERABLE - OPERABILITY

1.26 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 CONDITION - CONDITION

1.27 An OPERATIONAL CONDITION, i.e., CONDITION, shall be any one inclusive combination of mode switch position and average reactor coolant temperature as specified in Table 1.2.

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 calculated 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 Specifications 2.1.2 for both GE and Exxon 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. XN-NF-524(A)).

2.1.1 THERMAL POWER, Low Pressure or Low Flow

The use of the XN-3 correlation is valid for critical power calculations at pressures greater than 580 psig and bundle mass fluxes greater than 0.25×10^6 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 ANF 9x9 fuel design, the minimum bundle flow is greater than 30,000 lbs/hr. For the ANF and GE 8x8 fuel, the minimum bundle flow is greater than 28,000 lbs/hr. For all 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 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

2.1 SAFETY LIMITS

BASES

2.1.1 THERMAL POWER, Low Pressure or Low Flow (Continued)

25% thermal power a bundle power of 3.35 Mwt corresponds to a bundle radial peaking factor of greater than 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.

BASES

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. XN-NF-524 describes the methodology used in determining the Safety Limit MCPR.

The XN-3 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 XN-3 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. Still further conservatism is induced by the tendency of the XN-3 correlation to overpredict the number of rods in boiling transition. These conservatisms and the inherent accuracy of the XN-3 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.

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.1 All AVERAGE PLANAR LINEAR HEAT GENERATION RATES (APLHGRs) for each type of fuel as a function of AVERAGE PLANAR EXPOSURE for GE fuel and AVERAGE BUNDLE EXPOSURE for ANF fuel shall not exceed the limits shown in Figures 3.2.1-1, 3.2.1-2, and 3.2.1-3.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

ACTION:

With an APLHGR exceeding the limits of Figure 3.2.1-1, 3.2.1-2, or 3.2.1-3 initiate corrective action within 15 minutes and restore APLHGR to within the required limits within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.1 All APLHGRs shall be verified to be equal to or less than the limits determined from Figures 3.2.1-1, 3.2.1-2, and 3.2.1-3:

a. At least once per 24 hours,

b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

c. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for APLHGR.

d. The provisions of Specification 4.0.4 are not applicable.

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

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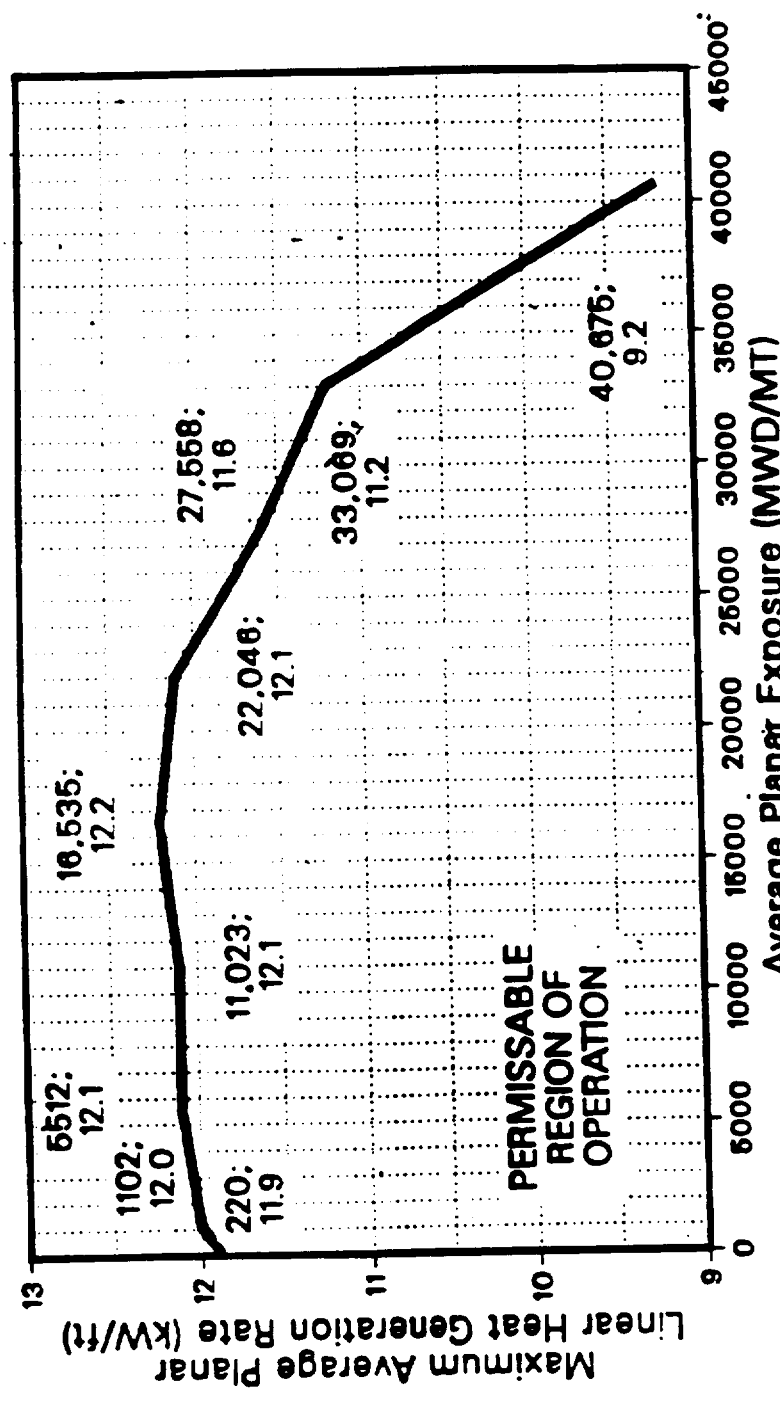
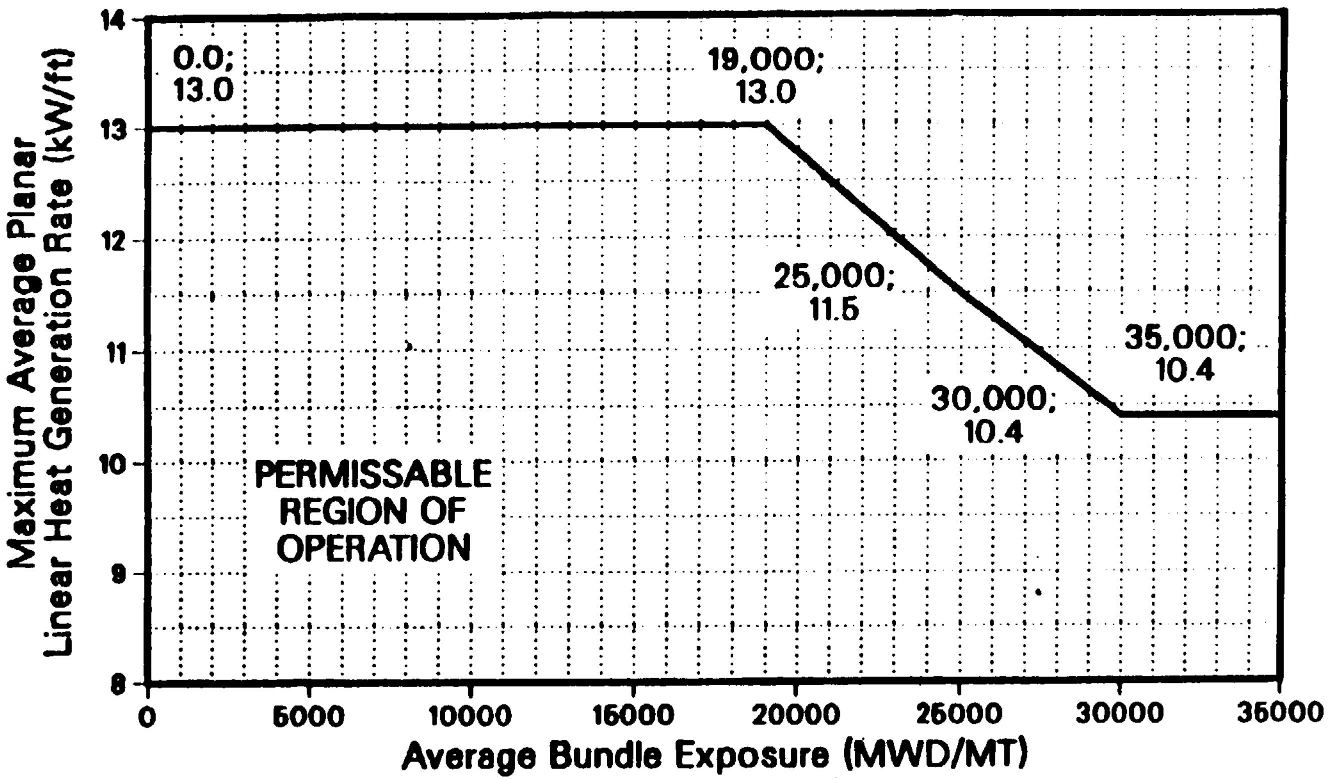
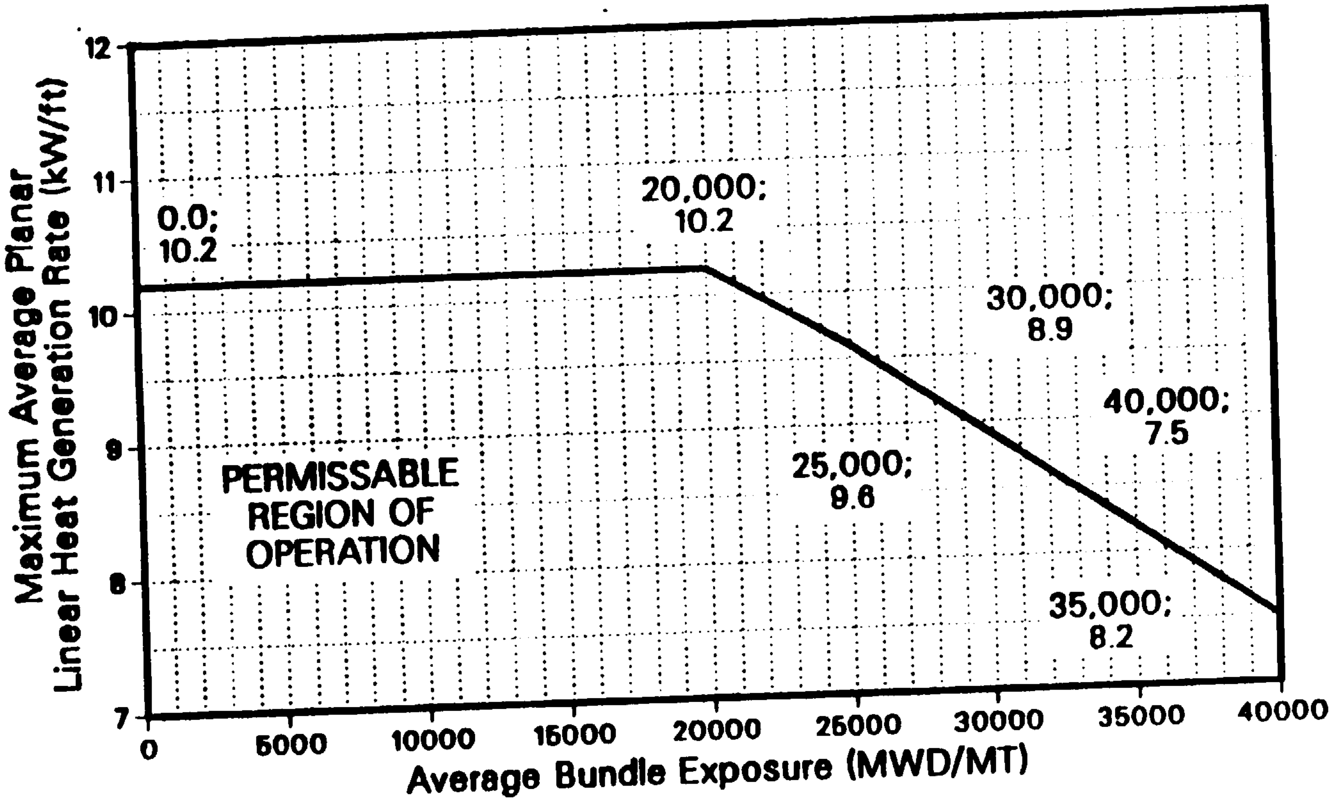
effective upon startup
following the Unit 1 second refueling



MAXIMUM AVERAGE PLANAR LINEAR HEAT
GENERATION RATE (MAPLHGR) VERSUS
AVERAGE PLANAR EXPOSURE
GE FUEL TYPES 8CR233 (2.33% ENRICHED)
FIGURE 3.2.1-1



MAXIMUM AVERAGE PLANAR LINEAR HEAT
GENERATION RATE (MAPLHGR) VERSUS
AVERAGE BUNDLE EXPOSURE
ANF 8X8 FUEL
FIGURE 3.2.1-2



MAXIMUM AVERAGE PLANAR LINEAR HEAT
GENERATION RATE (MAPLHGR) VERSUS
AVERAGE BUNDLE EXPOSURE
ANF 9X9 FUEL
FIGURE 3.2.1-3

3/4.2.2 APRM SETPOINTS

LIMITING CONDITION FOR OPERATION

3.2.2 The APRM flow biased simulated thermal power-upscale scram trip setpoint (S) and flow biased neutron flux-upscale control rod block trip setpoint (S_{RB}) shall be established according to the following relationships:

Trip Setpoint#	Allowable Value#
$S \leq (0.58W + 59\%)T$	$S \leq (0.58W + 62\%)T$
$S_{RB} \leq (0.58W + 50\%)T$	$S_{RB} \leq (0.58W + 53\%)T$

where: S and S_{RB} are in percent of RATED THERMAL POWER,

W = Loop recirculation flow as a percentage of the loop recirculation flow which produces a rated core flow of 100 million lbs/hr,

T = Lowest value of the ratio of FRACTION OF RATED THERMAL POWER divided by the MAXIMUM FRACTION OF LIMITING POWER DENSITY. Where:

a. The FRACTION OF LIMITING POWER DENSITY (FLPD) for GE fuel is the actual LINEAR HEAT GENERATION RATE (LHGR) divided by 13.4 per Specification 3.2.4.1, and

b. The FLPD for ANF fuel is the actual LHGR divided by the LINEAR HEAT GENERATION RATE from Figure 3.2.2-1.

T is always less than or equal to 1.0.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

ACTION:

With the APRM flow biased simulated thermal power-upscale scram trip setpoint and/or the flow biased neutron flux-upscale control rod block trip setpoint less conservative than the value shown in the Allowable Value column for S or S_{RB}, as above determined, initiate corrective action within 15 minutes and adjust S and/or S_{RB} to be consistent with the Trip Setpoint value* within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

*With MFLPD greater than the FRTP during power ascension up to 90% of RATED THERMAL POWER, rather than adjusting the APRM setpoints, the APRM gain may be adjusted such that APRM readings are greater than or equal to 100% times MFLPD, provided that the adjusted APRM reading does not exceed 100% of RATED THERMAL POWER, the required gain adjustment increment does not exceed 10% of RATED THERMAL POWER, and a notice of the adjustment is posted on the reactor control panel.

#See Specification 3.4.1.1.2.a for single loop operation requirements.

SURVEILLANCE REQUIREMENTS

4.2.2 The FRTP and the MFLPD shall be determined, the value of T calculated, and the most recent actual APRM flow biased simulated thermal power-upscale scram and flow biased neutron flux-upscale control rod block trip setpoints verified to be within the above limits or adjusted, as required:

a. At least once per 24 hours,

b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

c. Initially and at least once per 12 hours when the reactor is operating with MFLPD greater than or equal to FRTP.

d. The provisions of Specification 4.0.4 are not applicable.

3/4.2.3 MINIMUM CRITICAL POWER RATIO

LIMITING CONDITION FOR OPERATION

3.2.3 The MINIMUM CRITICAL POWER RATIO (MCPR) shall be greater than or equal to the greater of the two values determined from Figure 3.2.3-1 and Figure 3.2.3-2

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

ACTION:

With MCPR less than the applicable MCPR limit determined above, initiate corrective action within 15 minutes and restore MCPR to within the required limit within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

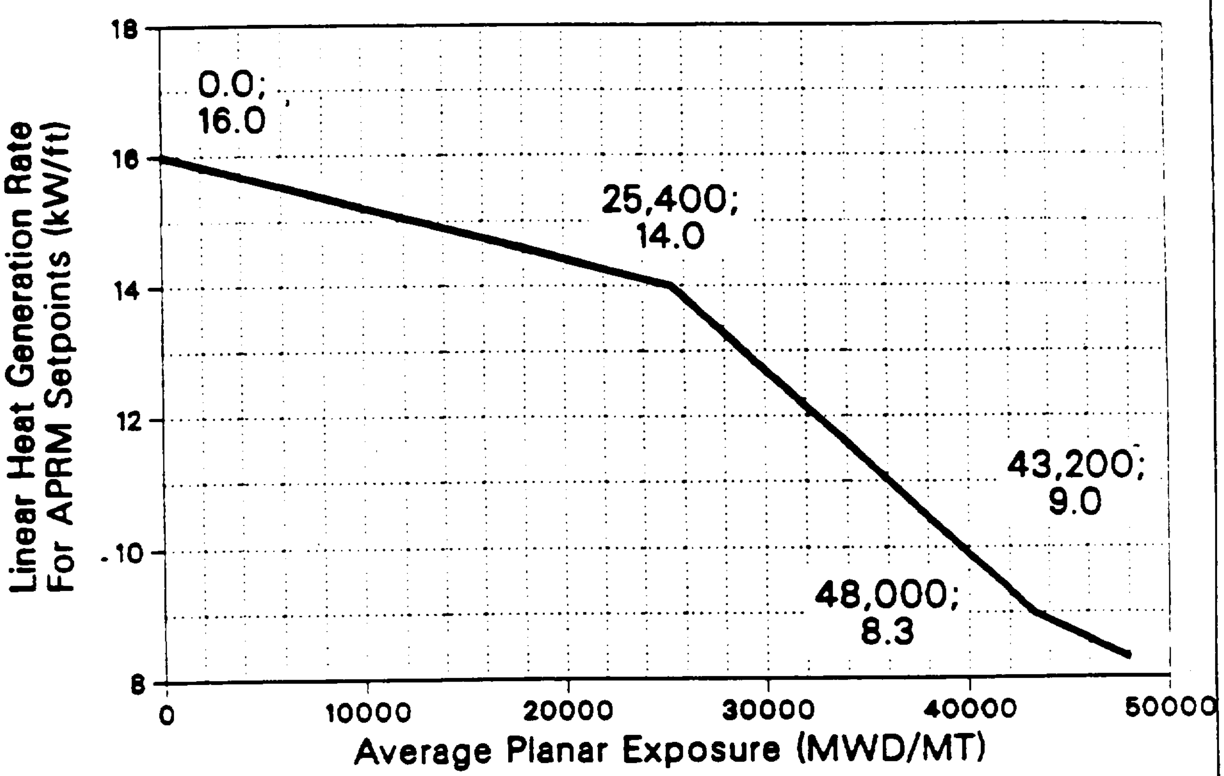
4.2.3.1 MCPR shall be determined to be greater than or equal to the applicable MCPR limit determined from Figure 3.2.3-1 and Figure 3.2.3-2:

a. At least once per 24 hours,

b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

c. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for MCPR.

d. The provisions of Specification 4.0.4 are not applicable.



LINEAR HEAT GENERATION RATE FOR APRM SETPOINTS
VERSUS AVERAGE PLANAR EXPOSURE
ANF FUEL
FIGURE 3.2.2-1

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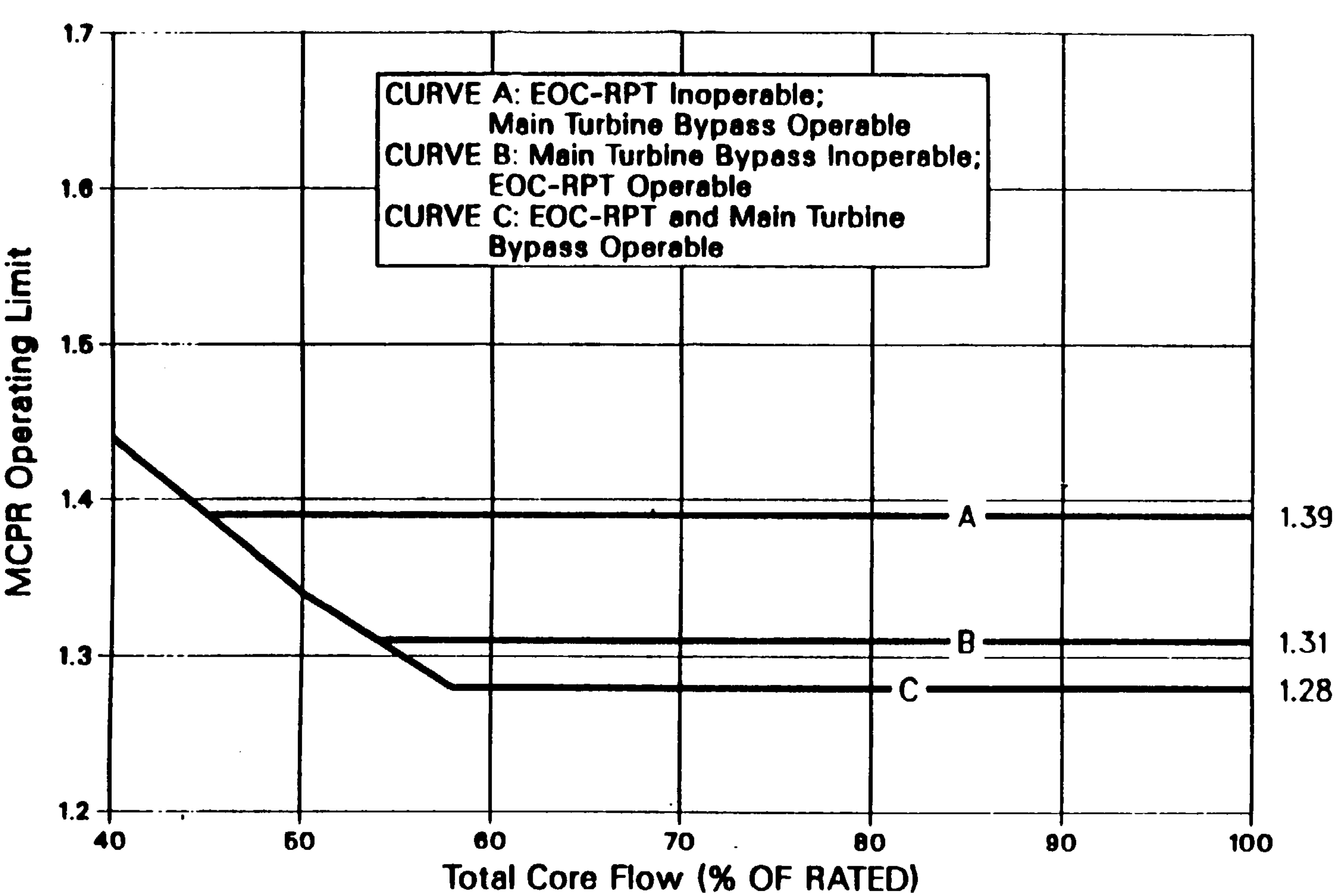
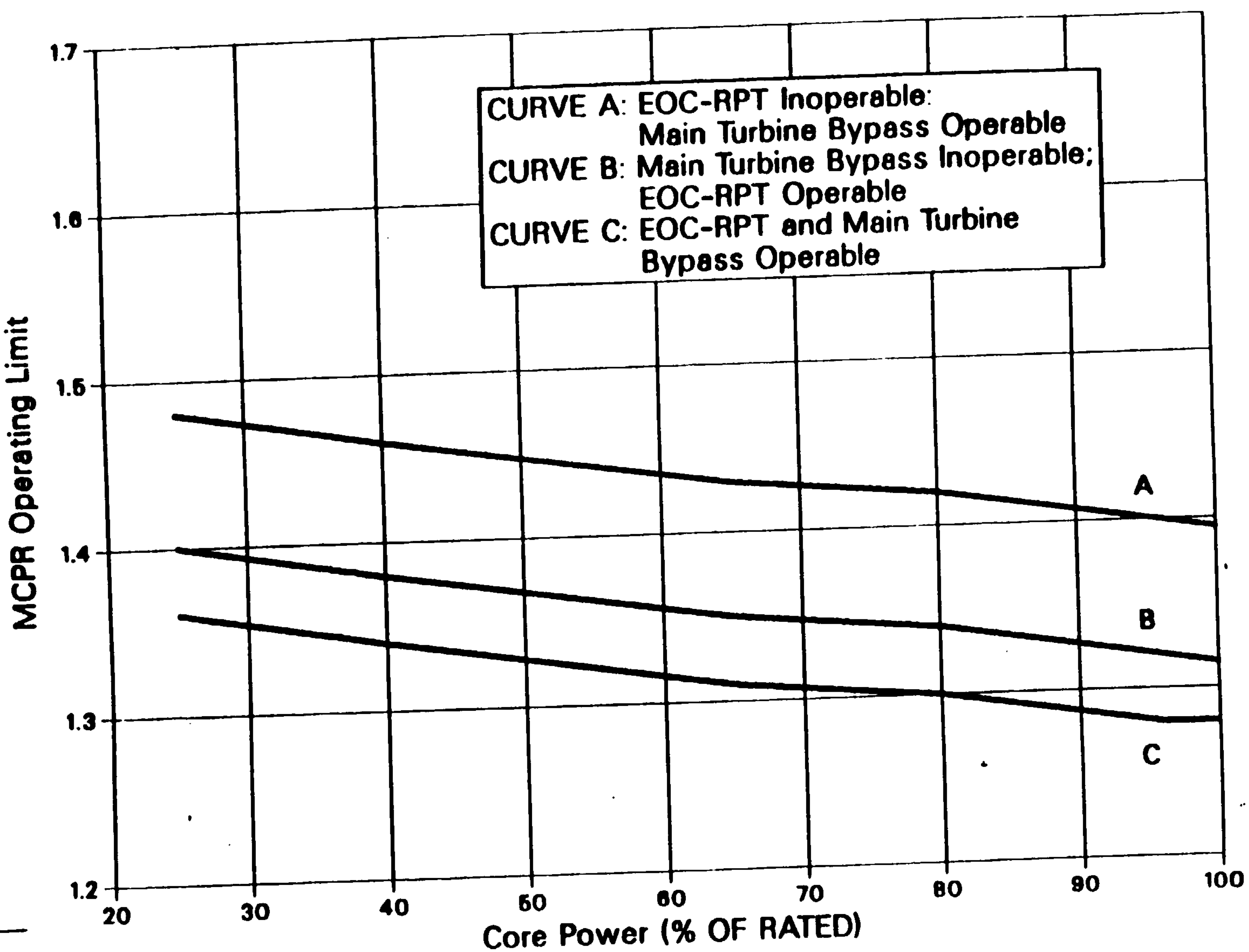


FLOW DEPENDENT MCPR OPERATING LIMIT
FIGURE 3.2.3-1



REDUCED POWER MCPR OPERATING LIMIT
Figure 3.2.3-2

3/4.2.4 LINEAR HEAT GENERATION RATE

ANF FUEL

LIMITING CONDITION FOR OPERATION

3.2.4.2 The LINEAR HEAT GENERATION RATE (LHGR) for ANF fuel shall not exceed the LHGR limit determined from Figures 3.2.4.2-1 and 3.2.4.2-2.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

ACTION:

With the LHGR of any fuel rod exceeding its applicable limit from Figure 3.2.4.2-1 or 3.2.4.2-2, initiate corrective action within 15 minutes and restore the LHGR to within the limit within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

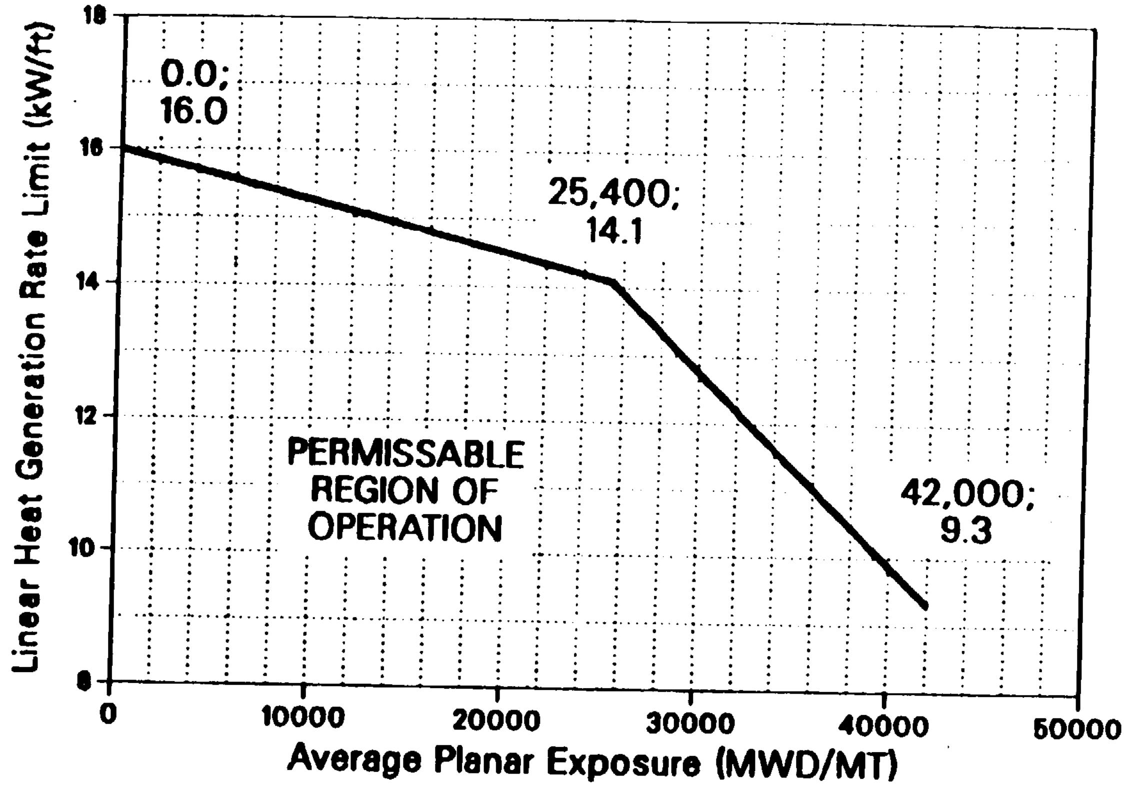
4.2.4.2 LHGRs for ANF fuel shall be determined to be equal to or less than the limit:

a. At least once per 24 hours,

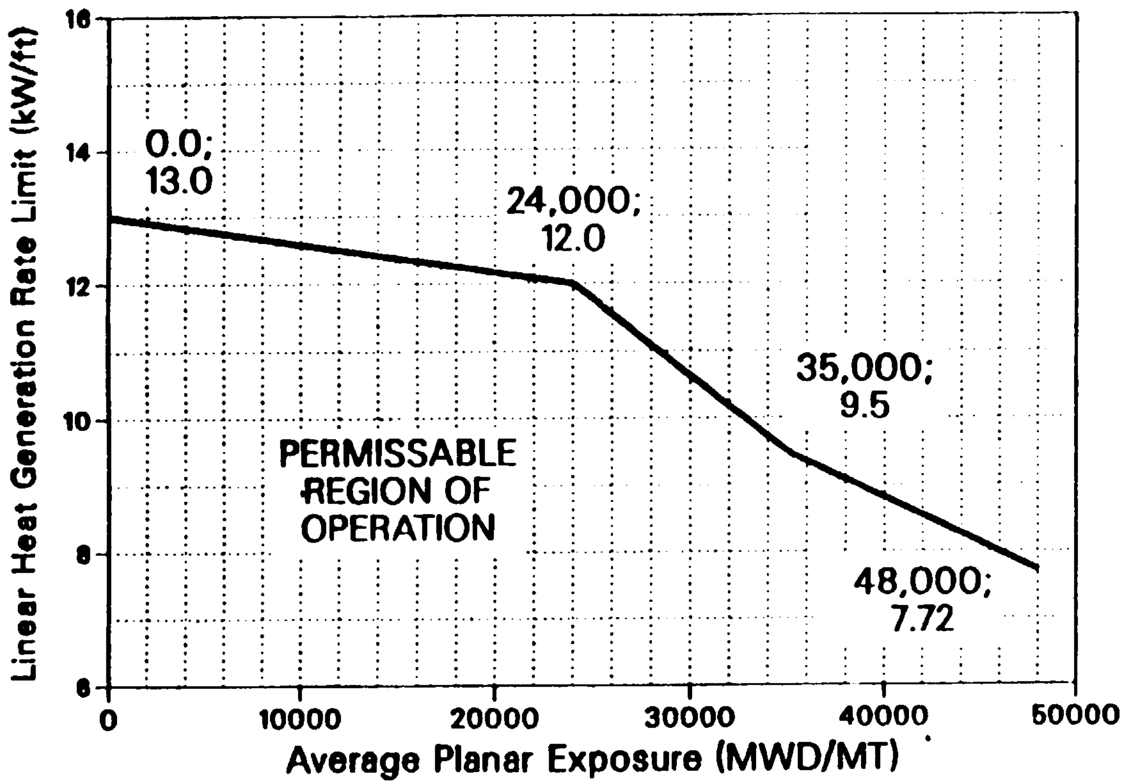
b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

c. Initially and at least once per 12 hours when the reactor is operating on a LIMITING CONTROL ROD PATTERN for LHGR.

d. The provisions of Specification 4.0.4 are not applicable.

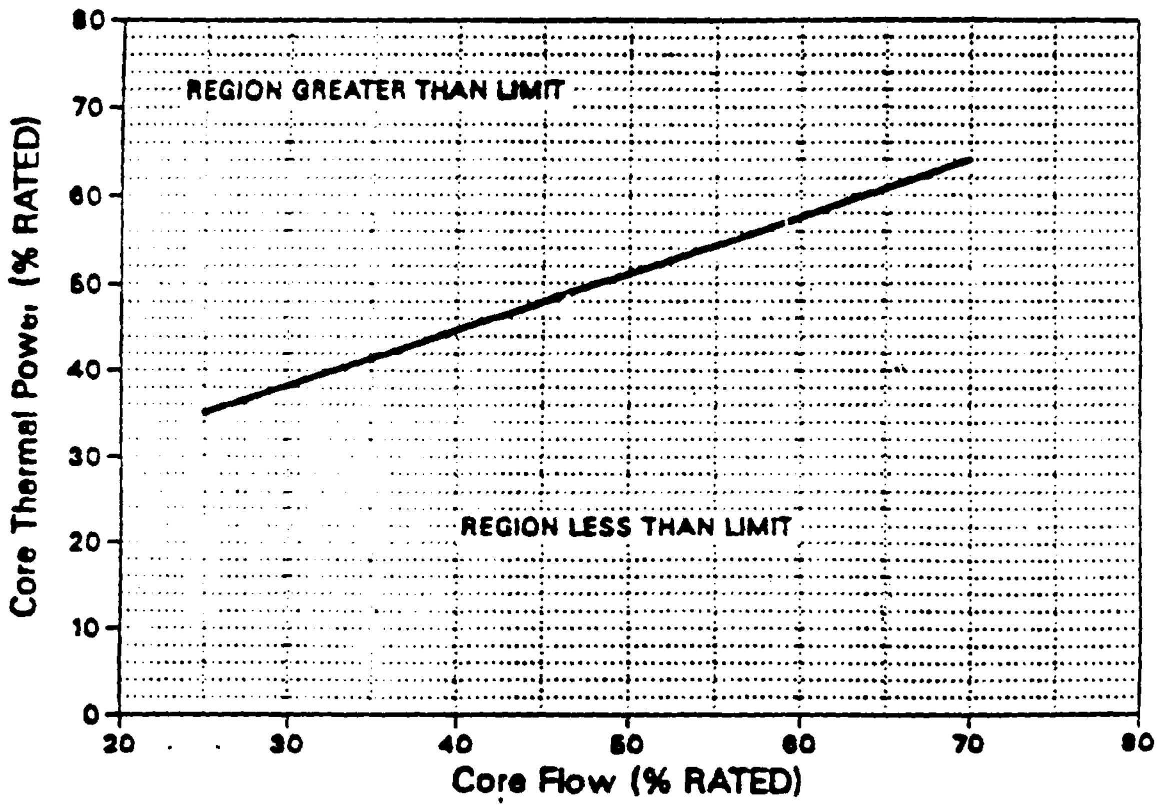


LINEAR HEAT GENERATION RATE (LHGR) LIMIT
VERSUS AVERAGE PLANAR EXPOSURE
ANF 8X8 FUEL
FIGURE 3.2.4.2-1



LINEAR HEAT GENERATION RATE (LHGR) LIMIT
VERSUS AVERAGE PLANAR EXPOSURE
ANF 9X9 FUEL
FIGURE 3.2.4.2-2

Figure 3.4.1.1.1-1
THERMAL POWER LIMITATIONS



RECIRCULATION LOOPS - SINGLE LOOP OPERATION

LIMITING CONDITION FOR OPERATION

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

a. the following revised specification limits shall be followed:

1. Specification 2.1.2: the MCPR Safety Limit shall be increased to 1.07.

2. Table 2.2.1-1: the APRM Flow-Biased Scram Trip Setpoints shall be as follows:

Trip Setpoint	Allowable Value
≤ 0.58W + 55%	≤ 0.58W + 58%.

3. Specification 3.2.1: The MAPLHGR limits shall be as follows:

a. GE fuel: the limits specified in Figure 3.2.1-1 multiplied by 0.81.

b. ANF fuel: the limits specified in Figures 3.2.1-2 and 3.2.1-3 multiplied by 0.0.

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

Trip Setpoint	Allowable Value
$S \leq (0.58W + 55\%)T$	$S \leq (0.58W + 58\%)T$
$S_{RB} \leq (0.58W + 46\%)T$	$S_{RB} \leq (0.58W + 49\%)T$

5. Table 3.3.6-2: the RBM/APRM Control Rod Block Setpoints shall be as follows:

	Trip Setpoint	Allowable Value
a. RBM - Upscale	≤ 0.66W + 37%	≤ 0.66W + 40%
b. APRM-Flow Biased	≤ 0.58W + 46%	≤ 0.58W + 49%

b. APRM and LPRM*** neutron flux noise levels shall be less than three times their established baseline levels when THERMAL POWER is greater than the limit specified in Figure 3/4.1.1.1-1.

c. Total core flow shall be greater than or equal to 42 million lbs/hr when THERMAL POWER is greater than the limit specified in Figure 3.4.1.1.1-1.

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

BASES

3/4.1.1 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

Since core reactivity values will vary through core life as a function of fuel depletion and poison burnup, the demonstration of SHUTDOWN MARGIN will be performed in the cold, xenon-free condition and shall show the core to be subcritical by at least R + 0.38% delta k/k or R + 0.28% delta k/k, as appropriate. The value of R in units of % delta k/k is the difference between the calculated beginning of cycle shutdown margin minus the calculated minimum shutdown margin in the cycle, where shutdown margin is a positive number. The value of R must be positive or zero and must be determined for each fuel loading cycle.

Two different values are supplied in the Limiting Condition for Operation to provide for the different methods of demonstration of the SHUTDOWN MARGIN. The highest worth rod may be determined analytically or by test. The SHUTDOWN MARGIN is demonstrated by control rod withdrawal at the beginning of life fuel cycle conditions, and, if necessary, at any future time in the cycle if the first demonstration indicates that the required margin could be reduced as a function of exposure. Observation of subcriticality in this condition assures subcriticality with the most reactive control rod fully withdrawn.

This reactivity characteristic has been a basic assumption in the analysis of plant performance and can be best demonstrated at the time of fuel loading, but the margin must also be determined anytime a control rod is incapable of insertion.

3/4.1.2 Reactivity Anomalies

Since the SHUTDOWN MARGIN requirement is small, a careful check on actual reactor conditions compared to the predicted conditions is necessary. Any changes in reactivity from that of the predicted (predicted core k_{eff}) can be determined from the core monitoring system (monitored core k_{eff}). In the absence of any deviation in plant operating conditions or reactivity anomaly, these values should be essentially equal since the calculational methodologies are consistent. The predicted core k_{eff} is calculated by a 3D core simulation code as a function of cycle exposure. This is performed for projected or anticipated reactor operating states/conditions throughout the cycle and is usually done prior to cycle operation. The monitored core k_{eff} is the k_{eff} as calculated by the core monitoring system for actual plant conditions.

Since the comparisons are easily done, frequent checks are not an imposition on normal operation. A 1% deviation in reactivity from that of the predicted is larger than expected for normal operation, and therefore should be throughly evaluated. A deviation as large as 1% would not exceed the design conditions of the reactor.

3/4.1.3 CONTROL RODS

The specification of this section ensure that (1) the minimum SHUTDOWN MARGIN is maintained, (2) the control rod insertion times are consistent with those used in the accident analysis, and (3) limit the potential effects of the rod drop accident. The ACTION statements permit variations from the basic requirements but at the same time impose more restrictive criteria for continued operation. A limitation on inoperable rods is set such that the resultant effect on total rod worth and scram shape will be kept to a minimum. The requirements for the various scram time measurements ensure that any indication of systematic problems with rod drives will be investigated on a timely basis.

Damage within the control rod drive mechanism could be a generic problem, therefore with a control rod immovable because of excessive friction or mechanical interference, operation of the reactor is limited to a time period which is reasonable to determine the cause of the inoperability and at the same time prevent operation with a large number of inoperable control rods.

Control rods that are inoperable for other reasons are permitted to be taken out of service provided that those in the nonfully-inserted position are consistent with the SHUTDOWN MARGIN requirements.

The number of control rods permitted to be inoperable could be more than the eight allowed by the specification, but the occurrence of eight inoperable rods could be indicative of a generic problem and the reactor must be shutdown for investigation and resolution of the problem.

The control rod system is designed to bring the reactor subcritical at a rate fast enough to prevent the MCPR from becoming less than the limit specified in Specification 2.1.2 during the core wide transient analyzed in the cycle specific transient analysis report. This analysis shows that the negative reactivity rates resulting from the scram with the average response of all the drives as given in the specifications, provide the required protection and MCPR remains greater than the limit specified in Specification 2.1.2. The occurrence of scram times longer then those specified should be viewed as an indication of a systematic problem with the rod drives and therefore the surveillance interval is reduced in order to prevent operation of the reactor for long periods of time with a potentially serious problem.

The scram discharge volume is required to be OPERABLE so that it will be available when needed to accept discharge water from the control rods during a reactor scram and will isolate the reactor coolant system from the containment when required.

Control rods with inoperable accumulators are declared inoperable and Specification 3.1.3.1 then applies. This prevents a pattern of inoperable accumulators that would result in less reactivity insertion on a scram than has been analyzed even though control rods with inoperable accumulators may still be inserted with normal drive water pressure. Operability of the accumulator ensures that there is a means available to insert the control rods even under the most unfavorable depressurization of the reactor.

Effective upon startup following Unit 1 second refueling outage

The specifications of this section assure that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the 2200°F limit specified in 10 CFR 50.46.

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the limit specified in 10 CFR 50.46.

The peak cladding temperature (PCT) following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is dependent only secondarily on the rod to rod power distribution within an assembly. For GE fuel, the peak clad temperature is calculated assuming a LHGR for the highest powered rod which is equal to or less than the design LHGR corrected for densification. This LHGR times 1.02 is used in the heatup code along with the exposure dependent steady state gap conductance and rod-to-rod local peaking factor. The Technical Specification AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) for GE fuel is this LHGR of the highest powered rod divided by its local peaking factor which results in a calculated LOCA PCT much less than 2200°F. The Technical Specification APLHGR for ANF fuel is specified to assure the PCT following a postulated LOCA will not exceed the 2200°F limit. The limiting value for APLHGR is shown in Figures 3.2.1-1, 3.2.1-2, and 3.2.1-3.

The calculational procedure used to establish the APLHGR shown on Figures 3.2.1-1, 3.2.1-2, and 3.2.1-3 is based on a loss-of-coolant accident analysis. The analysis was performed using calculational models which are consistent with the requirements of Appendix K to 10 CFR 50. These models are described in Reference 1 or XN-NF-80-19, Volumes 2, 2A, 2B and 2C.

3/4.2.2 APRM SETPOINTS

The flow biased simulated thermal power-upscale scram setting and flow biased simulated thermal power-upscale control rod block functions of the APRM instruments limit plant operations to the region covered by the transient and accident analyses. In addition, the APRM setpoints must be adjusted to ensure that $\geq$1% plastic strain and fuel centerline melting do not occur during the worst anticipated operational occurrence (AOO), including transients initiated from partial power operation.

For ANF fuel the T factor used to adjust the APRM setpoints is based on the FLPD calculated by dividing the actual LHGR by the LHGR obtained from Figure 3.2.2-1. The LHGR versus exposure curve in Figure 3.2.2-1 is based on ANF's Protection Against Fuel Failure (PAFF) line shown in Figure 3.4 of XN-NF-85-67(A), Revision 1. Figure 3.2.2-1 corresponds to the ratio of PAFF/1.2 under which cladding and fuel integrity is protected during AOOs.

For GE fuel the T factor used to adjust the APRM setpoints is based on the FLPD calculated by dividing the actual LHGR by the LHGR limit specified for GE fuel in Specification 3.2.4.1.

3/4.2.3 MINIMUM CRITICAL POWER RATIO

The required operating limit MCPRs at steady state operating conditions as specified in Specification 3.2.3 are derived from the established fuel cladding integrity Safety Limit MCPR, and an analysis of abnormal operational transients. For any abnormal operating transient analysis evaluation with the initial condition of the reactor being at the steady state operating limit, it is required that the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient assuming instrument trip setting given in Specification 2.2.

To assure that the fuel cladding integrity Safety Limit is not exceeded during any anticipated abnormal operational transient, the most limiting transients have been analyzed to determine which result in the largest reduction in CRITICAL POWER RATIO (CPR). The type of transients evaluated were loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest delta MCPR. When added to the Safety Limit MCPR, the required minimum operating limit MCPR of Specification 3.2.3 is obtained and presented in Figures 3.2.3-1 and 3.2.3-2.

The evaluation of a given transient begins with the system initial parameters shown in the cycle specific transient analysis report that are input to a Exxon-core dynamic behavior transient computer program. The outputs of this program along with the initial MCPR form the input for further analyses of the thermally limiting bundle. The codes and methodology to evaluate pressurization and non-pressurization events are described in XN-NF-79-71 and XN-NF-84-105. The principal result of this evaluation is the reduction in MCPR caused by the transient.

Figure 3.2.3-1 defines core flow dependent MCPR operating limits which assure that the Safety Limit MCPR will not be exceeded during a flow increase transient resulting from a motor-generator speed control failure. The flow dependent MCPR is only calculated for the manual flow control mode. Therefore, automatic flow control operation is not permitted. Figure 3.2.3-2 defines the power dependent MCPR operating limit which assures that the Safety Limit MCPR will not be exceeded in the event of a feedwater controller failure initiated from a reduced power condition.

Cycle specific analyses are performed for the most limiting local and core wide transients to determine thermal margin. Additional analyses are performed to determine the MCPR operating limit with either the Main Turbine Bypass inoperable or the EOC-RPT inoperable. Analyses to determine thermal margin with both the EOC-RPT inoperable and Main Turbine Bypass inoperable have not been performed. Therefore, operation in this condition is not permitted.

At THERMAL POWER levels less than or equal to 25% of RATED THERMAL POWER, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience indicates that the resulting MCPR value is in excess of requirements by a considerable margin. During initial start-up testing of the plant, a MCPR evaluation

3/4.4.1 RECIRCULATION SYSTEM

Operation with one reactor recirculation loop inoperable has been evaluated and found acceptable, provided that the unit is operated in accordance with Specification 3.4.1.1.2.

For single loop operation, the MAPLHGR limits for ANF fuel are multiplied by a factor of 0.0. This multiplication factor precludes extended operation with one loop out of service.

For single loop operation, the RBM and APRM setpoints are adjusted by a 7% decrease in recirculation drive flow to account for the active loop drive flow that bypasses the core and goes up through the inactive loop jet pumps.

Surveillance on the pump speed of the operating recirculation loop is imposed to exclude the possibility of excessive reactor vessel internals vibration. Surveillance on differential temperatures below the threshold limits on THERMAL POWER or recirculation loop flow mitigates undue thermal stress on vessel nozzles, recirculation pumps and the vessel bottom head during extended operation in the single loop mode. The threshold limits are those values which will sweep up the cold water from the vessel bottom head.

THERMAL POWER, core flow, and neutron flux noise level limitations are prescribed in accordance with the recommendations of General Electric Service Information Letter No. 380, Revision 1, "BWR Core Thermal Hydraulic Stability," dated February 10, 1984.

An inoperable jet pump is not, in itself, a sufficient reason to declare a recirculation loop inoperable, but it does, in case of a design-basis-accident, increase the blowdown area and reduce the capability of reflooding the core; thus, the requirement for shutdown of the facility with a jet pump inoperable. Jet pump failure can be detected by monitoring jet pump performance on a prescribed schedule for significant degradation.

Recirculation pump speed mismatch limits are in compliance with the ECCS LOCA analysis design criteria for two loop operation. The limits will ensure an adequate core flow coastdown from either recirculation loop following a LOCA. In the case where the mismatch limits cannot be maintained during the loop operation, continued operation is permitted in the single loop mode.

In order to prevent undue stress on the vessel nozzles and bottom head region, the recirculation loop temperatures shall be within 50°F of each other prior to startup of an idle loop. The loop temperature must also be within 50°F of the reactor pressure vessel coolant temperature to prevent thermal shock to the recirculation pump and recirculation nozzles. Since the coolant in the bottom of the vessel is at a lower temperature than the coolant in the upper regions of the core, undue stress on the vessel would result if the temperature difference was greater than 145°F.

3/4.4.2 SAFETY/RELIEF VALVES

The safety valve function of the safety/relief valves operate to prevent the reactor coolant system from being pressurized above the Safety Limit of 1325 psig in accordance with the ASME Code. A total of 10 OPERABLE safety-relief valves is required to limit reactor pressure to within ASME III allowable values for the worst case upset transient.

Demonstration of the safety/relief valve lift settings will occur only during shutdown and will be performed in accordance with the provisions of Specification 4.0.5.

3/4.4.3 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.3.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the reactor coolant pressure boundary.

3/4.4.3.2 OPERATIONAL LEAKAGE

The allowable leakage rates from the reactor coolant system have been based on the predicted and experimentally observed behavior of cracks in pipes. The normally expected background leakage due to equipment design and the detection capability of the instrumentation for determining system leakage was also considered. The evidence obtained from experiments suggests that for leakage somewhat greater than that specified for UNIDENTIFIED LEAKAGE the probability is small that the imperfection or crack associated with such leakage would grow rapidly. However, in all cases, if the leakage rates exceed the values specified or the leakage is located and known to be PRESSURE BOUNDARY LEAKAGE, the reactor will be shutdown to allow further investigation and corrective action.

The Surveillance Requirements for RCS pressure isolation valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA.

3/4.4.4 CHEMISTRY

The water chemistry limits of the reactor coolant system are established to prevent damage to the reactor materials in contact with the coolant. Chloride limits are specified to prevent stress corrosion cracking of the stainless steel. The effect of chloride is not as great when the oxygen concentration in the coolant is low, thus the 0.2 ppm limit on chlorides is permitted during POWER OPERATION. During shutdown and refueling operations, the temperature necessary for stress corrosion to occur is not present so a 0.5 ppm concentration of chlorides is not considered harmful during these periods.

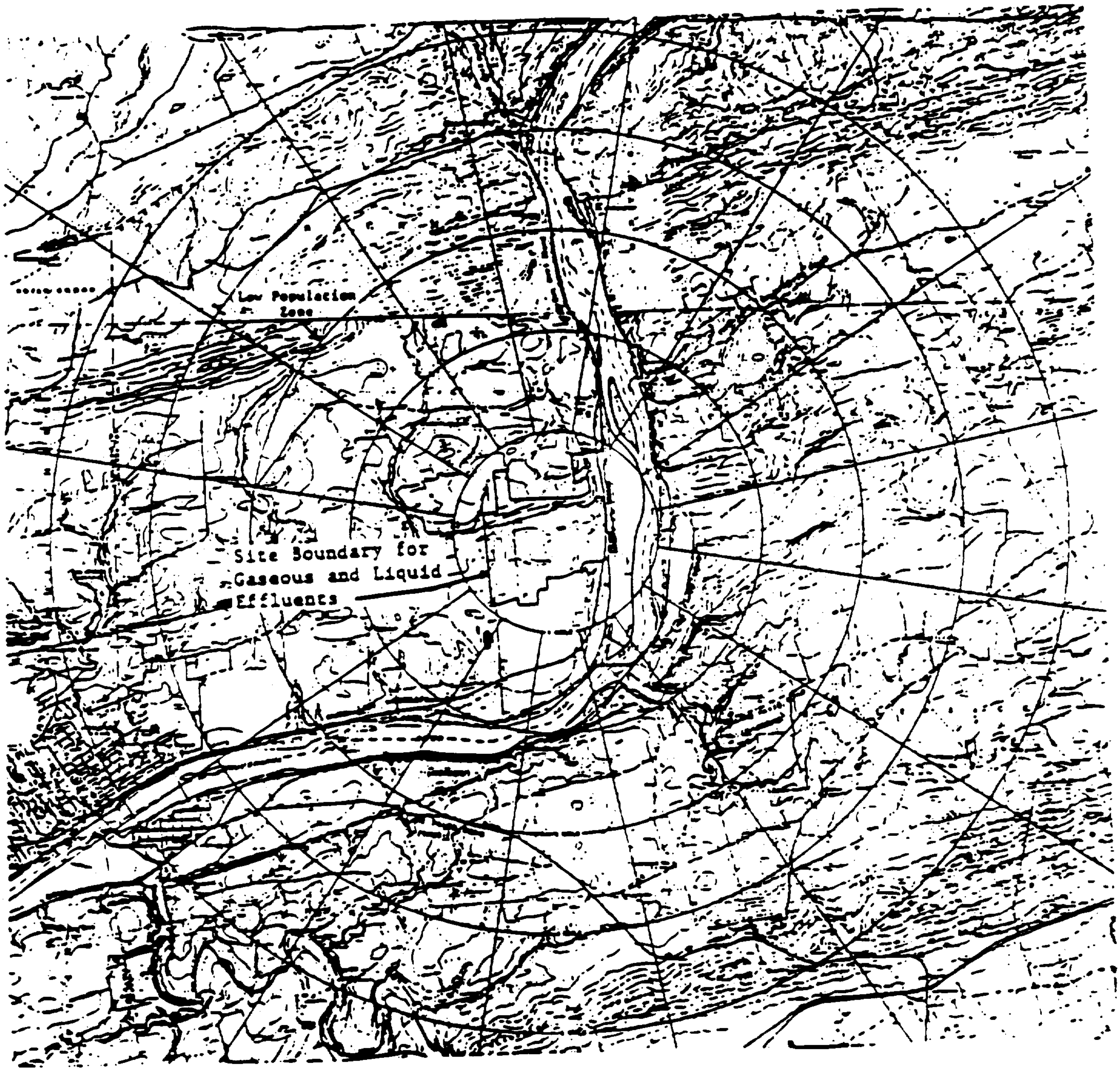


FIGURE 5.1.3-1b

MAP DEFINING UNRESTRICTED AREAS
FOR RADIOACTIVE GASEOUS AND LIQUID EFFLUENTS

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 764 fuel assemblies with each fuel assembly containing 62 or 79 fuel rods and two water rods clad with Zircaloy -2. Each fuel rod shall have a nominal active fuel length of 150 inches. The initial core loading shall have a maximum average enrichment of 1.90 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum average enrichment of 4.0 weight percent U-235.

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 185 control rod assemblies, each consisting of a cruciform array of stainless steel tubes containing 143 inches of boron carbide, B_4C, powder surrounded by a cruciform shaped stainless steel sheath.

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 528°F.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 72 TO FACILITY OPERATING LICENSE NO. NPF-14

PENNSYLVANIA POWER & LIGHT COMPANY

ALLEGHENY ELECTRIC COOPERATIVE, INC.

DOCKET NO. 50-387

SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 1

1.0 INTRODUCTION

By letter dated June 19, 1987, Pennsylvania Power & Light Company (the licensee) requested an amendment to Facility Operating License No. NPF-14 for the Susquehanna Steam Electric Station (SSES), Unit 1. The proposed amendment furnished information to support authorization for SSES Unit 1 operation with 9X9 reload fuel by Advanced Nuclear Fuels (ANF) Corporation and would revise the SSES Unit 1 Technical Specifications in support of the forthcoming fuel reload and restart for Cycle 4 operation. The Cycle 4 (hereafter referred to as S1C4) reload will consist of 240 new 9X9 fuel bundles intermixed with 488 ANF 8X8 and 36 General Electric (GE) 8X8 fuel bundles from the previous cycle. The new 9X9 bundles are comprised of 79 active fuel rods and two inert water rods. In support of the S1C4 reload, the licensee submitted topical reports which summarize the reload scope, the plant transient analyses, and the design and safety analyses. Specifically, the licensee has requested to change the following Technical Specifications:

- Definitions 1.2 and 1.13, related to fuel exposure and fraction of limiting power density
- Specification 3/4.2.1, related to Average Planar Linear Heat Generation Rate (APLHGR)
- Specification 3/4.2.2, related to Average Power Range Monitor (APRM) Setpoints
- Specification 3/4.2.3, related to Minimum Critical Power Ratio (MCPR)
- Specification 3/4.2.4, related to Linear Heat Generation Rate (LHGR)
- Specification 3/4.4.1.1-2, related to Recirculation Loops - Single Loop Operation (SLO)
- Specification 5.3.1, related to Fuel Assemblies

2.0 EVALUATION

The staff has evaluated the licensee's S1C4 core reload request by considering the adequacy of (1) fuel mechanical design, (2) thermal hydraulic design, (3) transient and accident analysis, and (4) the proposed Technical Specifiction changes. The staff's evaluation is summarized as follows.

2.1 Fuel Mechanical Design

The S1C4 core reload will include 240 ANF Corporation 9X9 fuel assemblies with the designation XN-3. These reload assemblies contain 79 fuel rods and two water rods. The 240 assemblies will have a bundle enrichment of 3.31 percent. The fuel design and safety analysis for the 9X9 fuel are described in the SSES 1 specific report PL-NF-87-005 and the generic mechanical design report XN-NF-85-67, Revision 1. The staff approved the latter report and issued its Safety Evaluation on July 23, 1986.

Table 2.1 of XN-NF-85-67, Revision 1 gives the pertinent design data for the ANF 9X9 fuel. Neutronic values specific to the S1C4 reload are given in Table 4.1 of PL-NF-87-005. The burnable poison rods contain 4.00 weight percent gadolinia blended with 3.27 weight percent U-235 to reduce the initial reactivity. The ANF SN-3 fuel is designed to fit into the existing GE channel boxes. A more detailed description can be found in Table 2.1 of XN-NF-85-67. Based on our review of the information in Table 2.1, we find the mechanical design of the ANF 9X9 fuel for the S1C4 reload to be acceptable. However, approval of extended exposure limits for future operating cycles is contingent upon our approval of Supplements to XN-NF-82-06(P) related to 9X9 fuel.

Rod Pressure

For the S1C4 ANF 9X9 reload fuel, calculation of the fuel rod internal pressure was done in accordance with acceptance criteria cited by ANF. The evaluation was performed with RODEX 2A which is a revision of the RODEX2 code (revised fission gas release model) used in the analysis of previous ANF fuel designs. Our review of the RODEX 2A topical report is complete and the staff Safety Evaluation was issued on June 24, 1986. The staff has concluded that the acceptance criteria for rod internal pressure can be fully met throughout the entire expected irradiation life of the 9X9 fuel.

Fuel Rod Bow

Our review of XN-NF-85-67, Revision 1 has been completed. Based on that review we conclude that ANF has demonstrated conformance to approved rod bow design limits for minimum gap spacing to a fuel assembly exposure of 23,000 MWD/MTU for the 9X9 fuel. Projected peak assembly burnups for the S1C4 reload is in the range of 11,000-13,000 MWD/MTU for the 9X9 fuel. Additional information on rod bow measurements on the ANF 9X9 Lead Test

Assemblies has been provided to justify burnup exposure levels up to 40,000 MWD/MTU. Our review of the information for the fuel assembly exposure level above 23,000 MWD/MTU is not complete. Therefore, the ANF 9X9 fuel is approved for S1C4 only (the fuel exposure level for S1C4 is not expected to exceed 13,000 MWD/MTU). Future approval of operation with ANF 9X9 fuel for exposures beyond 23,000 MWD/MTU is contingent upon our approval of the additional rod bow considerations when the staff review is complete.

Fuel Centerline Melting

The design basis for the ANF fuel centerline temperature is that no fuel centerline melting should result from normal operation including transient occurrences. The results of an evaluation reported in the S1C4 reload analysis report PN-NF-87-005 were based on RODEX 2A. RODEX 2A has been previously reviewed and approved and the staff has concluded that the generic methodology for the ANF 9X9 fuel is acceptable for the S1C4 reload fuel.

Cladding Swelling and Rupture

The cladding swelling and rupture models in XN-NF-82-07 (EXXON Nuclear Company Cladding Swelling and Rupture Model) have been approved for use in the ANF (old ENC) ECCS Evaluation Model and have been incorporated in the approved ANF EXEM/BWR ECCS model. This model was used in the ANF ECCS analysis for the S1C4. The staff has verified that ANF is using the approved model for the 9X9 fuel ECCS analysis and we find the application to be acceptable.

Linear Heat Generation Rate (LHGR) - Limit for ANF 9X9 Fuel

The licensee has provided a figure of LHGR Limit vs Planar Exposure for the ANF 9X9 fuel to be incorporated into the SSES Unit 1 Technical Specifications (Figure 3.2.4.2.-2). This Figure was approved in the staff's safety evaluation for licensing topical report XN-NF-85-67(P), dated July 23, 1987, and reflects the design values which have been previously reviewed and approved. Based on the results of the generic review we find the LHGR limits for the 9X9 fuel to be acceptable. This acceptability also applies to the exposure-dependent LHGR provided in proposed Figure 3.2.2-1 which is based on ANF's "Protection Against Fuel Failure" concept which was also part of the generic review.

LOCA-Seismic Mechanical Response

The licensee has discussed the mechanical response of the ANF 9X9 fuel assembly design during LOCA-seismic events. The discussion included a comparison of the physical and structural properties of the new 9X9 fuel and the prior ANF and GE 8X8 fuel. The staff has reviewed this information in connection with a previous review (the staff Safety

Evaluation Supporting Amendment No. 31 to Facility Operating License No. NPF-22 dated October 3, 1986). The staff has confirmed that the physical and structural characteristics of the ANF and GE fuel assemblies are sufficiently similar so that the mechanical response to design Seismic-LOCA events is essentially the same. Based on the considerations discussed above, we conclude that the staff's original analysis is applicable to SSES Unit 1 and the analysis indicating that the design limits are not exceeded is also acceptable.

Nuclear Design

ANF nuclear design methodologies for S1C4 are updated to reflect criteria applicable to the ANF fuel. The S1C4 reload replaces about one-third of Cycle 3 fuel with new ANF 9X9 fuel. The loading pattern is a normal type of scattered configuration. The bundle average enrichment of the new assemebly is 3.31 weight percent U235.

The beginning of cycle shutdown margin is calculated to be 1.63 percent delta-k/k, and the R factor is zero, thus the cycle minimum shutdown margin is well in excess of the required 0.38 percent delta-k/k. The Standby Liquid Control System also fully meets shutdown requirements.

The existing new fuel storage calculations are based on k-infinity of the assembly. Based on new calculations by ANF with consideration given to the 9X9 fuel, if the maximum enrichment zone is such that k-infinity is less than or equal to 1.388 at limiting state conditions then the required criticality limits are met. The existing spent fuel pool criticality calculations have met criteria using a U235 assembly average enrichment of less than 4.00 percent and no burnable poison. Since the maximum enrichment of the new fuel is 3.42 percent, the new calculations show adequate margin to spent fuel pool criticality.

The SSES will continue to use the EXXON (now ANF) POWERPLEX core monitoring system to monitor reactor parameters. The system has been in use during all SSES Unit 1 operating cycles and has provided suitable monitoring and predictive results.

2.2 THERMAL HYDRAULIC DESIGN

The review of the thermal-hydraulic aspects of the S1C4 reload consisted of the following; (a) the compatibility of the ANF 9X9 and prior ANF 8X8 fuel bundles; (b) the fuel cladding integrity safety limit; (c) the operating limit minimum critical power ratio (OLMCPR); (d) thermal hydraulic stability for S1C4; and (e) the proposed technical specifications.

The objective of the review was to confirm that the thermal-hydraulic design of the reload core was accomplished using acceptable analytical methods, provided an acceptable margin of safety from conditions which would lead to fuel damage during normal operation and anticipated operational occurrences and ensured that the core is not susceptible to thermal-hydraulic instability.

Hydraulic Compatibility

Since a BWR core is a series of parallel flow channels connected to a common lower and upper plenum, the total pressure drop across the bundles will be equal. However, differences in the hydraulic resistances of the fuel designs may cause variations in axial pressure drop profiles across the bundles. Component hydraulic resistances for the proposed constituent fuel types in the S1C4 core have been determined in single phase flow tests of full scale assemblies. Additional discussion of the effects of hydraulic compatibility on thermal margin were presented in the S1C4 reload report. Based on our review of the information provided in the pertinent documentation we conclude that the ANF fuel types are hydraulically compatible.

Thermal-Hydraulic Stability

The thermal-hydraulic stability (THS) of the projected Cycle 4 core was analyzed using the methods identified in Exxon Report XN-NF-80-19, Volume 4, Revision 1. That report cites the use of the COTRAN model for use in the analysis of core thermal-hydraulic stability. For two pump minimum flow, the maximum decay ratio computed with the ANF methodology for S1C4 operation is 0.74 at the APRM rod block intercept line (64 percent rated power).

The Cycle 4 reload is the first full reload batch of ANF 9X9 fuel for SSES Unit 1. On line stability measurements at the SSES and Grand Gulf 1 reactors have demonstrated that a single reload of ANF 9X9 fuel has little impact on the overall core stability. The licensee has previously implemented approved surveillance Technical Specifications for detecting and suppressing power oscillations in regions of the power-flow map considered susceptible to potential instability. Extended operation in the single loop operation mode is not presently permitted for SSES Unit 1. Based on these considerations we conclude that acceptable THS provisions have been made for the proposed one-third core reload with the ANF 9X9 fuel in S1C4.

2.3 TRANSIENT AND ACCIDENT ANALYSES

Minimum Critical Power Ratio Safety Limit

The minimum critical power ratio (MCPR) safety limit for the Cycle 4 reload was determined by the licensee to be 1.06 for all fuel types. The methodology for Cycle 4 is based on ANF revised critical power methodology in XN-NF-524, Revision 1, which incorporates a constant flow MCPR formulation for BWR applications. The staff has completed its generic review of XN-NF-524 and has concluded that the methodology for arriving at an MCPR safety limit is acceptable. The XN-3 correlation used to develop the MCPR safety limit has been approved for the new 9X9 fuel type. The methodology of XN-NF-524, Revision 1 was applied generically for the upcoming Cycle 4 and is considered applicable to the resident GE 8X8 fuel as well as the ANF fuel. The staff has verified through its review of the S1C4 transient analysis report XN-NF-87-22 that the methodology for determining uncertainties and the application in determining the MCPR safety limit is in accordance with NRC approved methodology and is acceptable.

Operational Transients

Various operational transients could reduce the MCPR below the intended safety limit. The most limiting transients have been analyzed to determine which event could potentially induce the largest reduction (delta-CPR) in the initial critical power ratio. The ANF transient methodology is basically the same as that used and approved for recent plant reloads with ANF 9X9 fuel. Certain aspects of the methodology as identified in the following discussion have received more recent NRC approval.

ANF examined the standard transient events and the S1C4 Transient Analysis and presented the results for the more limiting events. The most limiting core wide transients were the Load Rejection Without Bypass (LRWB) and the Feedwater Controller Failure (FWCF). The events were analyzed at the rated condition (104% power/100% flow) and with End-of-Cycle Recirculation Pump Trip (EOC-RPT) operable. The additional aspect of the ANF plant transient methodology recently approved by the staff is the XCOBRA-T code which is used in the determination of the thermal margin for the transients. The analyses were all done with approved methodologies and the results are acceptable. The calculated delta-CPR for the LWRB is equal to 0.22. The resulting MCPR operating limit of 1.28 is acceptable for incorporation into the S1C4 Technical Specifications for all fuel types.

It was assumed for these transients that the RPT is operable. The limiting MCPR event (LRWB) was also calculated for limiting extension conditions assuming an inoperable RPT. This resulted in increased MCPR limits which are also proposed for S1C4. These calculations follow standard procedures for the inoperable RPT extension and operation whithin these limits is acceptable for S1C4.

Compliance with overpressurization criteria was demonstrated by analysis of Main Steam Isolation Valve (MSIV) closure with MSIV position switch failure. Six safety-relief valves were assumed out of service. Maximum pressure was 105 percent of vessel design pressure, well under the 110 percent criterion. The calculation was done with approved methodology and results are acceptable.

The LOCA analyses for SSES Unit 2 Cycle 2 performed for a full core of ANF 9X9 fuel is applicable for the S1C4 residual and reload ANF fuel. These analyses have covered an acceptable range of conditions, have been performed with approved methodology, and the resulting Technical Specification MAPLHGR values for the ANF fuel remain acceptable.

Reactivity Insertion Transients

The control rod withdrawal error, the fuel loading error and the rod drop accident were evaluated for Cycle 4. The licensee used methods described in XN-NF-80-19, Volume 4. Using a Rod Block Monitor setting of 108 percent of full power results in a delta-CPR of 0.18 for the control rod withdrawal error transient for 9X9 fuel. The change in CPR due to a fuel loading error is 0.08. These values are comparable to previous reloads and are not limiting.

The rod drop accident was analyzed with approved ANF methodology. The resulting maximum fuel enthalpy of 91 cal/gm is within the established limit of 280 cal/gm. The staff finds that the analysis and results are acceptable.

2.4 TECHNICAL SPECIFICATION CHANGES

The following Susquehanna Steam Electric Station Unit 1 Technical Specification changes have been proposed for operation during reload Cycle 4:

(1) DEFINITIONS pages 1-2 and 1-3, parts of Bases pages B 2-1 an B 2-2, Limiting Conditions for Operating (LCO) pages 3/4 2-1, 3/4 2-10a and 3/4 4-1c, Figure 3.2.1-2, Bases pages B 3/4 1-1, B 3/4 2-1 and B 3/4 4-1, and Design Features page 5-6:

Changes were made to reflect the corporate change from Exxon Nuclear Company (ENC) to Advanced Nuclear Fuels (ANF) Corporation, to identify and describe the new fuel design and to incorporate editorial changes. These changes are administrative only with no safety significance and are therefore acceptable.

(2) Bases pages B 2-1 and B 2-2, Section 2.1.1 - THERMAL POWER, Low Pressure or Low Flow:

The changes provide a basis for the range of validity for use of the critical heat flux correlation for the reload 9X9 fuel type. The basis was approved as part of a generic review and is acceptable.

(3) Figure 3.2.1-3:

The MAPLHGR limits for the new fuel are added. This addition is acceptable.

(4) LCO page 3/4 2-5 and Figure 3.2.2-1:

For ANF fuel, the LHGR Limits and LCO for APRM setpoints are based on a generic review and approval and are acceptable.

(5) Figure 3.2.3-1 and 3.2.3-2 and LCO page 3/4 2-10a:

These figures reflect the new MCPR limits for Cycle 4 and are acceptable.

(6) LCO page 3/4 2-10c and Figures 3.2.4.2-1 and 3.2.4.2-2:

For ANF fuel, the LHGR limits are based on a generic review and approval and are acceptable.

The identified changes provided in the licensee's submittal are acceptable as proposed.

2.5 Restrictions

We have reviewed the reports submitted for the Cycle 4 operation of SSES Unit 1. Based on this review we conclude that appropriate material was submitted and that the fuel design, nuclear design, thermal-hydraulic design and transient and accident analyses are acceptable. Sufficient basis has been provided to allow the addition of 240 ANF 9X9 fuel bundles in the SSES Unit 1 core. The Technical Specification changes submitted for this reload suitably reflect the necessary modifications for operation in this cycle.

Our review as discussed in the evaluation sections above has identified certain restrictions relating to our incomplete review of the ANF 9X9 fuel. The approval of the ANF 9X9 fuel is therefore limited to the upcoming Cycle 4 only. Specifically, the approval of extended exposure limits for the 9X9 fuel beyond 30,000 MWD/MTU batch average exposure for future operating cycles is contingent upon our approval of XN-NF-82-06(P) and Supplements 1, 2, 4, and 5. Also, approval of the additional rod bow considerations is required for the ANF 9X9 fuel for exposure beyond 23,000 MWD/MTU expected to occur in the future cycles.

3.0 ENVIRONMENTAL CONSIDERATION

This amendment involves changes to requirements with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement nor environmental assessment need be prepared in connection with the issuance of this amendment.

4.0 CONCLUSION

The Commission made a proposed determination that the amendment involves no significant hazards consideration which was published in the Federal Register (52 FR 26593) on July 15, 1987 and consulted with the State of Pennsylvania. No public comments were received, and the State of Pennsylvania did not have any comments.

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security nor to the health and safety of the public.

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