

November 7, 1988

Docket No. 50-333

DISTRIBUTION

<u>Docket File</u>	NRC/PDR
Local PDR	PDI-1 Rdg
SVarga	BBoger
CVogan	DLaBarge
OGC	DHagan
EJordan	BGrimes
TBarnhart (4)	WJones
EButcher	GSchwenk
ACRS (10)	GPA/PA
ARM/LFMB	JJohnson, RI

Mr. John C. Brons  
Executive Vice President - Nuclear Generation  
Power Authority of the State of New York  
123 Main Street  
White Plains, New York 10601

Dear Mr. Brons:

The Commission has issued the enclosed Amendment No. 117 to Facility Operating License No. DPR-59 for the James A. FitzPatrick Nuclear Power Plant. The amendment consists of changes to the Technical Specifications (TS) in response to your application transmitted by letter dated July 29, 1988 (TAC 69013).

The amendment would revise the TS to support plant operation following refueling during the Reload 8/Cycle 9 outage.

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular bi-weekly Federal Register notice.

Sincerely,

David E. LaBarge, Project Manager  
Project Directorate I-1  
Division of Reactor Projects, I/II

Enclosures:

1. Amendment No. 117 to DPR-59
2. Safety Evaluation

cc: w/enclosures  
See next page

PDI-1  
CVogan  
10/26/88

PDI-1 *DL*  
DLaBarge:vr  
10/26/88

*RUW*  
PDI-1  
RCapra  
10/07/88

*OGC/BJ wanted  
my being  
10/1/88*

*10/17*

*DFol  
11*

*CP*

Mr. John C. Brons  
Power Authority of the State of New York

James A. FitzPatrick Nuclear  
Power Plant

cc:

Mr. Gerald C. Goldstein  
Assistant General Counsel  
Power Authority of the State  
of New York  
10 Columbus Circle  
New York, New York 10019

Ms. Donna Ross  
New York State Energy Office  
2 Empire State Plaza  
16th Floor  
Albany, New York 12223

Resident Inspector's Office  
U. S. Nuclear Regulatory Commission  
Post Office Box 136  
Lycoming, New York 13093

Regional Administrator, Region I  
U.S. Nuclear Regulatory Commission  
475 Allendale Road  
King of Prussia, Pennsylvania 19406

Mr. Radford J. Converse  
Resident Manager  
James A. FitzPatrick Nuclear  
Power Plant  
Post Office Box 41  
Lycoming, New York 13093

Mr. A. Klausman  
Senior Vice President - Appraisal  
and Compliance Services  
Power Authority of the State  
of New York  
10 Columbus Circle  
New York, New York 10019

Mr. J. A. Gray, Jr.  
Director Nuclear Licensing - BWR  
Power Authority of the State  
of New York  
123 Main Street  
White Plains, New York 10601

Mr. George Wilverding, Manager  
Nuclear Safety Evaluation  
Power Authority of the State  
of New York  
123 Main Street  
White Plains, New York 10601

Mr. Robert P. Jones, Supervisor  
Town of Scriba  
R. D. #4  
Oswego, New York 13126

Mr. R. E. Beedle  
Vice President Nuclear Support  
Power Authority of the State  
of New York  
123 Main Street  
White Plains, New York 10601

Mr. J. P. Bayne, President  
Power Authority of the State  
of New York  
10 Columbus Circle  
New York, New York 10019

Mr. S. S. Zulla  
Vice President Nuclear Engineering  
Power Authority of the State  
of New York  
123 Main Street  
White Plains, New York 10601

Mr. Richard Patch  
Quality Assurance Superintendent  
James A. FitzPatrick Nuclear  
Power Plant  
Post Office Box 41  
Lycoming, New York 13093

Mr. R. Burns  
Vice President Nuclear Operations  
Power Authority of the State  
of New York  
123 Main Street  
White Plains, New York 10601

Charlie Donaldson, Esquire  
Assistant Attorney General  
New York Department of Law  
120 Broadway  
New York, New York 10271



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

POWER AUTHORITY OF THE STATE OF NEW YORK

DOCKET NO. 50-333

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 117  
License No. DPR-59

1. The Nuclear Regulatory Commission (the Commission) has found that:
  - A. The application for amendment by Power Authority of the State of New York (the licensee) dated July 29, 1988, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and 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 rules and 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;
  - 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.(?) of Facility Operating License No. DPR-59 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No.117, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION



Robert A. Capra, Director  
Project Directorate I-1  
Division of Reactor Projects, I/II

Attachment:  
Changes to the Technical  
Specifications

Date of Issuance: November 7, 1988

ATTACHMENT TO LICENSE AMENDMENT NO. 117

FACILITY OPERATING LICENSE NO. DPR-59

DOCKET NO. 50-333

Revise Appendix A as follows:

Remove Pages

vii  
7  
12  
13  
31  
43a  
47a  
47b  
123  
130  
135h  
135k  
135l  
245

Insert Pages

vii  
7  
12  
13  
31  
43a  
47a  
47b  
123  
130  
135h  
135k  
135l  
245

JAFNPP

LIST OF FIGURES

<u>Figure</u>	<u>Title</u>	<u>Page</u>
3.1-1	Manual Flow Control	47a
3.1-2	Operating Limit MCPR versus	47b
4.1-1	Graphic Aid in the Selection of an Adequate Interval Between Tests	48
4.2-1	Test Interval vs. Probability of System Unavailability	87
3.4-1	Sodium Pentaborate Solution of System Volume-Concentration Requirements	110
3.4-2	Saturation Temperature of Sodium Pentaborate Solution	111
3.5-1	Thermal Power and Core Flow Limits of Specifications 3.5.J.1 and 3.5.J.2.	134
3.5-6	(Deleted)	135d
3.5-7	(Deleted)	135e
3.5-8	(Deleted)	135f
3.5-9	MAPLHGR Versus Planar Average Exposure Reload 4, P8DRB284L	135g
3.5-10	(Deleted)	135h
3.5-11	MAPLHGR Versus Planar Average Exposure Reload 6, BP8DRB299	135i
3.5-13	MAPLHGR Versus Planar Average Exposure Reload 8, BD336A	135k
3.5-14	MAPLHGR Versus Planar Average Exposure Reload 8, BD339A	135l
3.6-1	Reactor Vessel Pressure Temperature Limits	163
4.6-1	Chloride Stress Corrosion Test Results at 500°F	164
6.1-1	Management Organization Chart	259
6.2-1	Plant Staff Organization	260

Amendment No. 14, 22, 43, 64, 72, 74, 88, 98, 113 117

**1.1 FUEL CLADDING INTEGRITY**

**Applicability:**

The Safety Limits established to preserve the fuel cladding integrity apply to those variables which monitor the fuel thermal behavior.

**Objective:**

The objective of the Safety Limits is to establish limits below which the integrity of the fuel cladding is preserved.

**Specifications:**

**A. Reactor Pressure > 785 psig and Core Flow > 10% of Rated**

The existence of a minimum critical power ratio (MCPR) less than 1.04 shall constitute violation of the fuel cladding integrity safety limit, hereafter called the Safety Limit. An MCPR safety limit of 1.05 shall apply during single-loop operation.

**2.1 FUEL CLADDING INTEGRITY**

**Applicability:**

The Limiting Safety System Settings apply to trip settings of the instruments and devices which are provided to prevent the fuel cladding integrity Safety Limits from being exceeded.

**Objective:**

The objective of the Limiting Safety System Settings is to define the level of the process variables at which automatic protective action is initiated to prevent the fuel cladding integrity Safety Limits from being exceeded.

**Specifications:**

**A. Trip Settings**

The limiting safety system trip settings shall be as specified below:

**1. Neutron Flux Trip Settings**

- a. IRM - The IRM flux scram setting shall be set at  $\leq 120/125$  of full scale.

## 1.1 BASES

1.1 FUEL CLADDING INTEGRITY

The fuel cladding integrity limit is set such that no calculated fuel damage would occur as a result of an abnormal operational transient. Because fuel damage is not directly observable, a step-back approach is used to establish a Safety Limit such that the minimum critical power ratio (MCPR) is no less than 1.04. MCPR  $>1.04$  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 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 protection system safety 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 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.

A. Reactor Pressure  $>785$  psig and Core Flow  $> 10\%$  of Rated

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). It is assumed that the plant operation is controlled to the nominal protective setpoints via the instrumented variable, i.e., the operating domain. The current load line limit analysis contains the current operating domain map. The Safety Limit (MCPR of 1.04) has sufficient conservatism to assure that in the event of an abnormal operational transient initiated from the MCPR operating conditions in specification 3.1.B, more than 99.9% of the fuel rods in the core are expected to avoid boiling transition. The MCPR fuel cladding safety limit is increased by 0.01 for single-loop operation as discussed in Reference 2. The margin between MCPR of 1.0 (onset of transition boiling) and the Safety Limit is derived from a detailed statistical analysis considering all of the uncertainties in monitoring the core operating state including the uncertainty in the boiling transition correlation as described in Reference 1. The uncertainties employed in deriving the Safety Limit are



## 1.1 (cont'd)

provided at the beginning of each fuel cycle. Because the boiling transition correlation is based on a large quantity of full scale data there is a very high confidence that operation of fuel assembly at the Safety Limit would not produce boiling transition. Thus, although it is not required to establish the safety limit, additional margin exists between the Safety Limit and the actual occurrence of loss of cladding integrity.

However, if boiling transition were to occur, clad perforation would not be expected. Cladding temperatures would increase to approximately 1100°F which is below the perforation temperature of the cladding material. This has been verified by tests in the General Electric Test Reactor (GETR) where fuel similar in design to FitzPatrick operated above the critical heat flux for a significant period of time (30 minutes) without clad perforation.

If reactor pressure should ever exceed 1400 psia during normal power operation (the limit of applicability of the boiling transition correlation) it would be assumed that the fuel cladding integrity Safety Limit has been violated.

In addition to the boiling transition limit (Safety Limit), operation is constrained to a maximum LHGR of 14.4 KW/ft for GE8X8EB fuel and 13.4 KW/ft for the remainder.

At 100% power, this limit is reached with maximum fraction of limiting power density (MFLPD) equal to 1.0. In the event of operation with MFLPD greater than the fraction of rated power (FRP), the APRM scram and rod block settings shall be adjusted as required in specifications 2.1.A.1.c and 2.1.A.1.d.

B. Core Thermal Power Limit (Reactor Pressure  $\leq$  785 psig)

At pressures below 785 psig, the core elevation pressure drop is greater than 4.56 psi for no boiling in the bypass region. At low powers and flows, this pressure drop is due to the elevation pressure of the bypass region of the core. Analysis shows that for bundle power in the range of 1-5 MWt, the channel flow will never go below  $28 \times 10^3$  lb/hr. This flow results from the pressure differential between the bypass region and the fuel channel. The pressure differential is primarily a result of changes in the elevation pressure drop due to the density difference between the boiling water in the fuel channel and the non-boiling water in the bypass region. Full scale ATLAS test data taken at pressures from 0 to 785 psig indicate that the fuel assembly critical power at  $28 \times 10^3$  lb/hr is approximately 3.35 MWt. With the design peaking factors, this corresponds to a core thermal power of more than 50%. Thus, a core thermal power limit of 25% for reactor pressures below 785 psig is conservative.

3.1 (CONTINUED)

M CPR Operating Limit for Incremental  
Cycle Core Average Exposure

At RBM Hi-trip level setting	BOC to EOC-2GWD/t	EOC-2GWD/t to EOC-1GWD/t	EOC-1GWD/t to EOC
S = .66W + 39%	1.25	1.27	1.30
S = .66W + 40%	1.25	1.27	1.30
S = .66W + 41%	1.25	1.27	1.30
S = .66W + 42%	1.28	1.28	1.30
S = .66W + 43%	1.33	1.33	1.33
S = .66W + 44%	1.33	1.33	1.33

2. If requirement 4.1.E.1 is not met (i.e.  $\tau_B < \tau_{AVE}$ ) then the Operating Limit M CPR values (as a function of  $\tau$ ) is as given in Figure 3.1-2.

Where  $\tau = (\tau_{AVE} - \tau_B) / (\tau_A - \tau_B)$

and  $\tau_{AVE}$  = the average scram time to notch position 38 as defined in specification 4.1.E.2,

$\tau_B$  = the adjusted analysis mean scram time as defined in specification 4.1.E.3,

$\tau_A$  = the scram time to notch position 38 as defined in specification 3.3.C.1

C. M CPR shall be determined daily during reactor power operation at  $\geq 25\%$  of rated thermal power and following any change in power level or distribution that would cause operation with a limiting control rod pattern as described in the bases for Specification 3.3.B.5.

D. When it is determined that a channel has failed in the unsafe condition, the other RPS channels that monitor the same variable shall be functionally tested immediately before the trip system containing the failure is tripped. The trip system containing the unsafe failure may be placed in the untripped condition during the period in which surveillance testing is being performed on the other RPS channels.

E. Verification of the limits set forth in specification 3.1.B shall be performed as follows:

1. The average scram time to notch position 38 shall be:

$$\tau_{AVE} \leq \tau_B$$

2. The average scram time to notch position 38 is determined as follows:

$$\tau_{AVE} = \frac{\sum_{i=1}^n N_i \tau_i}{\sum_{i=1}^n N_i}$$

where: n = number of surveillance tests performed to date in the cycle,  $N_i$  = number of active rods measured in

JAFNPP  
TABLE 3.1-1 (cont'd)

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENT

NOTES OF TABLE 3.1-1 (cont'd)

14. The APRM flow biased high neutron flux signal is fed through a time constant circuit of approximately 6 seconds. The APRM fixed high neutron flux signal does not incorporate the time constant, but responds directly to instantaneous neutron flux.
15. This Average Power Range Monitor scram function is fixed point and is increased when the reactor mode switch is place in the Run position.
- 16.\* During the proposed Hydrogen Addition Test, the normal background radiation level will increase by approximately a factor of 5 for peak hydrogen concentration. Therefore, prior to performance of the test, the Main Steam Line Radiation Monitor Trip Level Setpoint will be raised to  $\leq$  three times the increased radiation levels. The test will be conducted at power levels  $>80\%$  of normal rated power. During controlled power reduction, the setpoint will be readjusted prior to going below  $20\%$  rated power without the setpoint change, control rod withdrawal will be prohibited until the necessary trip setpoint adjustment is made.
17. This APRM Flow Referenced Scram setting is applicable to two loop operation. For one loop operation this setting becomes  $S \leq (0.66W + 54\% - 0.66\Delta W)(FRP/MFLPD)$  where  $\Delta W$  = Difference between two-loop and single-loop effective drive flow at the same core flow.

\* This specification is in effect only during Operating Cycle 7.

JAFNPP

Figure 3.1-1

$K_f$  FACTOR

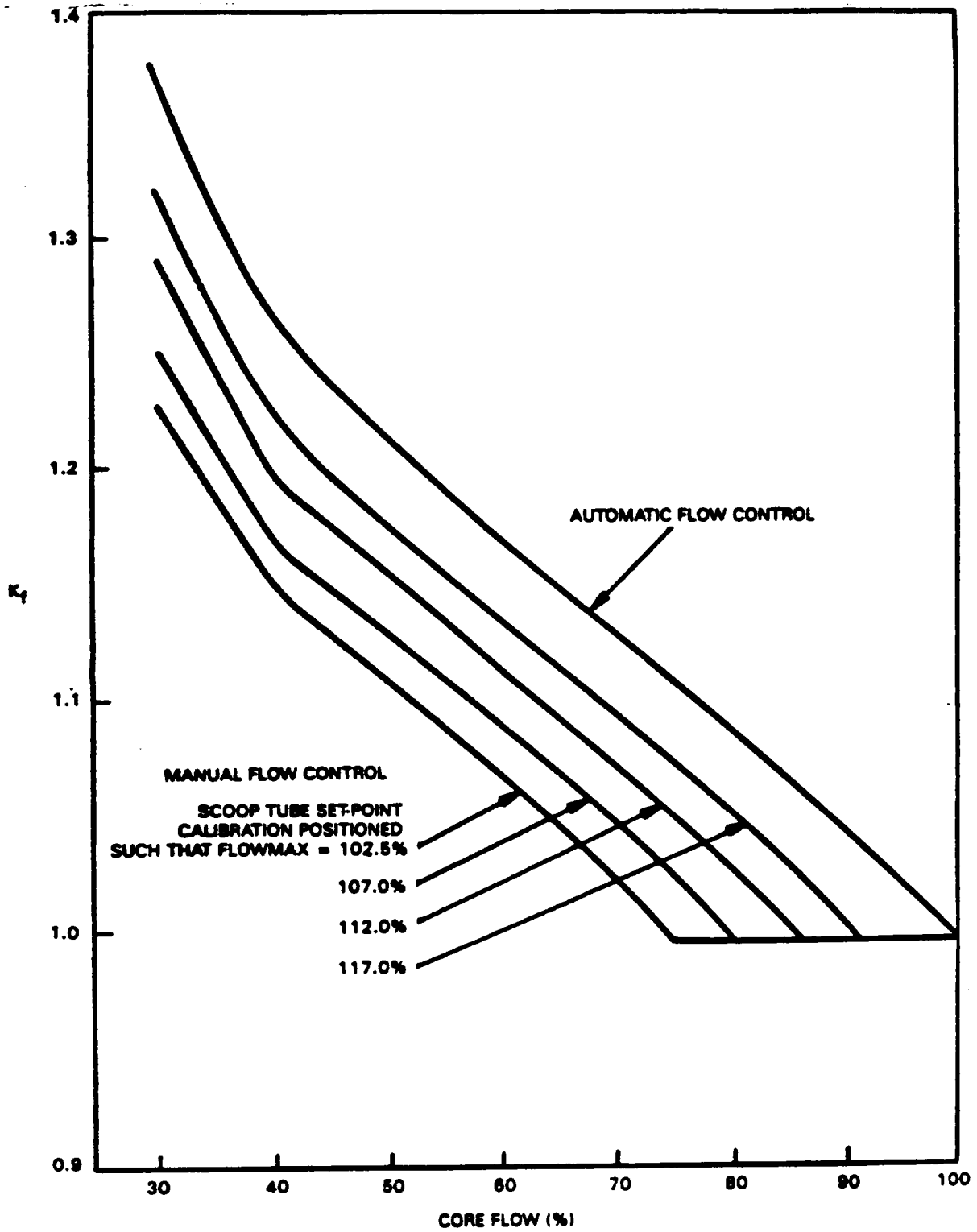
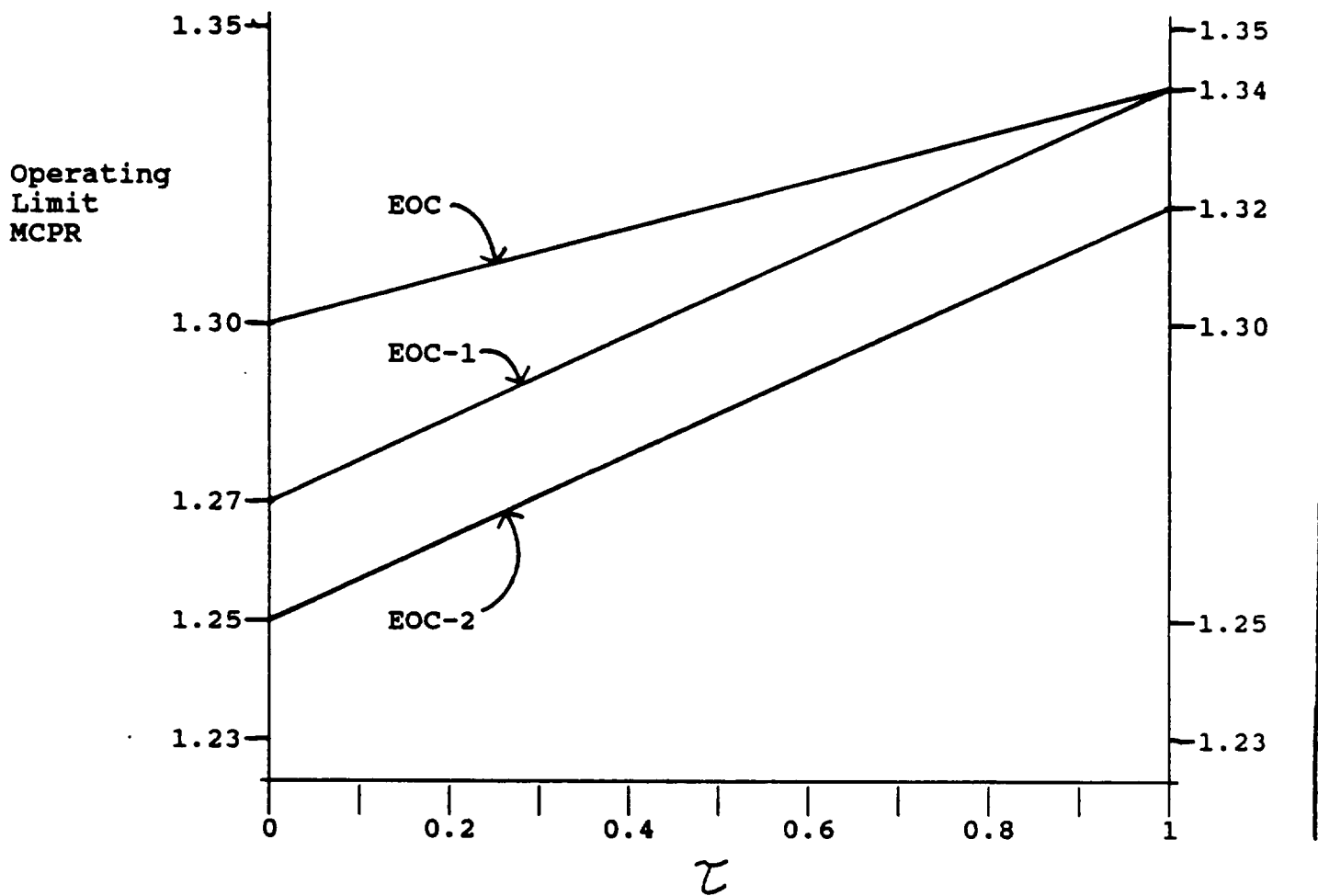


Figure 3.1-2

Operating Limit MCPR Versus  $\tau$   
(defined in Section 3.1.B.2)

FOR ALL FUEL TYPES



## 3.5 (cont'd)

condition, that pump shall be considered inoperable for purposes satisfying Specifications 3.5.A, 3.5.C, and 3.5.E.

H. Average Planar Linear Heat Generation Rate (APLHGR)

During power operation, the APLHGR for each type of fuel as a function of axial location and average planar exposure shall be within limits based on applicable APLHGR limit values which have been approved for the respective fuel and lattice types. When hand calculations are required, the APLHGR for each type of fuel as a function of average planar exposure shall not exceed the limiting value for the most limiting lattice (excluding natural uranium) shown in Figures 3.5-11 through 3.5-14 during two recirculation loop operation. During single loop operation, the APLHGR for each fuel type shall not exceed the above values multiplied by 0.84 (see Bases 3.5.K, Reference 1). If anytime during reactor power operation greater than 25% of rated power it is determined that the limiting value for APLHGR is being exceeded, action shall then be initiated within 15 minutes to restore operation to within the prescribed limits. If the APLHGR is not returned to within the prescribed limits within two (2) hours, an orderly reactor power reduction shall be commenced immediately. The reactor power shall be reduced to less than 25% of rated power within the next four hours, or until the APLHGR is returned to within the prescribed limits.

## 4.5 (cont'd)

2. Following any period where the LPCI subsystems or core spray subsystems have not been required to be operable, the discharge piping of the inoperable system shall be vented from the high point prior to the return of the system to service.
3. Whenever the HPCI, RCIC, or Core Spray System is lined up to take suction from the condensate storage tank, the discharge piping of the HPCI, RCIC, and Core Spray shall be vented from the high point of the system, and water flow observed on a monthly basis.
4. The level switches located on the Core Spray and RHR System discharge piping high points which monitor these lines to insure they are full shall be functionally tested each month.

H. Average Planar Linear Heat Generation Rate (APLHGR)

The APLHGR for each type of fuel as a function of average planar exposure shall be determined daily during reactor operation at  $\geq 25\%$  rated thermal power.

## 3.5 BASES (cont'd)

requirements for the emergency diesel generators.

G. Maintenance of Filled Discharge Pipe

If the discharge piping of the core spray, LPCI, RCIC, and HPCI are not filled, a water hammer can develop in this piping when the pump(s) are started. To minimize damage to the discharge piping and to ensure added margin in the operation of these systems, this technical specification requires the discharge lines to be filled whenever the system is required to be operable. If a discharge pipe is not filled, the pumps the supply that line must be assumed to be inoperable for technical specification purposes. However, if a water hammer were to occur, the system would still perform its design function.

H. Average Planar Linear Heat Generation Rate (APLHGR)

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 Appendix K.

The peak cladding temperature 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 only dependent secondarily on the rod to rod power distribution within an assembly. Since expected local variations in power distribution within a fuel assembly affect the calculated peak clad temperature by less than  $\pm 20^\circ\text{F}$  relative to the peak temperature for a typical fuel design, the limit on the average linear heat generation rate is sufficient to assure that calculated temperatures

are within the 10 CFR 50 Appendix K limit. The limiting values for APLHGR are given in Figures 3.5-11 through 3.5-14. Approved limiting values of APLHGR as a function of fuel type are given in NEDO-21662-2 (as amended) for Reload 6 fuel. Approved limiting values of APLHGR as a function of fuel and lattice types are given in NEDC-31317P (as amended) for Reload 7 and 8 fuel. These values are multiplied by 0.84 during Single Loop Operation. The derivation of this multiplier can be found in Bases 3.5.K, Reference 1.

I. Linear Heat Generation Rate (LHGR)

This specification assures that the linear heat generation rate in any rod is less than the design linear heat generation.

The LHGR shall be checked daily during reactor operation at 25% rated thermal power to determine if fuel burnup, or control rod movement, has caused changes in power distribution. For LHGR to be a limiting value below 25% rated thermal power, the ratio of local LHGR to average LHGR would have to be greater than 10 which is precluded by a considerable margin when employing any permissible control rod pattern.

JAFNPP

Figure 3.5-10

(This page is intentionally blank.)

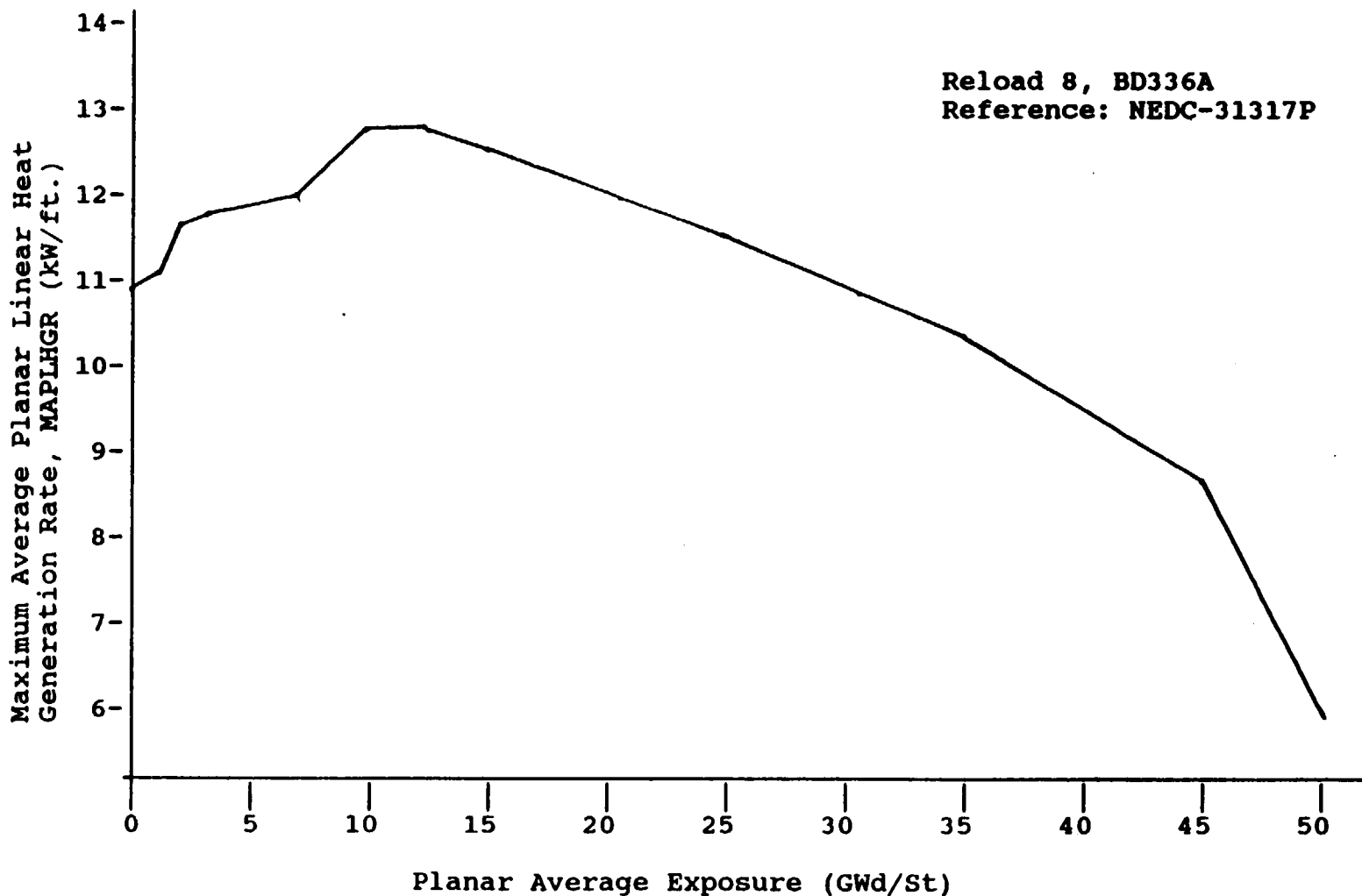
Amendment No. ~~64~~, ~~74~~, ~~98~~, 117

135h



Figure 3.5-13

Maximum Average Planar Linear Heat Generation Rate (MAPLHGR)  
Versus Average Planar Exposure

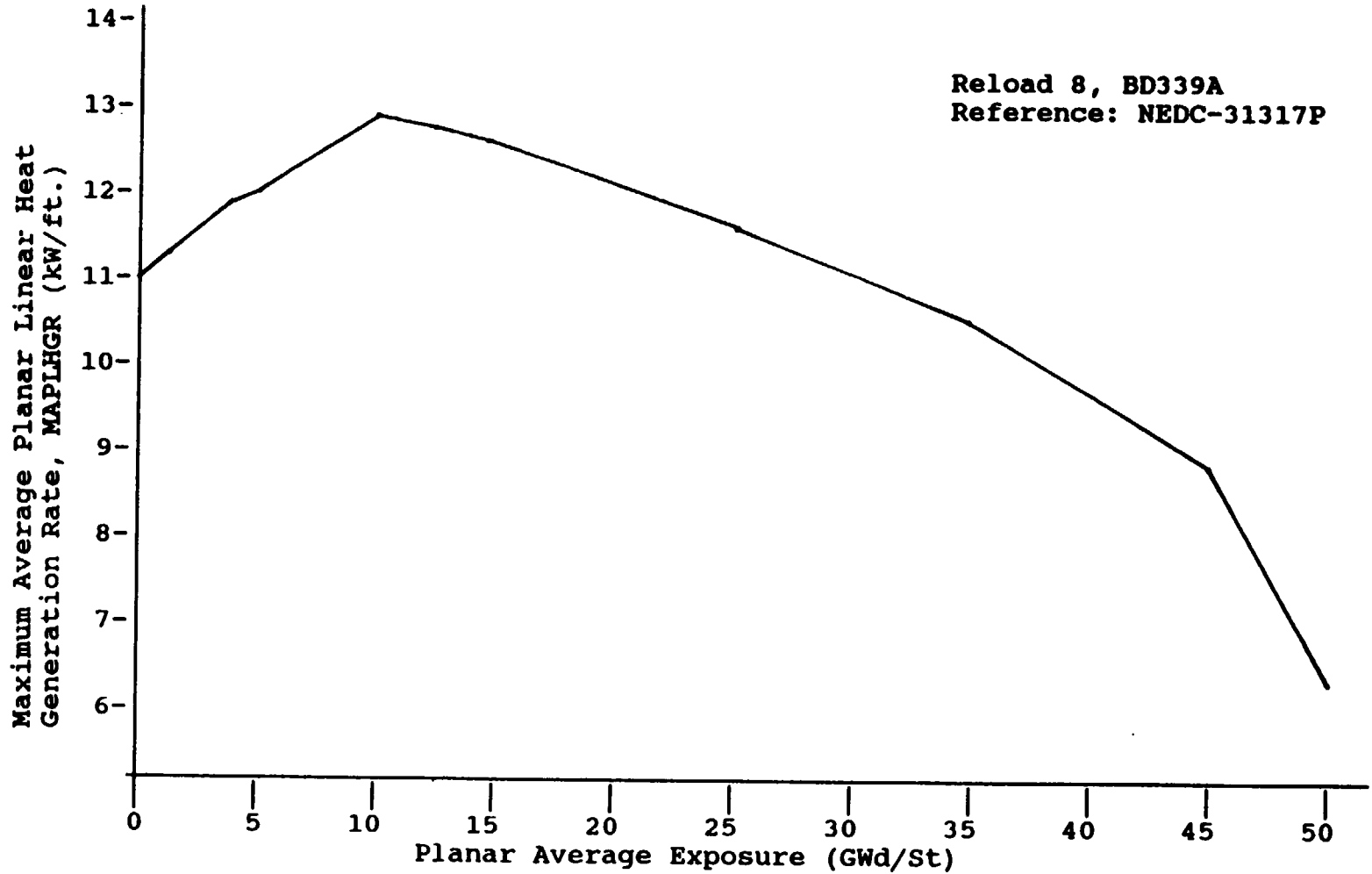


For single-loop operation, these MAPLHGR values are multiplied by 0.84.

This curve represents the limiting exposure dependent MAPLHGR values.

Figure 3.5-14

Maximum Average Planar Linear Heat Generation Rate (MAPLHGR)  
Versus Average Planar Exposure



For single-loop operation, these MAPLHGR values are multiplied by 0.84.

This curve represents the limiting exposure dependent MAPLHGR values.

Amendment No.117

**5.0 DESIGN FEATURES****5.1 SITE**

- A. The James A. FitzPatrick Nuclear Power Plant is located on the PASNY portion of the Nine Mile Point site, approximately 3,000 ft. east of the Nine Mile Point Nuclear Station, Unit 1. The NPP-JAF site is on Lake Ontario in Oswego County, New York, approximately 7 miles northeast of Oswego. The plant is located at coordinates north 4,819, 545.012 m, east 386, 968.945 m, on the Universal Transverse Mercator System.
- B. The nearest point on the property line from the reactor building and any points of potential gaseous effluents, with the exception of the lake shoreline, is located at the northeast corner of the property. This distance is approximately 3,200 ft. and is the radius of the exclusion areas as defined in 10 CFR 100.3.

**5.2 REACTOR**

- A. The reactor core consists of not more than 560 fuel assemblies. For the current cycle, three fuel types are present in the core: BP8X8R, GE8X8EB, and QUAD+. The GE fuel types are described in NEDO-24011. The BP8X8R fuel type has 62 fuel rods and 2 water rods and the GE8X8EB fuel type has 60 fuel rods and 4 water rods. The QUAD+ fuel type is described in WCAP-11159 and has 64 fuel rods.

- B. The reactor core contains 137 cruciform-shaped control rods as described in Section 3.4 of the FSAR.

**5.3 REACTOR PRESSURE VESSEL**

The reactor pressure vessel is as described in Table 4.2-1 and 4.2-2 of the FSAR. The applicable design codes are described in Section 4.2 of the FSAR.

**5.4 CONTAINMENT**

- A. The principal design parameters and characteristics for the primary containment are given in Table 5.2-1 of the FSAR.
- B. The secondary containment is as described in Section 5.3 and the applicable codes are as described in Section 12.4 of the FSAR.
- C. Penetrations of the primary containment and piping passing through such penetrations are designed in accordance with standards set forth in Section 5.2 of the FSAR.

**5.5 FUEL STORAGE**

- A. The new fuel storage facility design criteria are to maintain a  $K_{eff}$  dry  $< 0.90$  and flooded  $< 0.95$ . Compliance shall be verified prior to introduction of any new fuel design to this facility.



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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 117 TO FACILITY OPERATING LICENSE NO. DPR-59

POWER AUTHORITY OF THE STATE OF NEW YORK

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

DOCKET NO. 50-333

1.0 INTRODUCTION

By letter dated July 29, 1988 (Ref. 1), the Power Authority of the State of New York submitted proposed changes to the Technical Specifications for the James A. FitzPatrick Nuclear Power Plant to permit reloading and operation for Cycle 9. In support of these changes, the submittal included a Safety Evaluation, as well as the General Electric (GE) Report, "Supplemental Reload Licensing Submittal for the James A. FitzPatrick Nuclear Power Plant Reload 8," and the GE Report, "Loss-of-Coolant Analysis for James A. FitzPatrick Nuclear Power Plant." The staff has reviewed this submittal and has prepared the following evaluation.

2.0 EVALUATION

2.1 Reload Description

For Cycle 9, 184 irradiated fuel assemblies will be removed from the reactor core and replaced by 184 General Electric GE8x8EB assemblies.

2.2 Fuel Mechanical Design

The fuel (GE8x8EB) to be inserted into the core for Cycle 9 is similar to that customarily used for BWR reloads and is described in Reference 2. The mechanical design methodology is described in Reference 3 and was used in this design for the GE8x8EB fuel. Reference 3 has been approved by the staff (Ref. 4). We conclude that the fuel mechanical design for the GE8x8EB fuel is acceptable.

2.3 Nuclear Design

The nuclear design and analysis of the Cycle 9 reload was performed with methods and techniques which are described in Reference 3 and which are used in all reload analyses performed by GE. The results of the FitzPatrick analyses are within the range of those reload cores previously reviewed by the staff and found to be acceptable. We therefore conclude that the nuclear design and analysis of the Cycle 9 reload is acceptable.

2.4 Thermal-Hydraulic Design

The methods and procedures employed in the thermal-hydraulic (T-H) design and analysis of the Cycle 9 core are described in Reference 3. The value of 1.04

for the Minimum Critical Power Ratio (MCPR) safety limit, approved in that reference for the GEXL plus correlation, is used for Cycle 9. The methods and procedures used to obtain the operating limit MCPR are those described in Reference 3 and are acceptable.

## 2.5 Loss-of-Coolant Accident Analyses

The LOCA analyses in the reload were performed using the SAFER/GESTR code package and the application methodology described in Reference 5. Since the licensee used approved methods, and the results meet the staff's acceptance criteria, we conclude that these analyses are acceptable.

## 2.6 MCPR and MAPLHGR Limits

A safety limit MCPR has been imposed to assure that 99.9 percent of the fuel rods in the core will not experience boiling transition during normal operation and anticipated operational transients. As stated previously, the safety limit of 1.04 was used for Cycle 9.

To assure that the fuel cladding integrity safety limit MCPR will not be violated during any anticipated transient, the most limiting events were reanalyzed for this reload (Ref. 1) to determine which events result in the largest reduction in CPR. The operating limit MCPR was then established by adding the largest reduction factor in the CPR to the safety limit MCPR. Since acceptable methods (Ref. 3) have been used, we find the MCPR Technical Specification changes to be acceptable.

The Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limits specified in the proposed Technical Specification changes are less than or equal to the bounding MAPLHGR used in the SAFER/GESTR-LOCA analysis (Ref. 3) and are, therefore, acceptable.

## 2.7 Technical Specification Changes

The following Technical Specification changes proposed by the licensee reflect the new fuel Cycle 9:

1. Revise the List of Figures.
2. Revise the MCPR Safety and Operating Limits.
3. Reword of Core Thermal Power Limit Bases.
4. Correct a spelling error.
5. Update or add applicable MCPR, APLHGR, and MAPLHGR figures.
6. Revise reactor core description to include the new GE8X8EB fuel.

7. Delete specifications associated with the discharged fuel and with the Cycle 8 specific analysis.

The proposed changes are acceptable since they are based upon approved analytical methods as discussed above.

## 2.8 Summary Evaluation

Based on the above evaluation, we conclude that James A. FitzPatrick Nuclear Power Plant may be loaded and operated for Cycle 9. This conclusion is based on the following:

1. The safety analyses have been performed by previously approved methods and procedures.
2. The Cycle 9 core meets all of the staff's acceptance criteria.

## 3.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change in the installation or use of a facility component 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 Sec 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 CONCLUSION

We have 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 or to the health and safety of the public.

## 5.0 REFERENCES

1. Letter, John C. Brons (New York Power Authority) to USNRC, "Proposed Change to the Technical Specifications Regarding Reload 8/Cycle 9," dated July 29, 1988.
2. Supplemental Reload Licensing Submittal for James E. FitzPatrick Nuclear Power Plant Reload 6, General Electric, 23A 4825, November 1986.
3. GESTAR-II- "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A-8, July 1986.

4. Approval letter, D. G. Eisenhut (NRC) to R. Gridley (GE) dated May 12, 1978.
5. NEDE-23785-1-PA, "The GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident," Volume I, II and III, General Electric Company, June 1984.

Dated: November 7, 1988

PRINCIPAL CONTRIBUTOR:

G. Schwenk