May 31, 1990

Docket No. 50-397

DISTRIBUTION

Docket File
NRC & LPDRs
PD5 Reading
LA
JZwolinski
RSamworth
GPA/PA
OGC
DHagan
EJordan
FRosa
GHill (4)
Wanda Jones
JCalvo
ACRS (10)
ARM/LFMB
Region V (4)
FBurrows

Mr. G. C. Sorensen, Manager
Regulatory Programs
Washington Public Power Supply System
P.O. Box 968
3000 George Washington Way
Richland, Washington 99352

Dear Mr. Sorensen:

SUBJECT: ISSUANCE OF AMENDMENT NO. 84 TO FACILITY OPERATING LICENSE NO. NPF-21 - WPPSS NUCLEAR PROJECT NO. 2 (TAC NO. 76071)

The U.S. Nuclear Regulatory Commission has issued the enclosed amendment to Facility Operating License NPF-21 to the Washington Public Power Supply System for WPPSS Nuclear Project No. 2, located in Benton County near Richland, Washington. This amendment is in response to your letter dated February 27, 1990 (G02-90-032) as supplemented by letter dated April 13, 1990 (G02-90-075).

This amendment revises Technical Specifications 3/4.2.1, "Average Linear Heat Generation Rate," 3/4.2.3, "Minimum Critical Power Ratio," and 3/4.2.4, "Linear Heat Generation Rate" to effect limiting conditions of operation, action statements and surveillance requirements applicable to the sixth cycle of operation. Changes are also made to include limitations applicable to the four General Electric and four Asea Brown Boveri Atom lead fuel assemblies to be placed in the core for the sixth cycle.

A copy of the related safety evaluation supporting the amendment is enclosed. A Notice of Issuance will be included in the Commission's bi-weekly Federal Register notice.

Sincerely,

original signed by

Robert B. Samworth, Senior Project Manager
Project Directorate V
Division of Reactor Projects - III,
IV, V and Special Projects
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 84 to Facility Operating License No. NPF-21
2. Safety Evaluation

cc w/enclosures:
See next page

DRSP/PD5	DRSP/PD5	OGC	(A)DRSP/D:PD5
EHylton	RSamworth:dr		JLarkins
5/ /90	5/10/90	5/24/90	5/31/90

[WNP2 AM76071]

9006080247 900531
PDR ADOCK 05000397
P PDC

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

May 31, 1990

Docket No. 50-397

Mr. G. C. Sorensen, Manager
Regulatory Programs
Washington Public Power Supply System
P.O. Box 968
3000 George Washington Way
Richland, Washington 99352

Dear Mr. Sorensen:

SUBJECT: ISSUANCE OF AMENDMENT NO. 84 TO FACILITY OPERATING LICENSE NO. NPF-21 - WPPSS NUCLEAR PROJECT NO. 2 (TAC NO. 76071)

The U.S. Nuclear Regulatory Commission has issued the enclosed amendment to Facility Operating License NPF-21 to the Washington Public Power Supply System for WPPSS Nuclear Project No. 2, located in Benton County near Richland, Washington. This amendment is in response to your letter dated February 27, 1990 (G02-90-032) as supplemented by letter dated April 13, 1990 (G02-90-075).

This amendment revises Technical Specifications 3/4.2.1, "Average Linear Heat Generation Rate," 3/4.2.3, "Minimum Critical Power Ratio," and 3/4.2.4, "Linear Heat Generation Rate" to effect limiting conditions of operation, action statements and surveillance requirements applicable to the sixth cycle of operation. Changes are also made to include limitations applicable to the four General Electric and four Asea Brown Boveri Atom lead fuel assemblies to be placed in the core for the sixth cycle.

A copy of the related safety evaluation supporting the amendment is enclosed. A Notice of Issuance will be included in the Commission's bi-weekly Federal Register notice.

Sincerely,

Robert B. Samworth

Robert B. Samworth, Senior Project Manager
Project Directorate V
Division of Reactor Projects - III,
IV, V and Special Projects
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 84 to Facility Operating License No. NPF-21
2. Safety Evaluation

cc w/enclosures:
See next page

Mr. G. C. Sorensen

WPPSS Nuclear Project No. 2
(WNP-2)

cc:
Mr. C. M. Powers
WNP-2 Plant Manager
Washington Public Power Supply System
P.O. Box 968, MD 927M
Richland, Washington 99352

Mr. G. E. Doupe, Esquire
Washington Public Power Supply System
P. O. Box 968
3000 George Washington Way
Richland, Washington 99532

Mr. Curtis Eschels, Chairman
Energy Facility Site Evaluation Council
Mail Stop PY-11
Olympia, Washington 98504

Mr. Alan G. Hosler, Licensing Manager
Washington Public Power Supply System
P. O. Box 968, MD 956B
Richland, Washington 99352

Mr. A. Lee Oxsen
Assistant Managing Director for Operations
Washington Public Power Supply System
P. O. Box 968, MD 1023
Richland, Washington 99352

Mr. Gary D. Bouchey, Director
Licensing and Assurance
Washington Public Power Supply System
P. O. Box 968, MD 280
Richland, Washington 99352

Regional Administrator, Region V
U.S. Nuclear Regulatory Commission
1450 Maria Lane, Suite 210
Walnut Creek, California 94596

Chairman
Benton County Board of Commissioners
Prosser, Washington 99350

Mr. Christian Bosted
U. S. Nuclear Regulatory Commission
P. O. Box 69
Richland, Washington 99352

Nicholas S. Reynolds, Esq.
Bishop, Cook, Purcell
& Reynolds
1400 L Street NW
Washington, D.C. 20005-3502

(10)

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

WASHINGTON PUBLIC POWER SUPPLY SYSTEM

DOCKET NO. 50-397

NUCLEAR PROJECT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 84
License No. NPF-21

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

   A. The application for amendment filed by the Washington Public Power Supply System (the licensee), dated February 27, 1990 and supplemented by letter dated April 13, 1990 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 Facility Operating License No. NPF-21 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. 84, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This amendment is effective as of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John T. Larkins

John T. Larkins, Acting Director
Project Directorate V
Division of Reactor Projects - III,
IV, V and Special Projects
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 31, 1990

ATTACHMENT TO LICENSE AMENDMENT NO. ______

FACILITY OPERATING LICENSE NO. NPF-21

DOCKET NO. 50-397

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the areas of change. Also to be replaced are the following overleaf pages.

AMENDMENT PAGE	OVERLEAF PAGE
xx	xix
xx(a)	--
1-2	1-1
B2-1	--
B2-2*	--
3/4 2-1	3/4 2-2
3/4 2-4D	3/4 2-4C
3/4 2-4E	--
3/4 2-7	--
3/4 2-7A	--
3/4 2-8	--
3/4 2-9	3/4 2-10
3/4 2-10C	--
3/4 2-10D	--
B3/4 2-1	B3/4 2-2
B3/4 2-3	--
B3/4 2-4	--
B3/4 2-5*	--
B3/4 2-6*	--
5-5	5-6

*The text on these pages has been shifted but no change is made to the content of the text.

INDEX

ADMINISTRATIVE CONTROLS

SECTION	PAGE
CORPORATE NUCLEAR SAFETY REVIEW BOARD (Continued)	
CONSULTANTS	6-11
MEETING FREQUENCY	6-11
QUORUM	6-11
REVIEW	6-11
AUDITS	6-12
RECORDS	6-13
6.6 REPORTABLE OCCURRENCE ACTION	6-13
6.7 SAFETY LIMIT VIOLATION	6-14
6.8 PROCEDURES AND PROGRAMS	6-14
6.9 REPORTING REQUIREMENTS	6-16
6.9.1 ROUTINE REPORTS AND REPORTABLE OCCURRENCES	6-16
STARTUP REPORT	6-16
ANNUAL REPORTS	6-16
MONTHLY OPERATING REPORTS	6-17
REPORTABLE OCCURENCES	6-17
PROMPT NOTIFICATION WITH WRITTEN FOLLOWUP	6-17
THIRTY DAY WRITTEN REPORTS	6-19
ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT	6-20
SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT	6-21
6.9.2 SPECIAL REPORTS	6-22
6.10 RECORD RETENTION	6-22
6.11 RADIATION PROTECTION PROGRAM	6-24
6.12 HIGH RADIATION AREA	6-24
6.13 PROCESS CONTROL PROGRAM	6-25
6.14 OFFSITE DOSE CALCULATION MANUAL	6-25
6.15 MAJOR CHANGES TO RADIOACTIVE LIQUID, GASEOUS, AND SOLID WASTE TREATMENT SYSTEMS	6-26

LIST OF FIGURES

FIGURE		PAGE
3.1.5-1	SODIUM PENTABORATE SOLUTION SATURATION TEMPERATURE...	3/4 1-21
3.1.5-2	SODIUM PENTABORATE TANK, VOLUME VERSUS CONCENTRATION REQUIREMENTS..	3/4 1-22
3.2.1-1	MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS AVERAGE PLANAR EXPOSURE, INITIAL CORE FUEL TYPE 8CR183........................	3/4 2-2
3.2.1-2	MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS AVERAGE PLANAR EXPOSURE, INITIAL CORE FUEL TYPE 8CR233........................	3/4 2-3
3.2.1-3	MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS AVERAGE BUNDLE EXPOSURE ANF 8x8 RELOAD FUEL..................................	3/4 2-4
3.2.1-4	MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS AVERAGE PLANAR EXPOSURE, INITIAL CORE FUEL TYPE 8CR183..............................	3/4 2-4A
3.2.1-5	MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS AVERAGE PLANAR EXPOSURE, INITIAL CORE FUEL TYPE 8CR233..............................	3/4 2-4B
3.2.1-6	MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS AVERAGE BUNDLE EXPOSURE ANF 9x9-IX AND 9x9-9X FUEL.............................	3/4 2-4C
3.2.1-7	MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS BUNDLE AVERAGE EXPOSURE SVEA-96 LEAD FUEL ASSEMBLIES..................................	3/4 2-4D
3.2.1-8	MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS BUNDLE AVERAGE EXPOSURE GE11 LEAD FUEL ASSEMBLIES.............................	3/4 2-4E
3.2.3-1	REDUCED FLOW MCPR OPERATING LIMIT....................	3/4 2-8
3.2.4-1	LINEAR HEAT GENERATION RATE (LHGR) LIMIT VERSUS AVERAGE PLANAR EXPOSURE ANF 8x8 RELOAD FUEL..........	3/4 2-10
3.2.4-2	LINEAR HEAT GENERATION RATE (LHGR) LIMIT VERSUS AVERAGE PLANAR EXPOSURE ANF 9x9-IX FUEL..............	3/4/2-10A
3.2.4-3	LINEAR HEAT GENERATION RATE (LHGR) LIMIT VERSUS AVERAGE PLANAR EXPOSURE ANF 9x9-9X FUEL..............	3/4 2-10B
3.2.4-4	LINEAR HEAT GENERATION RATE (LHGR) LIMIT VERSUS AVERAGE PLANAR EXPOSURE SVEA-96 LEAD FUEL ASSEMBLIES..	3/4 2-10C

LIST OF FIGURES

FIGURE		PAGE
3.2.4-5	LINEAR HEAT GENERATION RATE (LHGR) LIMIT VERSUS AVERAGE PLANAR EXPOSURE GE11 LEAD FUEL ASSEMBLIES	3/4 2-10D
3.2.6-1	OPERATING REGION LIMITS OF SPEC. 3.2.6	3/4 2-12
3.2.7-1	OPERATING REGION LIMITS OF SPEC. 3.2.7	3/4 2-14
3.2.8-1	OPERATING REGION LIMITS OF SPEC. 3.2.8	3/4 2-16
3.4.1.1-1	THERMAL POWER LIMITS OF SPEC. 3.4.1.1-1	3/4 4-3a
3.4.6.1-1	MINIMUM REACTOR VESSEL METAL TEMPERATURE VERSUS REACTOR VESSEL PRESSURE (INITIAL VALUES)	3/4 4-20
3.4.6.1-2	MINIMUM REACTOR VESSEL METAL TEMPERATURE VERSUS REACTOR VESSEL PRESSURE (OPERATIONAL VALUES)	3/4 4-21
4.7-1	SAMPLE PLAN 2) FOR SNUBBER FUNCTIONAL TEST	3/4 7-15
3.9.7-1	HEIGHT ABOVE SFP WATER LEVEL VS. MAXIMUM LOAD TO BE CARRIED OVER SFP	3/4 9-10
B 3/4 3-1	REACTOR VESSEL WATER LEVEL	B 3/4 3-8
B 3/4.4.6-1	FAST NEUTRON FLUENCE (E>1MeV) AT 1/4 T AS A FUNCTION OF SERVICE LIFE	B 3/4 4-7
5.1-1	EXCLUSION AREA BOUNDARY	5-2
5.1-2	LOW POPULATION ZONE	5-3
5.1-3	UNRESTRICTED AREAS AND SITE BOUNDARY FOR RADIOACTIVE GASEOUS AND LIQUID EFFLUENTS	5-4

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 BUNDLE EXPOSURE

1.2 The AVERAGE BUNDLE EXPOSURE is 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 bundle.

AVERAGE PLANAR EXPOSURE

1.3 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

1.4 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

1.5 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

1.6 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

1.7 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

1.8 CORE ALTERATION shall be the addition, removal, relocation or movement of fuel, sources, incore instruments or reactivity controls within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of the movement of a component to a safe conservative position.

CRITICAL POWER RATIO

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

DOSE EQUIVALENT I-131

1.10 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

1.11 $\bar{E}$ shall be the average, weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling, of the sum of the average beta and gamma energies per disintegration, in MeV, for isotopes, 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

1.12 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.

2.0 SAFETY LIMITS and LIMITING SAFETY SYSTEM SETTINGS

BASES

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 1.06 for two recirculation loop operation and 1.07 for single recirculation loop operation for all nuclear fuel in WNP-2. MCPR greater than 1.06 for two recirculation loop operation and 1.07 for single recirculation loop operation 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 integrity 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 percent of the fuel rods in the core do not experience transition boiling (Reference: XN-NF-524(A), Rev. 1; ABB Atom Report UK90-126; GE11 Lead Fuel Assembly Report for Washington Public Power Supply System Nuclear Project No. 2, Reload 5, Cycle 6). The latter two references support application of the above established safety limit to GE11 and SVEA-96 LFA fuel in WNP-2.

2.1 SAFETY LIMITS

2.1.1. THERMAL POWER, Low Pressure or Low Flow

For certain conditions of pressure and flow, the XN-3 correlation is not valid for all critical power calculations. The XN-3 correlation is not valid for bundle mass velocities less than .25 x 10^6 lbs/hr-ft^2 or pressures less than 585 psig. Therefore, the fuel cladding integrity Safety Limit is established by other means. This is done by establishing a limiting condition on core THERMAL POWER with the following basis. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be greater than 4.5 psi. Analyses show that with a bundle flow of 28 x 10^3 lbs/h (approximately a mass velocity of .25 x 10^6 lbs/hr-ft^2), bundle pressure drop is nearly independent of bundle power

THERMAL POWER, Low Pressure or Low Flow (Continued)

and has a value of 3.5 psi. Thus, the bundle flow with a 4.5 psi driving head will be greater than 28 x 10^3 lbs/h. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors, this corresponds to a THERMAL POWER of more than 50% of RATED THERMAL POWER. Thus, a THERMAL POWER limit of 25% of RATED THERMAL POWER for reactor pressure below 585 psig is conservative.

2.1.2 THERMAL POWER, High Pressure and High Flow

The fuel cladding integrity Safety Limit is set such that no fuel damage is calculated to occur if the limit is not violated. Since the parameters which result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions resulting in a departure from nucleate boiling have been used to mark the beginning of the region where fuel damage could occur. Although it is recognized that a departure from nucleate boiling would not necessarily result in damage to BWR fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. However, the uncertainties in monitoring the core operating state and in the procedures used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore, the fuel cladding integrity Safety Limit is defined as the CPR in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition considering the power distribution within the core and all uncertainties.

The Safety Limit MCPR is determined using the ANF Critical Power Methodology for boiling water reactors[a] which is a statistical model that combines all of the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the ANF nuclear critical heat fluxenthalpy XN-3 correlation. The XN-3 correlation is valid over the range of conditions used in the tests of the data used to develop the correlation.

The required input to the statistical model are the uncertainties listed in Bases Table B2.1.2-1.

The bases for the uncertainties in the core parameters are given in XN-NF-524(A), Rev. 1[a] and the basis for the uncertainty in the XN-3 correlation is given in XN-NF-512(A), Rev. 1[b]. The power distribution is based on a typical 764 assembly core in which the rod pattern was arbitrarily chosen to produce a skewed power distribution having the greatest number of assemblies at the highest power levels. The worst distribution during any fuel cycle would not be as severe as the distribution used in the analysis.

a. Exxon Nuclear Critical Power Methodology for Boiling Water Reactors, XN-NF-524(A), Rev. 1.

b. Exxon Nuclear Company XN-3 Critical Power Correlation, XN-NF-512(A), Rev. 1.

3/4.2 POWER DISTRIBUTION LIMITS

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 initial core fuel and average bundle exposure for ANF, SVEA-96 and GE11 LFA fuel shall not exceed the limits shown in Figures 3.2.1-1, 3.2.1-2, 3.2.1-3, 3.2.1-6, 3.2.1-7 and 3.2.1-8 when in two loop operation, and Figures 3.2.1-3, 3.2.1-4, 3.2.1-5, 3.2.1-6, 3.2.1-7 and 3.2.1-8 when in single loop operation.

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, 3.2.1-3 3.2.1-6, 3.2.1-7 or 3.2.1-8 in two loop operation or Figure 3.2.1-3, 3.2.1-4, 3.2.1-5, 3.2.1-6, 3.2.1-7 or 3.2.1-8 in single loop operation, 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.

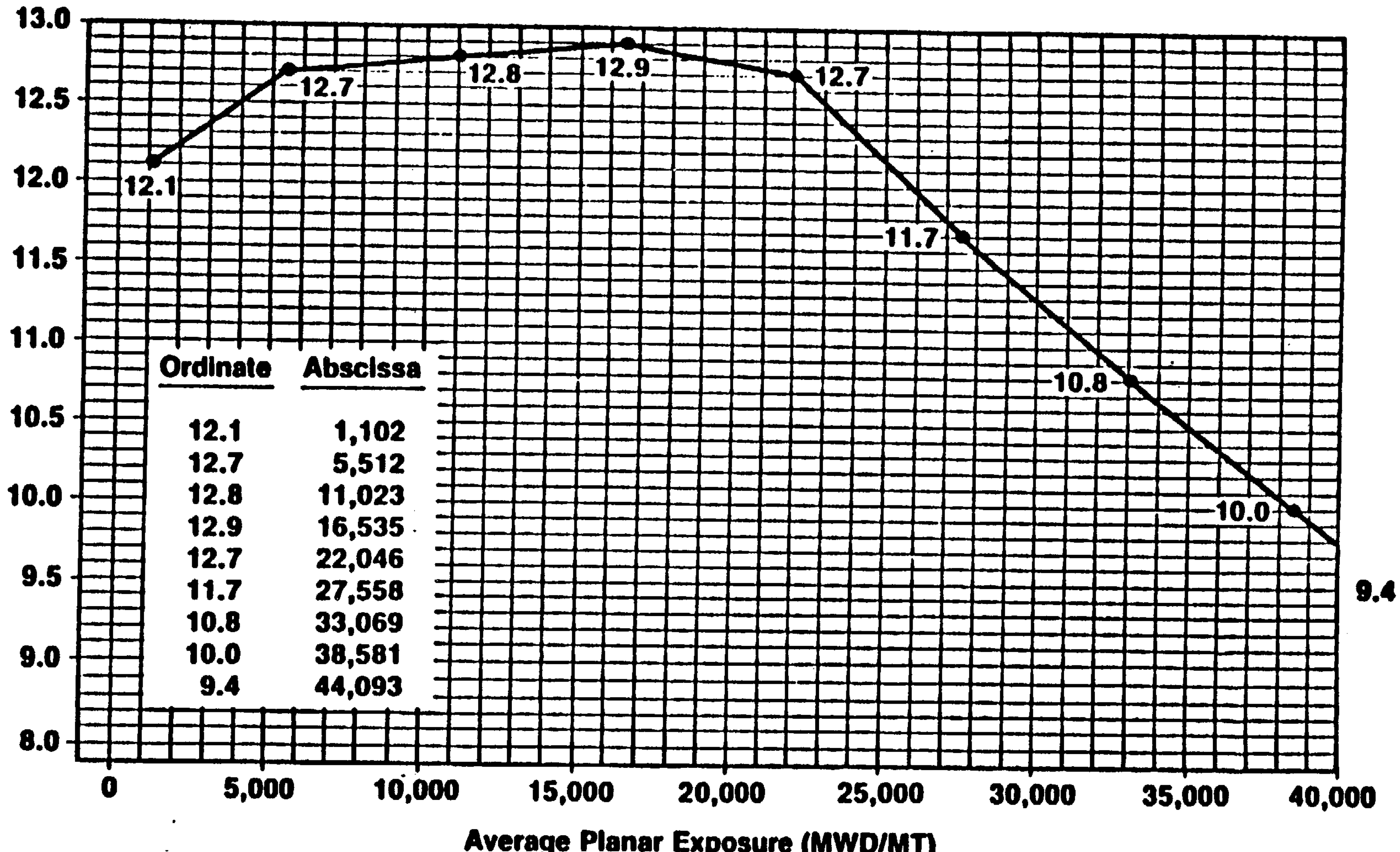
SURVEILLANCE REQUIREMENTS

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, 3.2.1-3, 3.2.1-4, 3.2.1-5, 3.2.1-6, 3.2.1-7 and 3.2.1-8.

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.

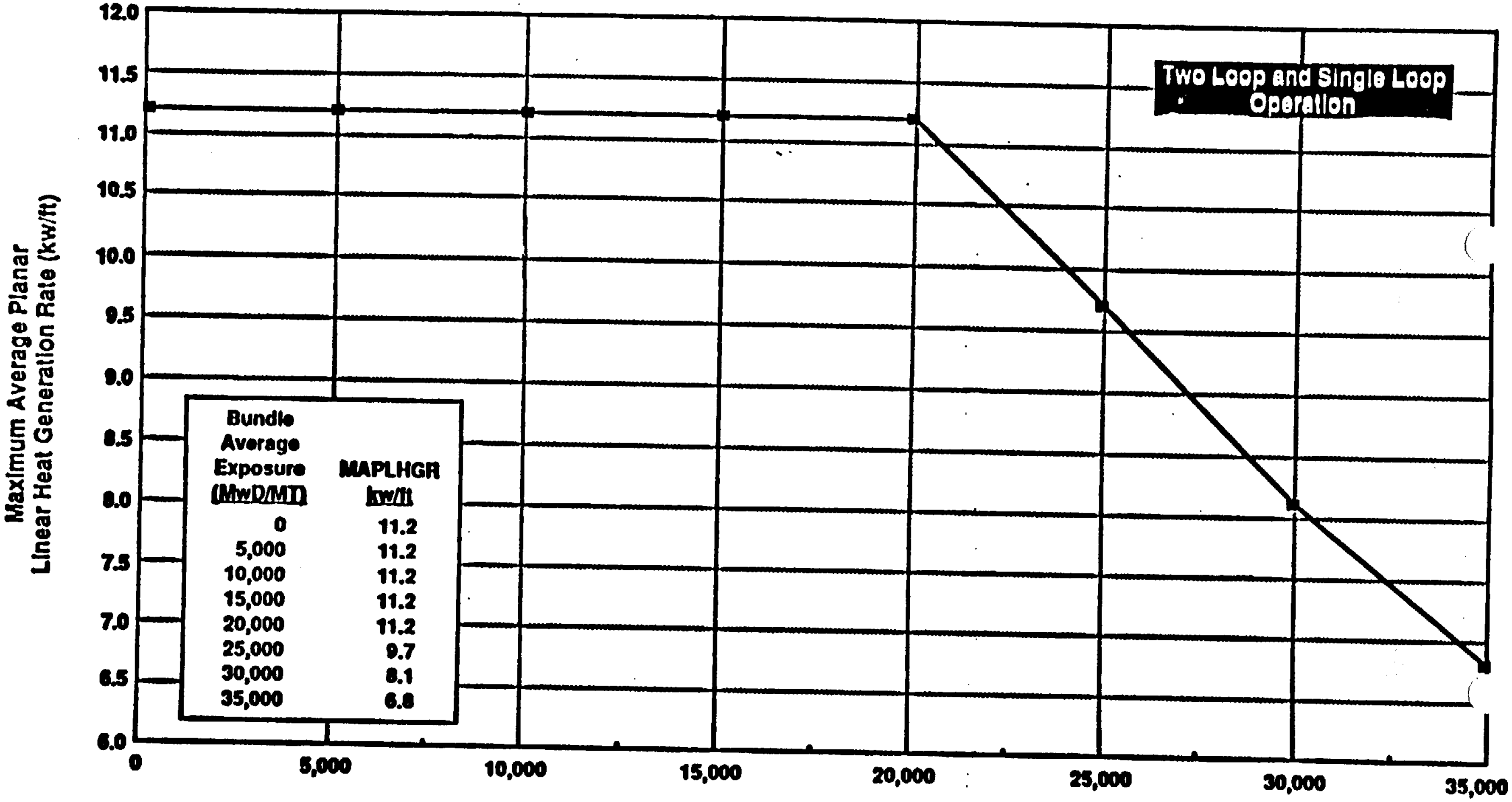


Ordinate	Abscissa
12.1	1,102
12.7	5,512
12.8	11,023
12.9	16,535
12.7	22,046
11.7	27,558
10.8	33,069
10.0	38,581
9.4	44,093

Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) Versus Average Planar Exposure Initial Core Fuel Type 8CR183

Figure 3.2.1-1

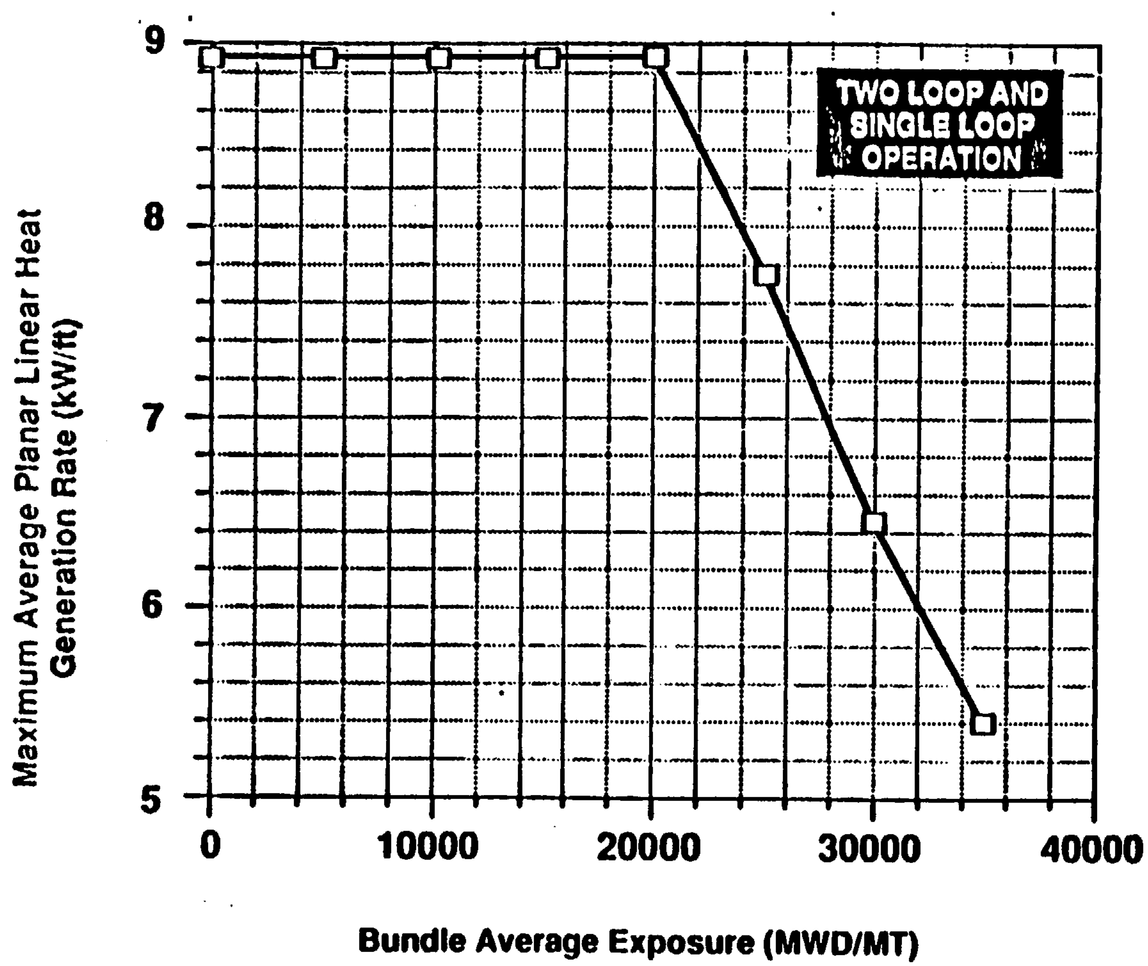
860599.4A



Bundle Average Exposure (MwD/MT)	MAPLHGR kw/ft
0	11.2
5,000	11.2
10,000	11.2
15,000	11.2
20,000	11.2
25,000	9.7
30,000	8.1
35,000	6.8

Bundle Average Exposure (MWD/MT)
ANF 9 X 9 - IX AND 9 X 9 - 9X Reload Fuel
Maximum Average Planar Linear Heat
Generation Rate (MAPLHGR) Versus
Bundle Average Exposure

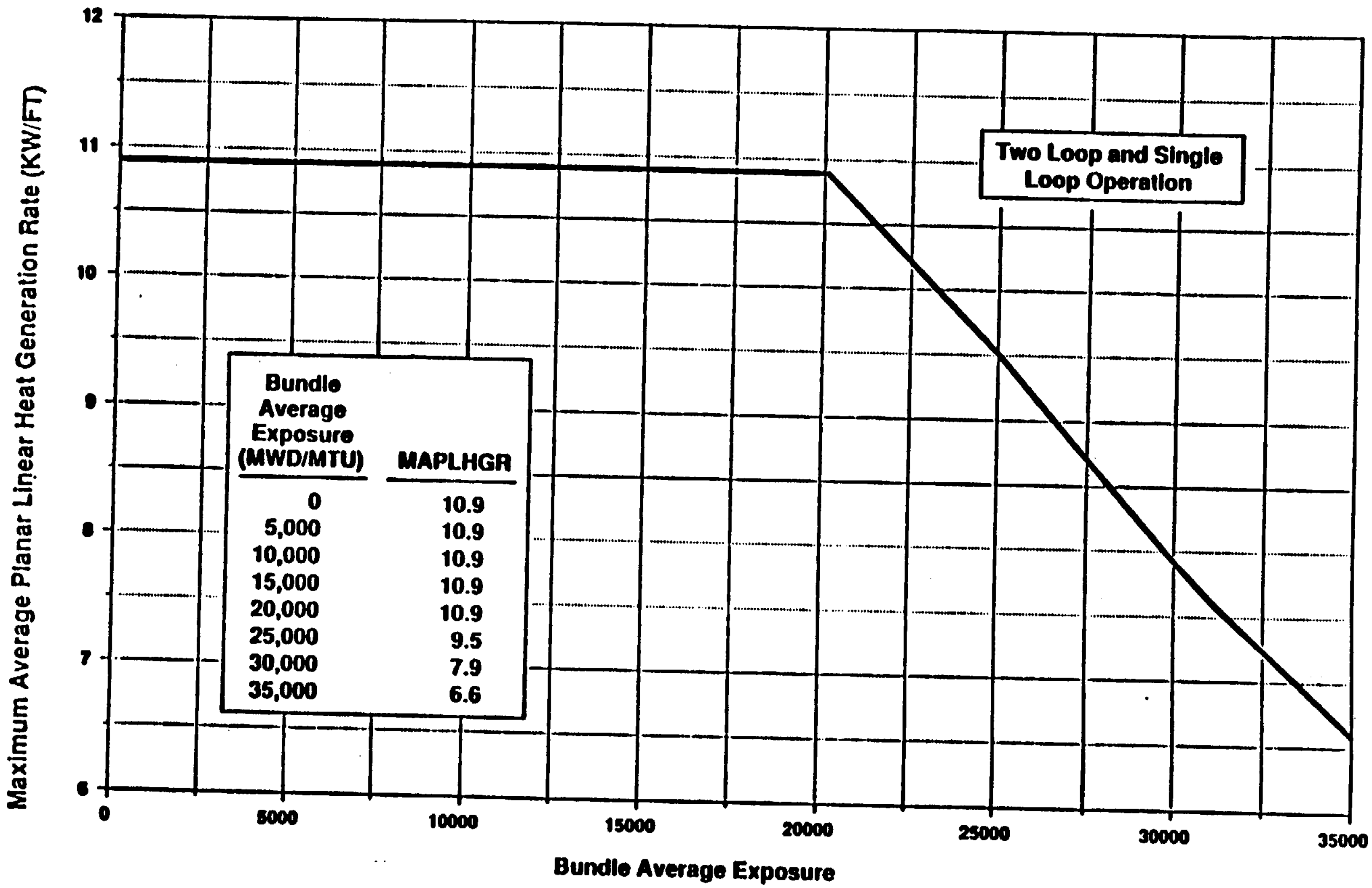
Figure 3.2.1-6



Bundle Average Exposure (MWD/MTU)	MAPLHGR (kW/ft)
0	8.90
5,000	8.90
10,000	8.90
15,000	8.90
20,000	8.90
25,000	7.74
30,000	6.44
35,000	5.41

Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) Versus Bundle Average Exposure SVEA-96 Lead Fuel Assemblies

Figure 3.2.1-7



Bundle Average Exposure (MWD/MTU)	MAPLHGR
0	10.9
5,000	10.9
10,000	10.9
15,000	10.9
20,000	10.9
25,000	9.5
30,000	7.9
35,000	6.6

Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) Versus Bundle Average Exposure GE 11 Lead Fuel Assemblies

Figure 3.2.1-8

Table 3.2.3-1
MCPR OPERATING LIMITS

	Cycle Exposure	Equipment Status	MCPR Operating Limit Up to 106% Core Flow: 8x8 ANF Fuel***	MCPR Operating Limit Up to 106% Core Flow: SVEA-96 LFA FUEL
1.	0 MWD/MTU - 3750 MWD/MTU	*	1.24	1.37
2.	3750 MWD/MTU - EOC MWD/MTU****	Normal scram times**	1.31	1.48
3.	3750 MWD/MTU - EOC MWD/MTU****	Control rod insertion bounded by Tech. Spec. limits (3.1.3.4 - p 3/4 1-8)	1.36	1.55
4.	3750 MWD/MTU - EOC MWD/MTU	RPT inoperable Normal scram times**	1.36	1.55
5.	3750 MWD/MTU - EOC MWD/MTU	RPT inoperable Control rod insertion bounded by Tech. Spec. limits (3.1.3.4 - p 3/4 1-8)	1.40	1.61
6.	0 MWD/MTU - EOC MWD/MTU	Single loop operation RPT operable Normal scram times**	1.35	1.54

*In this portion of the fuel cycle, operation with the given MCPR operating limits is allowed for both normal and Tech. Spec. scram times and for both RPT operable and inoperable.

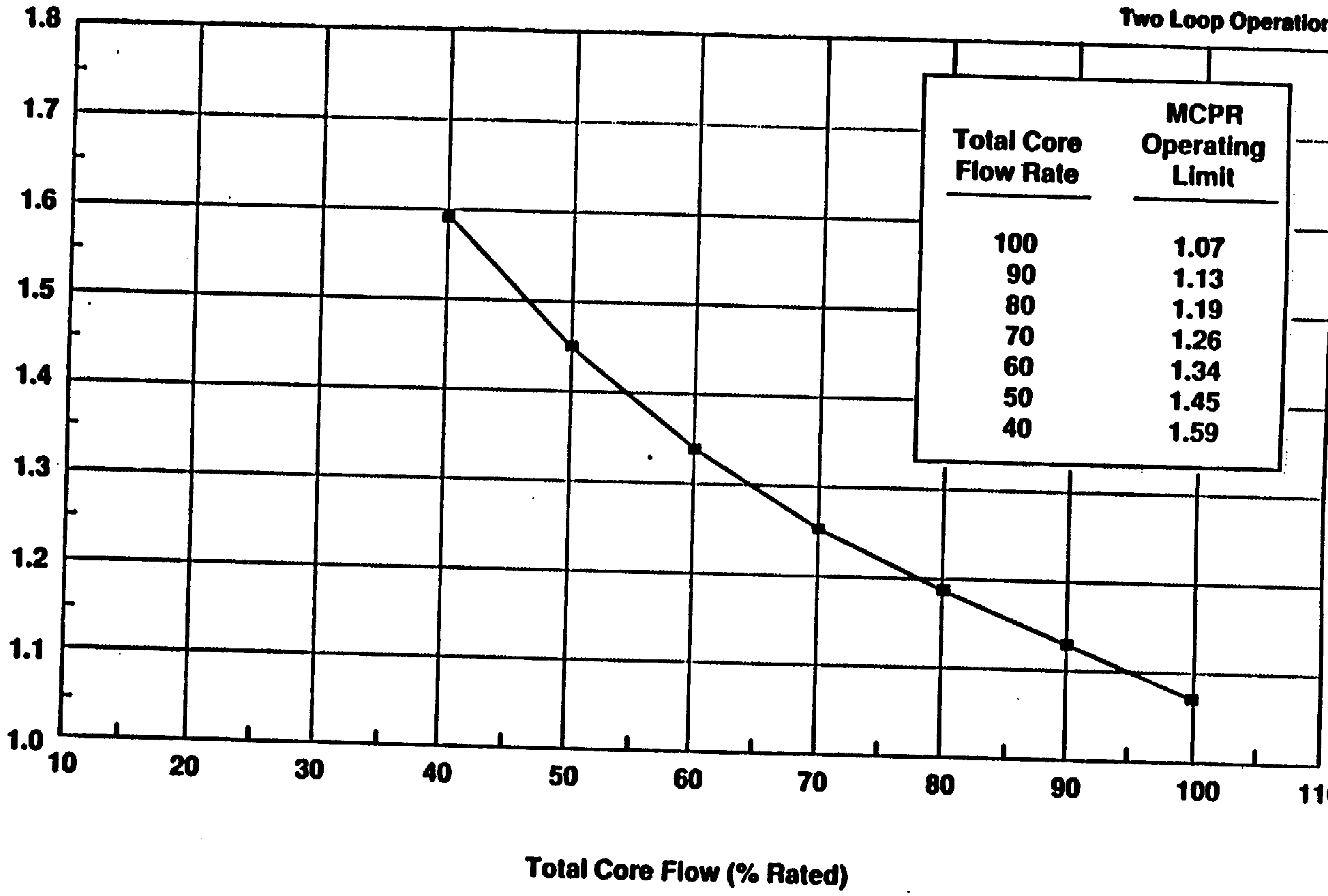
**These MCPR values are based on the ANF Reload Safety Analysis performed using the control rod insertion times shown below (defined as normal scram). In the event that surveillance 4.1.3.2 shows these scram insertion times have been exceeded, the plant thermal limits associated with normal scram times default to the values associated with Tech. Spec. scram times (3.1.3.4-p 3/4 1-8), and the scram insertion times must meet the requirements of Tech. Spec. 3.1.3.4.

Position Inserted From Fully Withdrawn	Slowest measured average control rod insertion times to specified notches for all operable control rods for each group of 4 control rods arranged in a a two-by-two array (seconds)
Notch 45	.404
Notch 39	.660
Notch 25	1.504
Notch 5	2.624

Table 3.2.3-1 (Continued)
MCPR OPERATING LIMITS

***The GE11 LFA fuel, the ANF LFA fuel and the GE initial core fuel are also monitored to the ANF 8x8 fuel MCPR Operating Limits (Reference: Power Distribution Limits, Bases, 3/4.2.3, Minimum Critical Power Ratio, p. B 3/4 2-3).

****For Final Feedwater Temperature Reduction rated conditions beyond all rods out point, add .02 to the MCPR for all fuel in the WNP-2 core except for the SVEA-96 LFA fuel. For the SVEA-96 LFA fuel, add .03 to the MCPR for Final Feedwater Temperature Reduction rated conditions beyond the all rods out point.



Reduced Flow MCPR Operating Limit
This Curve is Applicable to ANF Reload Fuel, GE Initial Core Fuel, ANF 9 X 9 LFA Fuel, GE 11 LFA Fuel, and SVEA-96 LFA Fuel
This curve is also applicable to FFTR operation
Figure 3.2.3-1

3/4.2.4 LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.4 The LINEAR HEAT GENERATION RATE (LHGR) for GE initial core fuel shall not exceed 13.4 kW/ft. The LHGR for reload fuel shall not exceed the values shown in Figures 3.2.4-1, 3.2.4-2, 3.2.4-3, 3.2.4-4 and 3.2.4-5.

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 the limit, 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.

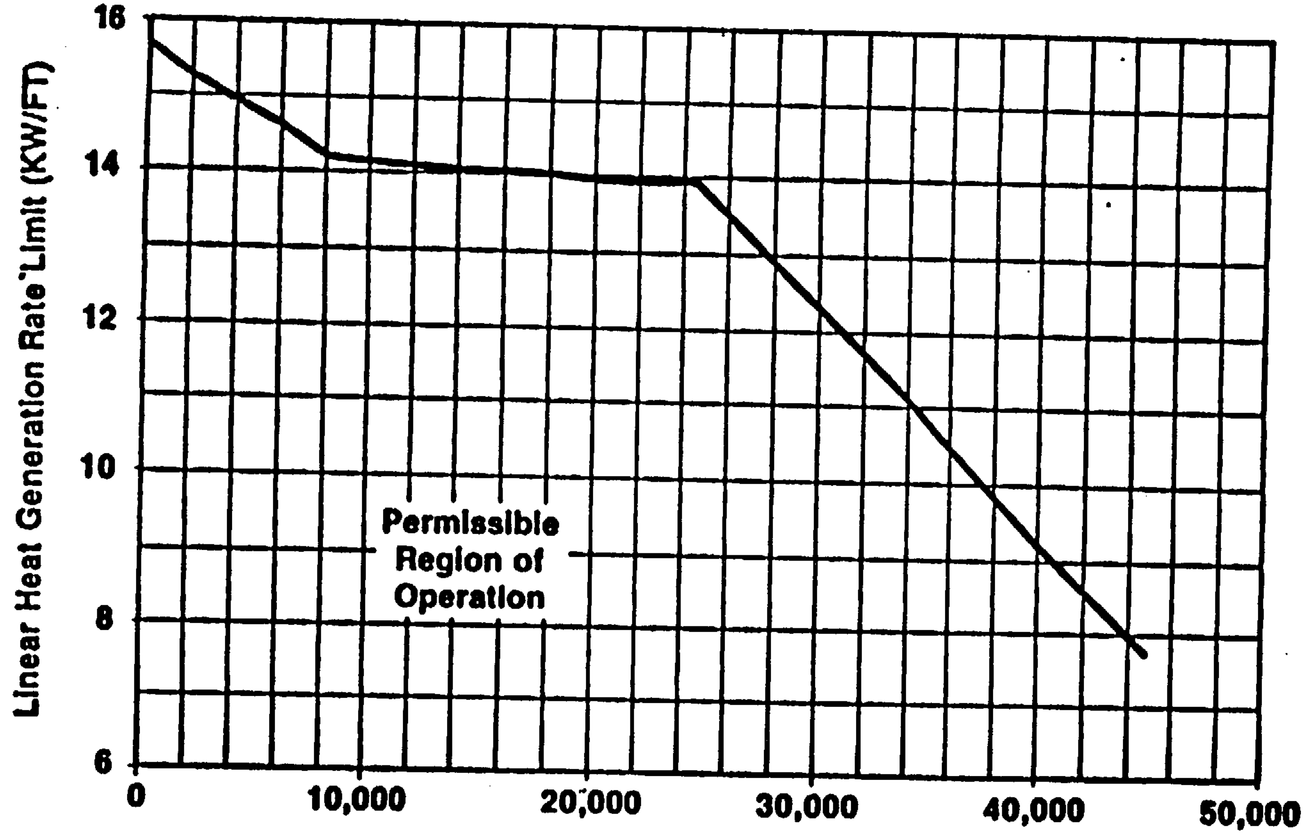
SURVEILLANCE REQUIREMENTS

4.2.4 LHGRs 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.



EXP	LHGR
0	15.62
510	15.621
2,580	15.10
5,230	14.71
7,940	14.19
10,470	14.13
13,220	14.06
15,990	14.06
18,708	14.00
21,590	13.93
24,420	13.93
27,280	13.08
30,150	12.24
33,050	11.40
35,960	10.47
38,900	9.55
41,830	8.65
44,760	7.77

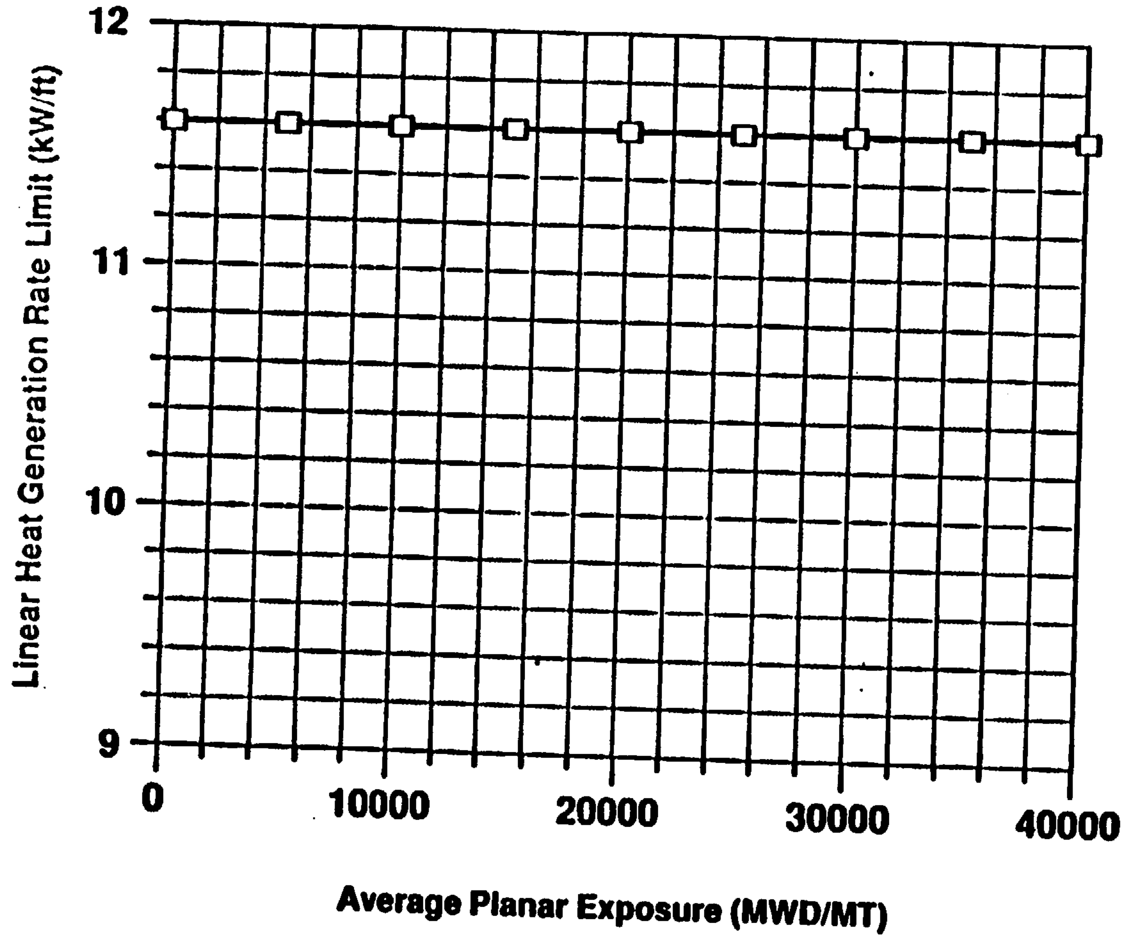
Average Planar Exposure (MWD/MT)

ANF 8x8 Reload Fuel

Linear Heat Generation Rate (LHGR) Limit

Versus Average Planar Exposure

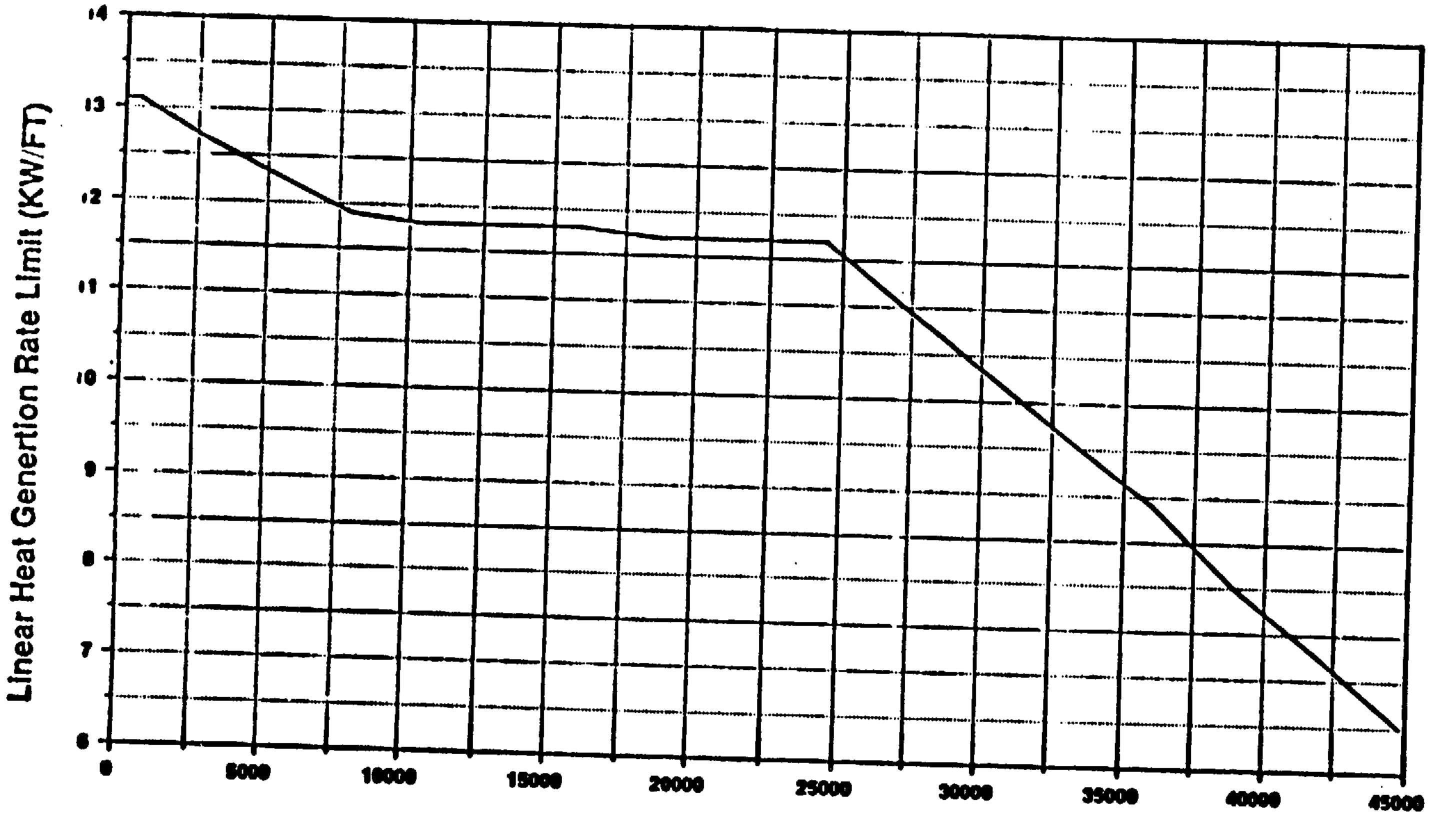
Figure 3.2.4-1



Exposure (MWD/MTU)	LHGR (kW/ft)
0 to 40,000	11.6

Linear Heat Generation Rate (LHGR) Limit Versus Average Planar Exposure
SVEA-96 Lead Fuel Assemblies

Figure 3.2.4-4



EXP	LHGR
0	13.1
510	13.1
2,580	12.7
5,230	12.3
7,940	11.9
10,470	11.8
13,220	11.8
15,990	11.8
18,708	11.7
21,590	11.7
24,420	11.7
27,280	11.0
30,150	10.3
33,050	9.6
35,960	8.9
38,900	8.0
41,830	7.3
44,760	6.5

Linear Heat Generation Rate (LHGR) Limit
Versus Average Planar Exposure
GE 11 Lead Fuel Assemblies

Figure 3.2.4-5

BASES

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

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 and 3.2.1-2 for two recirculation loop operation and Figures 3.2.1-4 and 3.2.1-5 for single loop operation. Figures 3.2.1-3, 3.2.1-6, 3.2.1-7 and 3.2.1-8 apply to both single and two loop operation.

The calculational procedure used to establish the APLHGR shown on Figures 3.2.1-1, 3.2.1-2, 3.2.1-3, 3.2.1-4, 3.2.1-5, and 3.2.1-6 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 Part 50. These models are described in NEDO-20566P or XN-NF-80-19, Volumes 2, 2A, 2B and 2C, Rev. 1. The methods for establishing the LFA APLHGR values shown in Figures 3.2.1-7 and 3.2.1-8 are given in UK90-126 and the GE11 Lead Fuel Assembly Report for Washington Public Power Supply System Nuclear Project No. 2 Reload 5 Cycle 6.

3/4.2.2 APRM SETPOINTS

The flow biased simulated thermal power-upscale scram setting and control rod block functions of the APRM instruments limit plant operations to the region covered by the transient and accident analysis. In addition, the APRM setpoints must be adjusted for both two recirculation loop operation and single recirculation loop operation to ensure that the MCPR does not become less than the fuel cladding safety limit or that > 1% plastic strain does not occur in the degraded situation. The scram settings and rod block settings are adjusted in accordance with the formula in this specification when the combination of THERMAL POWER and MFLPD indicates a higher peaked power distribution to ensure that an LHGR transient would not be increased in the degraded condition.

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 Table 3.2.3-1.

The evaluation of a given transient begins with the system initial parameters shown in the cycle specific transient analysis report that are input to an ANF 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 nonpressurization events are described in XN-NF-79-71(P) and XN-NF-84-105(A). The principal result of this evaluation is the reduction in MCPR caused by the transient.

The purpose of the $MCPR_f$ of Figure 3.2.3-1 is to define operating limits at other than rated core flow conditions. At less than 100% of rated flow the required MCPR is the maximum of the rated flow MCPR determined from Table 3.2.3-1 and the reduced flow MCPR determined from Figure 3.2.3-1, $MCPR_f$ assures that the Safety Limit MCPR will not be violated. $MCPR_f$ is only calculated for the manual flow control mode. Automatic flow control operation is not permitted.

Lead Fuel Assemblies (LFA's) from Advanced Nuclear Fuels (ANF), General Electric (GE) and ABB Atom (ABB) reside in the reactor core. Analyses performed by the three vendors indicate that the transient CPR changes for the LFA's are greater than the CPR change calculated for the dominant ANF 8x8 fuel, due primarily to the shorter thermal time constants of the smaller diameter rods. All vendors state that their LFA's have inherently higher thermal margins (by design) than the dominant 8x8 fuel. Each vendor chose to address the CPR limit in a slightly different fashion. These methods are discussed as follows.

MINIMUM CRITICAL POWER RATIO (Continued)

GE concludes that the inherent high thermal margin of the LFA's is sufficient to compensate for the larger CPR change associated with the shorter time constant and that the ANF 8x8 limits can be conservatively applied to the GE11 LFA's.

The XN-3 CHF correlation used by ANF in the analysis was developed for the ANF 8x8 fuel. A review of the correlation and comparison to CHF data obtained for the 9x9 LFA's concluded XN-3 is conservative when applied to the ANF LFA's and that the LFA's should be conservatively assumed to have a CPR performance at least equal to that calculated for an 8x8 assembly for the same power and inlet conditions. In addition, due to the water canister in the interior of the bundle, ANF modified the S-factors for the 9x9 LFA's to improve the XN-3 predictive capability. These S-factors were used in the analysis and were provided for use in monitoring the LFA's.

ABB Atom chose to take a more conservative approach and performed analyses which established conservative and unique MCPR values for the SVEA-96. The resulting MCPRs are included in the Technical Specifications.

In addition to the conservatisms discussed, the Supply System has committed to load the LFA's in core locations which have been analyzed to have sufficient margins such that the LFA's are not expected to be the limiting assemblies in the core on either a nodal or a bundle power basis. This approach is to prevent the possibility of the LFA's from ever being the limiting fuel bundle and adds additional margin to the event of a plant transient.

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 will be made at 25% of RATED THERMAL POWER level with minimum recirculation pump speed. The MCPR margin will thus be demonstrated such that future MCPR evaluation below this power level will be shown to be unnecessary. The daily requirement for calculating MCPR when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR when a limiting control rod pattern is approached ensures that MCPR will be known following a change in THERMAL POWER or power shape, regardless of magnitude, that could place operation at a thermal limit.

At EOC during FFTR, the LOAD REJECTION WITHOUT BYPASS transient is slightly more severe when compared to the same transient without FFTR, which is accounted for by an increased MCPR operating limit. The analysis conservatively reduces the feedwater temperature by 65°F and burns the produced power shape to achieve the final core conditions used in the transient analysis. This depletion causes the power peak to shift upwards, slightly increasing the time required for the normal scram to suppress the flux.

3/4.2.4 LINEAR HEAT GENERATION RATE

This specification assures that the Linear Heat Generation Rate (LHGR) in any rod is less than the design linear heat generation even if fuel pellet densification is postulated.

3/4.2.6 POWER/FLOW INSTABILITY

At the high power/low flow corner of the operating domain, a small probability of limit cycle neutron flux oscillations exists depending on combinations of operating conditions (e.g., power shape, bundle power, and bundle flow).

In February, 1984, GE issued SIL 380 addressing boiling instability and supplying several recommendations. In this SIL, the power/flow map was divided into several regions of varying concern. It also discussed the objectives and philosophy of "detect and suppress," coining the phrase.

The ANF topical report for COTRAN (XN-NF-691P) discusses boiling instability. The SER written on this topical (dated May 10, 1984) interprets the topical to require that the detect-and-suppress surveillance be used in regions which have code calculated decay ratios .75 or greater and that operation is forbidden in regions having calculated decay ratios of .9 and greater.

The NRC Generic Letter 86-02 addressed both GE and ANF (then EXXON) stability calculation methodology and stated that due to uncertainties, General Design Criteria 10 and 12 could not be met using analytic procedures on a BWR 5 design. The letter espoused GE SIL 380 and stated that General Design Criteria 10 and 12 could be met by imposing the SIL 380 recommendations in operating regions of potential instability. The NRC concluded that regions of potential instability constituted calculated decay ratios of .8 and greater by the GE methodology and .75 and greater by the EXXON methodology.

Predicated on the SIL 380 endorsement, WNP-2 has divided the power/flow map on the following boundary lines:

1. 80% rod line
2 45% core flow line
3. 100% rod line
4. Natural Circulation flow line
5. Minimum Forced Circulation for normal recirculation lineup.

This division conforms to the SIL 380 recommendations. For LCO 3.2.6, the region of concern (Region A) is bounded by the more conservative of either the 100% rodline or a line defining a calculated decay ratio of 0.9, the natural circulation flow line, and the 45% core flow line. Calculated decay ratios outside Region A must be less than 0.9. Operation in the region between the two flow lines and above the more conservative of either the 100% rodline or a line defining a calculated decay ratio of 0.9 is forbidden due to the potential for boiling instabilities.

3/4.2.7 STABILITY MONITORING - TWO LOOP OPERATION

At the high power/low flow corner of the operating domain, a small probability of limit cycle neutron flux oscillations exists depending on combinations of operating conditions (e.g., rod patterns, power shape). To provide assurance that neutron flux limit cycle oscillations are detected and suppressed, APRM and LPRM neutron flux signal decay ratios should be monitored while operating in this region.

Stability tests at operating BWRs were reviewed to determine a generic region of the power/flow map in which surveillance of neutron flux noise levels should be performed. A conservative decay ratio of 0.75 was chosen as the basis for determining the generic region for surveillance to account for the plant to plant variability of decay ratio with core and fuel designs. This generic region has been determined to correspond to a core flow of less than or equal to 45% of rated core flow and a thermal power greater than that corresponding to the 80% rodline.

Stability monitoring is performed utilizing the ANNA system. The system shall be used to monitor APRM and LPRM signal decay ratio and peak-to-peak noise values when operating in the region of concern. A minimum number of LPRM and APRM signals are required to be monitored in order to assure that both global (in-phase) and regional (out-of-phase) oscillations are detectable. Decay ratios are calculated from 30 seconds worth of data at a sample rate of 10 samples/second. This sample interval results in some inaccuracy in the decay ratio calculation, but provides rapid update in decay ratio data. A decay ratio of 0.75 is selected as a decay ratio limit for operator response such that sufficient margin to an instability occurrence is maintained. When operating in the region of applicability, decay ratio and peak-to-peak information shall be continuously calculated and displayed. A surveillance requirement to continuously monitor decay ratio and peak-to-peak noise values ensures rapid response such that changes in core conditions do not result in approaching a point of instability.

3/4.2.8 STABILITY MONITORING - SINGLE LOOP OPERATION

The basis for stability monitoring during single loop operation is consistent with that given above for two loop operation. The smaller size of the region of allowable operation, Region C, is due to a limit on the allowed flow above the 80% rodline. When operating above the 80% rodline in single loop operation, the core flow is required to be greater than 39%.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 764 fuel assemblies with each initial core fuel assembly containing 62 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 except that the reload fuel may employ a 9 x 9 array of fuel rods. Lead Fuel Assembly (LFA) designs with the same material constituents but different geometric configurations are allowed.

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 pump.
2. 1650 psig from the recirculation pump discharge to the outlet side of the discharge shutoff valve.
3. 1550 psig from the discharge shutoff valve 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 22,539 cubic feet at a nominal steam dome saturation temperature of 545°F.

5.5 METEOROLOGICAL TOWER LOCATION

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

5.6 FUEL STORAGE

CRITICALITY

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

a. A k_{eff} equivalent to less than or equal to 0.95 when flooded with unborated water, including all calculational uncertainties and biases as described in Section 9.1.2 of the FSAR.

b. A nominal 6.5-inch center-to-center distance between fuel assemblies placed in the storage racks.

5.6.1.2 The k_{eff} for new fuel for the first core loading stored dry in the spent fuel storage racks shall not exceed 0.95 when flooding with water is assumed.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 605 ft 7 in.

CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 2658 fuel assemblies.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 84 TO FACILITY OPERATING LICENSE NO. NPF-21

WASHINGTON PUBLIC POWER SUPPLY SYSTEM

NUCLEAR PROJECT NO. 2

DOCKET NO. 50-397

1.0 INTRODUCTION

By letters dated February 27, 1990 (Ref. 1) and April 13, 1990 (Ref. 2), Washington Public Power Supply System (WPPSS), the licensee, proposed to amend Facility Operating License No. NPF-21 to support Cycle 6 operation of their Nuclear Project No. 2 (WNP-2) with Advanced Nuclear Fuels Corporation (ANF) reload fuel, four General Electric Company (GE) lead fuel assemblies (LFAs), and four ABB Atom, Inc. (ABB) LFAs. The proposed amendment would revise Technical Specification (TS) operating limits and bases affected by Cycle 6 operation. In support of the Cycle 6 reload, the licensee submitted reports consisting of a reload summary (Ref. 3) and its revision (Ref. 27), the reload analysis (Ref. 4), the plant transient analysis (Ref. 5), the GE LFA analysis (Ref. 6), the ABB LFA analysis (Ref. 7), and the proposed TS changes (Ref. 8).

The reload for Cycle 6 is generally a normal reload with no unusual core features or characteristics, except for the LFAs. TS changes are primarily related to Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limits, Linear Heat Generation Rate (LHGR) limits, and Minimum Critical Power Ratio (MCPR) limits. The new reload fuel was supplied by ANF and is designated ANF-4 and ANF-5, both of which are of the 8x8C type that has been previously used in WNP-2 reloads. The reload also incorporates three optional modes of operation: (1) operation at extended core flow, (2) single loop operation (SLO), and (3) operation at a final feedwater temperature reduction (FFWTR).

The Reference 2 submittal addresses the concerns on channel box bow discussed in NRC Information Notice 89-69 (Ref. 9) and NRC Bulletin 90-02 (Ref. 10). This submittal discusses the use of channel boxes for the WNP-2 Cycle 6 reload and states that the highest exposed reinserted channel boxes at the end of Cycle 6 will have an exposure of 37,800 MWd/MTU. This projected channel box exposure is well below the exposure range of 50,000 to 60,000 MWd/MTU that Bulletin 90-02 considers to be the range for accelerated channel box bow. Thus, the channel boxes in WNP-2 Cycle 6 are equivalent to single bundle lifetime channel boxes. The licensee references a letter from ANF to NRC, dated April 9, 1990 (Ref. 28) to address the effect on MCPR of channel box bow for channel boxes used for a single bundle lifetime. In its letter, ANF states that a large margin to MCPR limits is available to BWRs using the XN-3 critical power correlation. This margin was determined by comparing the critical powers computed for both 8x8 and 9x9 ANF fuel for the XN-3 and the ANFB correlations. The ANFB is a new ANF critical power correlation which was recently reviewed and approved by the NRC and is based on an extensive data base including full bundle data. The comparisons indicate that the XN-3 correlation is conservative and that the amount of conservatism in delta-CPR is at least twice that required to accommodate any bounding delta-CPR penalty as discussed in NRC Bulletin 90-02. The licensee proposes that no penalty need be assessed on MCPR limits for Cycle 6. Based on its review, the staff agrees with the licensee's assessment that no penalty needs to be assessed on Cycle 6 MCPR operating limits. The staff notes that the licensee intends to use channel boxes for bundle burnups in the 50 to 60 GWD/MTU range in future cycles and that the licensee will address this usage at a later date in accordance with NRC Bulletin 90-02.

The addition of 8 ANF 8x8 fuel assemblies proposed in the April 13, 1990, submittal did not significantly alter the design of the core reload. Thus, it did not substantially change the action noticed, or affect the initial proposed no significant hazards determination published in the Federal Register on April 4, 1990.

2.0 EVALUATION

2.1 Reload Description

The WNP-2 Cycle 6 reload will incorporate 152 unirradiated fuel assemblies which will replace 144 initial core assemblies manufactured by GE and eight ANF 8x8 fuel assemblies loaded in Cycle 2. The unirradiated fuel consists of 144 ANF-4/ANF-5 fuel assemblies, four GE11 LFAs, and four ABB SVEA-96 LFAs. The remainder of the core consists of 132 ANF 8x8C assemblies loaded for Cycle 5, 152 ANF 8x8C assemblies loaded for Cycle 4, 148 ANF 8x8C assemblies loaded for Cycle 3, 120 ANF 8x8C assemblies loaded for Cycle 2, 56 GE P8x8R assemblies reinserted from Cycles 1 and 2, and four ANF 9x9 LFAs loaded for Cycle 5. The ANF-4 and ANF-5 fuel assemblies are identical in design characteristics. The GE and ABB LFAs are designed to be compatible with the ANF-4 assemblies. The Cycle 6 loading is a conventional scatter pattern with low reactivity fuel on the core periphery.

2.2 Fuel Design

The mechanical design of the ANF 8x8C reload fuel is described in References 11, 12, and 13. The design of the GE11 LFAs is described in Reference 6. The design of the ABB SVEA-96 LFAs is described in Reference 7. The remaining fuel types to be returned to the Cycle 6 core were approved for operation in previous cycles. The 144 ANF 8x8C reload fuel assemblies manufactured for loading in Cycle 6 are essentially identical to the ANF 8x8C reload assemblies originally fabricated for reload in Cycle 2 in all major physical characteristics except for the uranium-235 enrichment. The Cycle 2 ANF fuel is described in Reference 14. Based on the considerations discussed above, we conclude that the mechanical designs of the ANF reload fuel, the GE11 LFAs, and the ABB SVEA-96 LFAs are acceptable because approved methodologies were used and acceptable results were obtained for various fuel design parameters.

2.3 Thermal-Hydraulic Design

The ANF thermal-hydraulic methodology and criteria used for the Cycle 6 design and analysis is the same as that used for previous WNP-2 reloads. The previous reload reviews concluded that hydraulic compatibility between ANF and GE fuel is satisfactory and the calculation of core bypass flow and the safety limit minimum critical power ratio (SLMCPR) is acceptable. The methodology for Cycle 6 is based on ANF's revised critical power methodology (Ref. 15) which has been approved by the NRC. The ANF XN-3 correlation used in the critical power ratio (CPR) analysis has been approved for application to both the ANF 8x8C and GE 8x8R fuel types (Ref. 16). Therefore, the SLMCPR of 1.06 for the reload fuel in Cycle 6 is acceptable. GE and ABB have analyzed their LFAs and confirm that the SLMCPR of 1.06 is conservative. Thus, we conclude that the SLMCPR of 1.06 is acceptable for all of the fuel types in Cycle 6.

2.4 Thermal-Hydraulic Stability

The NRC issued Amendment 71 to Facility Operating License No. NPF-21. This amendment defines regions on the power-flow map where operations are precluded and regions where operation is allowed with surveillance provided by the ANF stability monitoring system ANNA™. Core hydrodynamic stability analyses for Cycle 6 support the stability regions specified in Amendment 71. The results of the Cycle 6 analyses support operation below a line defined by the following power/flow points: 42% power/23.8% flow, 47% power/27.6% flow, and 65% power/45% flow. The stability line defined by these power/flow points bounds the stability regions specified in the Technical Specifications. Therefore, we conclude that the stability analysis for Cycle 6 is acceptable.

2.5 Nuclear Design

The nuclear design for WNP-2 Cycle 6 has been performed by ANF with the approved methodology described in Reference 17. The results of these analyses are given in the reload report (Ref. 4). The results are within the range of those usually encountered for BWR reloads. In particular, the shutdown margin is 1.2 percent delta-K with the value of R = 0.21 percent delta-K (R is the decrease in the shutdown margin at the exposure of minimum shutdown margin), thus fully meeting the required 0.38 or 0.28 percent delta-K shutdown margin. The Standby Liquid Control System (SLCS) also meets shutdown requirements with a shutdown margin of 3.59 percent delta-K. Because these and other WNP-2 Cycle 6 nuclear design parameters have been obtained with previously approved methods and fall within expected ranges, the nuclear design is acceptable.

2.6 Transient Analyses

Core wide transients were analyzed with the COTRANSA code (Ref. 18) which includes a one-dimensional neutron kinetics model for evaluation of the axial power shape response during pressurization transients (generator load rejection and feedwater controller failure). The referenced report has been reviewed by the staff and the methods for calculating the system transient response were found to be acceptable.

Calculation of the change in critical power ratio (CPR) during the core wide transient event involves the use of COTRANSA results which serve as input to a XCOBRA-T hot channel analysis model (Ref. 19) used to calculate delta CPR values. The XCOBRA-T code has been reviewed by the staff and found to be acceptable. The licensee evaluated several categories of potential core wide transients for Cycle 6 and provided specific results for the three limiting transients: load rejection without bypass (LRNB), feedwater controller failure (FWCF), and loss of feedwater heating (LOFH). For operation at rated power in the range of end-of-cycle (EOC)-2000 MWd/MTU to EOC, the LRNB is the limiting transient. The calculated delta-CPR, assuming WNP-2 measured scram speed, was 0.25 for ANF fuel and 0.24 for GE fuel resulting in MCPR Limits of 1.31 and 1.30 for ANF and GE fuel, respectively. The ANF 8x8 fuel results bound the GE 8x8 results in all cases; hence, only ANF 8x8 results will be reported in the following discussions of the transient analysis results.

If the recirculation pump trip (RPT) should become inoperable, the limiting transient between EOC-2000 MWd/MTU and EOC is still the LRNB. Assuming normal scram speeds, at EOC exposures, the MCPR operating limit is 1.36 for ANF 8x8 fuel with an inoperable RPT. For TS scram times, the MCPR operating limits are 1.36 and 1.40 for RPT operable and inoperable, respectively. These values are incorporated in the proposed Cycle 6 TS MCPR limits.

The control rod withdrawal error (RWE) was determined to be the most limiting event from beginning-of-cycle (BOC) to EOC-2000 MWd/MTU. The delta CPR for the RWE is 0.18 for a Rod Block Monitor (RBM) setting of 106%. This result for ANF 8x8 reload fuel bounds the result for GE reload fuel which is located in non-limiting core locations for Cycle 6. Thus, the TS MCPR operating limit is 1.24 from BOC to EOC-2000 MWd/MTU.

The most limiting event for reactor vessel overpressurization is the main steamline isolation valve (MISV) closure event without direct scram on valve position. The maximum value of the sensed pressure in the steam dome was 1291 psig which corresponds to a maximum vessel pressure of 1317 psig in the vessel lower plenum. These values are less than the TS limit of 1325 psig as measured by the steam dome pressure indicator and the 1375 psig ASME vessel pressure limit (110% of vessel design limit). Therefore, the results for the limiting overpressurization event are acceptable because vessel pressures remain below 110% of the vessel design limit and because conservative assumptions have been made in the analysis (no direct scram and six safety-relief valves out-of-service).

The limiting plant system transients discussed above were all analyzed at a core flow of 106% of rated core flow. ANF has performed analyses which demonstrate that the ANF 8x8C fuel assembly can operate satisfactorily from a mechanical standpoint at this increased flow. GE has also performed analyses for the reactor internals and for the GE fuel assembly and obtained acceptable results for operation at this increased core flow.

Based on the above considerations and on the similarity between the two reload fuel types used in Cycle 6, the staff concludes that WNP-2 can operate safely with extended core flow up to 106% of rated core flow during Cycle 6.

The licensee reviewed the recirculation flow run-up analysis performed for Cycle 2 and concluded that the Cycle 2 analysis is applicable to Cycle 6 except for the six degree reduction in feedwater temperature at full power conditions. Thus, the reduced flow MCPR for Cycle 6 is changed on the conservative side from earlier cycles. The reduced flow MCPR operating limit is acceptable and will be incorporated into the TSs.

2.7 Accident Analyses

The control rod drop accident (CRDA) yields a value of 97 cal/gm for the maximum deposited fuel rod enthalpy. This is well below the NRC required limit of 280 cal/gm and, therefore, is acceptable. The loss of coolant accident analysis for WNP-2 was performed for a full core of ANF 8x8C fuel (Refs. 20 and 21) and remains applicable for the ANF 8x8C and GE 8x8R initial and reload fuel. These LOCA analyses have covered an acceptable range of conditions, have been performed with approved methodology, and the current TS MAPLHGR values for ANF fuel were found acceptable. Because ANF 8x8C fuel is hydraulically and neutronically compatible with the GE 8x8R fuel in Cycle 6, the existing GE LOCA analysis and MAPLHGR limits remain applicable to the GE fuel. MAPLHGR limits have also been determined for the GE11 LFAs and the SVEA-96 LFAs. MAPLHGR limits for the ANF 9x9 LFAs are discussed in Reference 22. These analyses are acceptable.

2.8 Single Loop Operation (SLO)

Single loop operation was approved in Amendment 62 for Cycle 4. The description and evaluation of SLO in Amendment 62 (Refs. 23, 24, and 25) are applicable for Cycle 6. For the GE reload fuel, a multiplier of 0.84 is applied to the GE MAPLHGR limits for conservative application to SLO. For the ANF reload fuel, the two-loop ANF 8x8 MAPLHGR limits can be used for SLO. Two-loop MAPLHGRs also apply to the GE11 LFAs, SVEA-96 LFAs, and the ANF LFAs. For SLO, the SLMCPR remains at 1.07 for Cycle 6 and the MCPR operating limit remains at 1.35.

2.9 Final Feedwater Temperature Reduction (FFWTR)

Final feedwater temperature reduction (FFWTR) with the reactor at EOC and at an all rods out (ARO) conditions was approved in Amendment 77 for Cycle 5. The description and evaluation of FFWTR in Amendment 77 (Ref. 26) are applicable for Cycle 6. The results for the LRNB and the FWCF events are conservatively bounded by adding a delta-CPR of 0.02 to the delta-CPR results for normal temperatures. A supplemental analysis demonstrates that a 0.03 delta-CPR addition is applicable to the SVEA-96 LFAs. This is acceptable and allows the WNP-2 reactor to extend the fuel cycle beyond its design limits with a power coastdown.

2.10 GE11 Lead Fuel Assemblies

The licensee plans to load four GE11 LFAs in core locations which have been analyzed to have sufficient margin so that the LFAs are not expected to be the limiting assemblies in the core. GE evaluated the GE11 LFAs, which are 9x9 fuel assemblies, using methods that have been approved by the staff. Both core wide and local events were evaluated. Results of the calculations are discussed in Reference 6 and demonstrate that the GE11 LFAs have greater margins to licensing and design limits than the ANF 8x8C reload fuel assemblies which is the dominant fuel type in Cycle 6. Based on our review, we conclude that the loading of the four GE11 LFAs in Cycle 6 is acceptable and that appropriate TS limits have been established for these LFAs.

2.11 ABB SVEA-96 Lead Fuel Assemblies

The licensee plans to load four SVEA-96 LFAs in core locations which have been analyzed to have sufficient margin so that the LFAs are not expected to be the limiting assemblies in the core. ABB evaluated the SVEA-96 LFAs, which are 10x10 fuel assemblies with a centrally located water cross, using methods that have been approved by the staff. Both core wide and local events were evaluated. Results of the calculations are discussed in Reference 7. MCPR and MAPLHGR operating limits have been developed for Cycle 6 for the SVEA-96 LFAs. Based on our review, we conclude that the loading of the four ABB SVEA-96 LFAs in Cycle 6 is acceptable and that appropriate TS limits for these LFAs have been established.

2.12 Technical Specifications

The TS changes associated with the WNP-2 Cycle 6 reload have been reviewed. We found the following specifications to be acceptable.

(1) Definition 1.9 - Critical Power Ratio

This definition is being changed to reflect the need to use different critical power correlations for different fuel types in the core. This change is acceptable.

(2) Basis 2.0 - Introduction

The basis is being changed to reflect the use of other than ANF lead fuel assemblies in WNP-2. This change is acceptable.

(3) Specification and Basis 3/4.2.1 - Average Planar Linear Heat Generation Rate

This specification and basis are being changed to incorporate MAPLHGR curves for the GE11 LFAs and the SVEA-96 LFAs. These changes are acceptable.

(4) Specification and Basis 3/4.2.3 - Minimum Critical Power Ratio

This specification and basis are being changed to incorporate MCPR values for the ANF 8x8C and the SVEA-96 LFAs that reflect the Cycle 6 transient analyses including the FFWTR option. The GE 8x8 reload fuel, the ANF LFAs, and the GE11 LFAs will be monitored to the ANF 8x8C MCPR limits. In addition, the title of Figure 3.2.3-1 (MCPR Operating Limit versus Total Core Flow) is changed. These changes are acceptable.

(5) Specification 3/4.2.4 - Linear Heat Generation Rate

This specification is changed to include the LHGR limits for the GE11 LFAs and SVEA-96 LFAs. This change is acceptable.

(6) Specification 5.3.1 - Fuel Assemblies

This specification is being revised to permit the use of LFAs. This change is acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change in the installation and use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that this 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 or environmental assessment need be prepared in connection with the issuance of this amendment.

4.0 CONTACT WITH STATE OFFICIAL

The Commission made a proposed determination that the amendment involves no significant hazards consideration (55 FR 12603, April 4, 1990) and consulted with the State of Washington. No public comments were received, and the State of Washington did not have any comment.

5.0 CONCLUSION

We have reviewed the reports submitted for Cycle 6 operation of WNP-2. 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. The Technical Specification changes submitted for this reload suitably reflect the necessary modifications for operation in this cycle. The Technical Specification changes proposed by the licensee are acceptable because they are consistent with the requirements of 10 CFR 50.36. In addition, we conclude that the licensee's response to NRC Bulletin 90-02 on the channel box bow issue is acceptable for Cycle 6 operation.

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, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

6.0 REFERENCES

1. Letter (G02-90-032) from G. C. Sorensen (WPPSS) to NRC, dated February 27, 1990.

2. Letter (G02-90-075) from G. C. Sorensen (WPPSS) to NRC, dated April 13, 1990.

3. WPPSS-EANF-126, "WNP-2 Cycle 6 Reload Summary Report," February 1990.

4. ANF-90-02, "WNP-2 Cycle 6 Reload Analysis," January 1990.

5. ANF-90-01, "WNP-2 Cycle 6 Plant Transient Analysis," January 1990.

6. "GE11 Lead Fuel Assembly Report for Washington Public Power Supply System Nuclear Project No. 2, Reload 5 Cycle 6," December 18, 1989.

7. ABB Atom Report UK90-126, "Supplemental Lead Fuel Assembly Licensing Report SVEA-96 LFAs to WNP-2 - Summary," January 1990.

8. Attachment to WNP-2 Cycle 6 Reload Summary Report (Reference 3), "Summary Justification for Technical Specification Changes."

REFERENCES CONTINUED

9. NRC Information Notice No. 89-69, "Loss of Thermal Margin Caused by Channel Box Bow," September 29, 1989.

10. NRC Bulletin No. 90-02, "Loss of Thermal Margin Caused by Channel Box Bow," March 20, 1990.

11. XN-NF-81-21(A), Revision 1, "Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel," Exxon Nuclear Company, January 1982.

12. XN-NF-81-21(A), Revision 1, Supplement 1, "Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel," Exxon Nuclear Company, March 1985.

13. XN-NF-85-67(A), Revision 1, "Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel," Exxon Nuclear Company, September 1986.

14. WPPS-EANF-101, "WNP-2 Cycle 2 Reload Summary Report," February 1986.

15. XN-NF-524(P)(A), Revision 1, "Exxon Nuclear Critical Power Methodology for BWRs," November 1983.

16. Letter from H. Bernard (NRC) to G. F. Owsley (ENC), "Acceptance for Referencing of Topical Report XN-NF-512, Revision 1," July 22, 1982.

17. XN-NF-80-19(A), Volume 1 and Volume 1 Supplements 1 and 2, "Exxon Nuclear Methodology for Boiling Water Reactors: Neutronic Methods for Design and Analysis," November 1981.

18. XN-NF-79-71(P), Revision 2 (as supplemented), "Exxon Nuclear Power Plant Transient Methodology," November 1981.

19. XN-NF-84-105(A), Volume 1 (as supplemented), "XCOBRA-T: A Computer Code for BWR Transient Thermal Hydraulic Core Analysis," February 1987.

20. XN-NF-85-138(P), "LOCA Break Spectrum Analysis for a BWR5," Exxon Nuclear Company, December 1985.

21. XN-NF-85-139, "WNP-2 LOCA-ECCS Analysis MAPLHGR Results," Exxon Nuclear Company, December 1985.

22. WPPSS-EANF-124, "WNP-2 Cycle 5 Reload Summary Report," February 1989.

23. ANF-87-119, "WNP-2 Single Loop Operation Analysis," September 1987.

24. ANF-87-118, "WNP-2 LOCA Analysis for Single Loop Operation," September 1987.

REFERENCES CONTINUED

25. WPPSS-EANF-115, "WNP-2 Single Loop Operation Summary Report," February 1988.

26. Letter from Robert B. Samworth (NRC) to G. C. Sorensen (WPPSS), dated March 1, 1990. (This letter transmitted Amendment 77 to WPPSS.)

27. WPPSS-EANF-126, Revision 1, "WNP-2 Cycle 6 Reload Summary Report," April 1990.

28. Letter (RAC:030:90) from R. A. Copeland (ANF) to Robert C. Jones (NRC), dated April 9, 1990.

Principal Contributor: D. B. Fieno, SRXB

Dated: May 31, 1990