August 26, 1983

Docket No. 50-333

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Mr. J. P. Bayne
Executive Vice President,
Nuclear Generation
Power Authority of the State
of New York
123 Main Street
White Plains, New York 10601

Dear Mr. Bayne:

The Commission has issued the enclosed Amendment No. 74 to Facility Operating License No. DPR-59 for the James A. FitzPatrick Nuclear Power Plant. The amendment consists of changes to the Technical Specifications in response to your request dated May 25, 1983. The changes were necessitated as a result of new fuel loaded during the current refueling outage (Reload 5) to support power operation in Cycle 6.

The amendment revises the Minimum Critical Power Ratio Operating Limits to accommodate the new fuel; adds additional Rod Block Monitor Trip Level Settings to facilitate control rod withdrawals; and, deletes references to fuel types and the supporting analyses for fuel removed from the core.

A copy of the Safety Evaluation is also enclosed.

Sincerely,

Original signed by/

Joseph D. Hegner, Project Manager
Operating Reactors Branch #2
Division of Licensing

Enclosures:
1. Amendment No. 74 to
   License No. DPR-59
2. Safety Evaluation

cc w/enclosures:
See next page

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Power Authority of the State of New York
James A. FitzPatrick Nuclear Power Plant

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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. 74
License No. DPR-59

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

   A. The application for amendment by the Power Authority of the State of New York (the licensee) dated May 25, 1983, 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(2) 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. 74, 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 its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

DBVassallo

Domenic B. Vassallo, Chief
Operating Reactors Branch #2
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 26, 1983

ATTACHMENT TO LICENSE AMENDMENT NO. 74

FACILITY OPERATING LICENSE NO. DPR-59

DOCKET NO. 50-333

Revise the Appendix "A" Technical Specifications as follows:

Remove	Replace
vii	vii
6	6
-	6a
9	9
10	10
13	13
31	31
31a	31a
43	43
47b	47b
47c	47c
47d	47d
73	73
123	123
124	124
130	130
135a	135a
135b	135b
135c	135c
135h	135h
245	245
246	246

LIST OF FIGURES

surveillance tests, checks, calibrations, and examinations shall be performed within the specified surveillance intervals. These intervals may be adjusted ± 25 percent. The interval as pertaining to instrument and electric surveillance shall never exceed one operating cycle. In cases where the elapsed interval has exceeded 100 percent of the specified interval, the next surveillance interval shall commence at the end of the original specified interval.

U. Thermal Parameters

1. Minimum critical power ratio (MCPR)-Ratio of that power in a fuel assembly which is calculated to cause some point in that fuel assembly to experience boiling transition to the actual assembly operating power as calculated by application of the GEXL correlation (Reference NEDE-10958).

2. Fraction of Limiting Power Density - The ratio of the linear heat generation rate (LHGR) existing at a given location to the design LHGR. The design LHGR is 13.4 KW/ft.

3. Maximum Fraction of Limiting Power Density- The Maximum Fraction of Limiting Power Density (MFLPD) is the highest value existing in the core of the Fraction of Limiting Power Density (FLPD).

4. Transition Boiling - Transition boiling means the boiling region between nucleate and film boiling. Transition boiling is the region in which both nucleate and film boiling occur intermittently with neither type being completely stable.

V. Electrically Disarmed Control Rod

To disarm a rod drive electrically, the four amphenol type plug connectors are removed from the drive insert and withdrawal solenoids rendering the rod incapable of withdrawal. This procedure is equivalent to valving out the drive and is preferred. Electrical disarming does not eliminate position indication.

W. High Pressure Water Fire Protection System

The High Pressure Water Fire Protection System consists of: a water source and pumps; and distribution system piping with associated post indicator valves (isolation valves). Such valves include the yard hydrant curb valves and the first valve ahead of the water flow alarm device on each sprinkler or water spray subsystem.

X. Staggered Test Basis

A Staggered Test Basis shall consist of:

a. A test schedule for a systems, subsystems, trains or other designated components obtained by dividing the specified test interval into n equal subintervals.

b. The testing of one system, subsystem, train or other designated component at the beginning of each subinterval.

Y. Rated Recirculation Flow

That drive flow which produces a core flow of 77.0 x 10^6 lb/hr.

Z. Top of Active Fuel

The Top of Active Fuel, corresponding to the top of the enriched fuel column of each fuel bundle, is located 352.5 inches above vessel zero, which is the lowest point in the inside bottom of the reactor vessel. (See General Electric drawing No. 919D690BD.)

AA. Rod Density

Rod density is the number of control rod notches inserted expressed as a fraction of the total number of control rod notches. All rods fully inserted is a condition representing 100 percent rod density.

1.1 (cont'd)

D. Reactor Water Level (Hot or Cold) Shutdown Conditions)

Whenever the reactor is in the shutdown condition with irradiated fuel in the reactor vessel, the water level shall not be less than that corresponding to 18 inches above the Top of Active Fuel when it is seated in the core.

2.1 (cont'd)

In the event of operation with a maximum fraction of limiting power density (MFLPD) greater than the fraction of rated power (FRP), the setting shall be modified as follows:

$$S \leq (0.66\ W + 54\%) \times \frac{FRP}{MFLPD}$$

Where:

FRP = fraction of rated thermal power (2436 MWt)

MFLPD = maximum fraction of limiting power density where the limiting power density is 13.4 KW/ft.

The ratio of FRP to MFLPD shall be set equal to 1.0 unless the actual operating value is less than the design value of 1.0, in which case the actual operating value will be used.

(2) Fixed High Neutron Flux Scram Trip Setting

When the Mode Switch is in the RUN position, the APRM fixed high flux scram trip setting shall be:

$$S \leq 120\%\ \text{Power}$$

1.1 (cont'd)

2.1 (cont'd)

A.1.d. <u>APRM Rod Block Trip Setting</u>

The APRM Rod block trip setting shall be:

$$S \leq 0.66 W + 42\%$$

where:

S = Rod block setting in percent of thermal power (2436 MWt)

W = Loop recirculation flow rate in percent of rated

In the event of operation with a maximum fraction limiting power density (MFLPD) greater than the fraction of rated power (FRP), the setting shall be modified as follows:

$$S \leq (0.66 W + 42\%) \left[\frac{FRP}{MFLPD} \right]$$

where:

FRP = fraction of rated thermal power (2436 MWt)

MFLPD = maximum fraction of limiting power density where the limiting power density is 13.4 KW/ft.

The ratio of FRP to MFLPD shall be set equal to 1.0 unless the actual operating value is less than the design value of 1.0, in which case the actual operating value will be used.

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.

Howerver, 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 13.4 kw/ft.

At 100% power, this limit is reached with a maximum fraction of limiting power density (MFLPD) equal to 1.0. In the event of operation with a 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 < 785 psig)

At pressures below 785 psig the core elevation pressure drop (0 power, 0 flow) is greater than 4.56 psi. At low powers and flows this pressure differential is maintained in the bypass region of the core. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low powers and flows will always be greater than 4.56 psi. Analyses show that with a flow of 28 x 10^3 lbs/hr bundle flow, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.56 psi driving head will be greater than 28 x 10^3 lbs/hr. Full scall ATLAS test data taken at pressures from 0 psig to 785 psig indicate that the fuel assembly critical power at this flow 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)

MCPR 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%	1.21	1.25	1.29
S = .66W + 40%	1.22	1.25	1.29
S = .66W + 41%	1.24	1.25	1.29
S = .66W + 42%	1.25	1.25	1.29
S = .66W + 43%	1.27	1.27	1.29
S = .66W + 44%	1.33	1.33	1.33

C. MCPR 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 posit n 38 shall be: $\tau_{AVE} \leq \tau_B$

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

$$\tau_{AVE} = \sum_{i=1}^{n} Ni\,\tau i \Bigg/ \sum_{i=1}^{n} Ni$$

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

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

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

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

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

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

*Note: Should the operating limit MCPR obtained from this figure be less than the operating limit MCPR found in Specification 3.1.B.1 for the applicable RBM trip level setting then specification 3.1.B.1 shall apply.

If anytime during reactor operation greater than 25% of rated power it is determined that the limiting value for MCPR is being exceeded, action shall then be initiated within fifteen (15) minutes to restore operation to within the prescribed limits. If the MCPR 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 MCPR is returned to within the proscribed limits. For core flows other than rated, the MCPR operating limit shall be multiplied by the appropriate k_f is as shown in figure 3.1.1.

the ith surveillance, and τ_i = average scram time to notch position 38 of all rods measured in the ith surveillance test.

3. The adjusted analysis mean scram time is calculated as follows:

$$\tau_B(\text{sec}) = \mu + 1.65 \quad \sigma \sqrt{\frac{N_1}{\sum_{i=1}^{n} N_i}}$$

where μ = mean of the distribution for the average scram insertion time to notch position 38 = 0.723 sec.

σ = standard deviation of the distribution for average scram insertion time to notch position 38=0.054 sec.

N_i = the total number of active rods measured in specification 4.3.C.1

The number of rods to be scram tested and the test intervals are given in specification 4.3.C.

TABLE 3.1-1 (cont'd)

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENT

Notes of Table 3.1-1 (cont'd)

C. High Flux IRM

D. Scram Discharge Volume High Level when any control rod in a control cell containing fuel is not fully inserted.

E. APRM 15% Power Trip

7. Not required to be operable when primary containment integrity is not required.

8. Not required to be operable when the reactor pressure vessel head is not bolted to the vessel.

9. The APRM downscale trip is automatically bypassed when the IRM Instrumentation is operable and not high.

10. An APRM will be considered operable if there are at least 2 LPRM inputs per level and at least 11 LPRM inputs of the normal complement.

11. See Section 2.1.A.1.

12. This equation will be used in the event of operation with a maximum fraction of limiting power density (MFLPD) greater than the fraction of rated power (FRP).

Where: FRP = Fraction of Rated Thermal Power (2436 MWt)

MFLPD = Maximum Fraction of Limiting Power Density where the limiting power density is 13.4 KW/ft.

The ratio of FRP to MFLPD shall be set equal to 1.0 unless the actual operating value is less than the design value of 1.0, in which case the actual operating value will be used.

W = Loop Recirculation Flow in percent of rated

S = Scram Setting in percent of initial

13. The Average Power Range Monitor scram function is varied as a function of recirculation flow (W). The trip setting of this function must be maintained in accordance with Specification 2.1.A.1.c.

Figure 3.1-2

Operating Limit MCPR
Versus τ (defined in Section 3.1.B.2)

FOR ALL FUEL TYPES

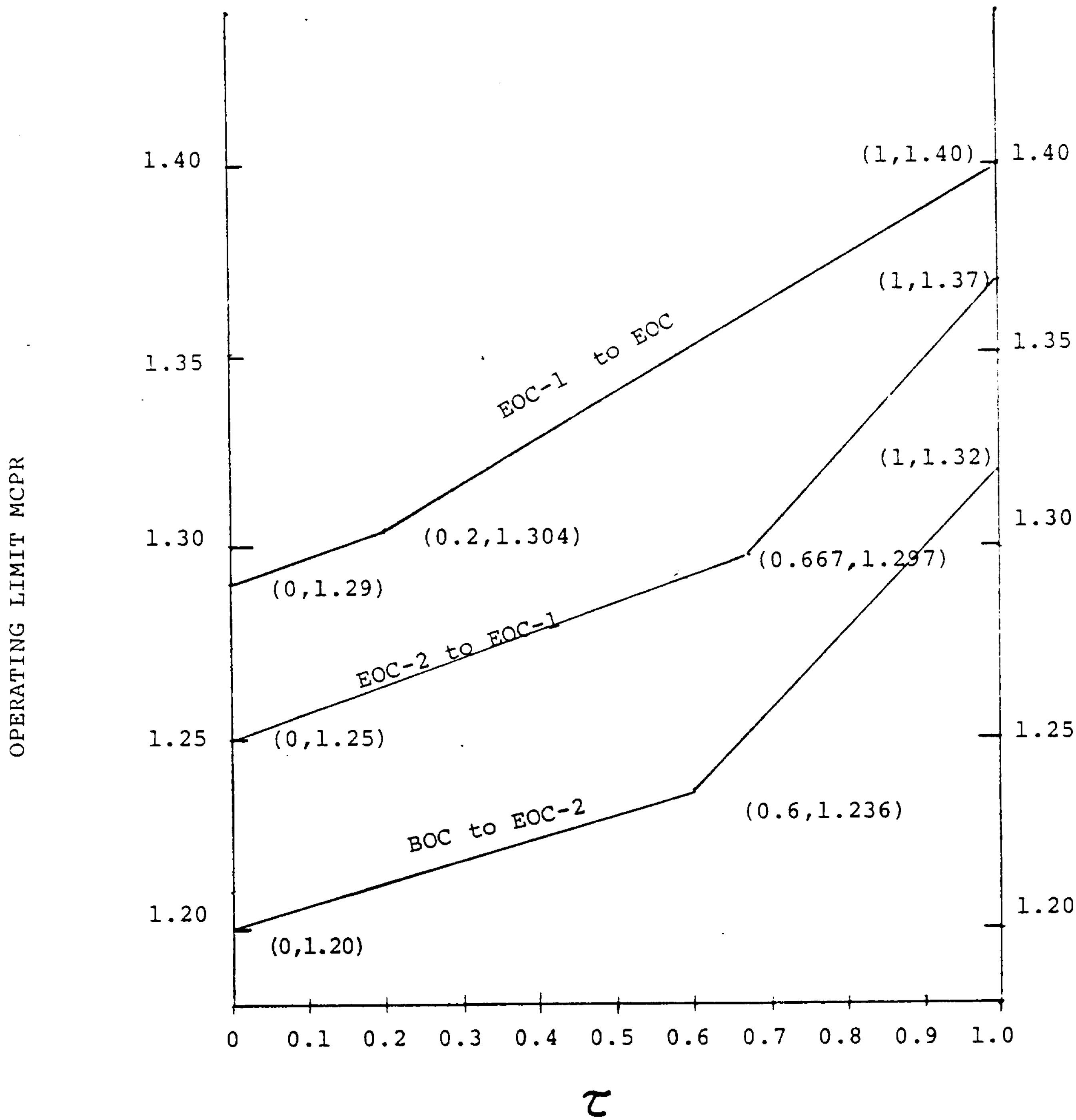

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TABLE 3.2-3 (Cont'd)

INSTRUMENTATION THAT INITIATES CONTROL ROD BLOCKS

NOTES FOR TABLE 3.2-3 (Cont'd)

The APRM and RBM rod blocks need not be operable in start-up mode. From and after the time it is found that the first column cannot be met for one of the two trip systems, this condition may exist for up to seven days provided that during that time the operable system is functionally tested immediately and daily thereafter; if this condition lasts longer than seven days, the system shall be tripped. From and after the time it is found that the first column cannot be met for both trip systems, the systems shall be tripped.

2. IRM downscale is bypassed when it is on its lowest range.

3. This function is bypassed when the count is ≥ 100 cps.

4. One of the four SRM inputs may be bypassed.

5. This SRM function is bypassed when the IRM range switches are on range 8 or above.

6. The trip is bypassed when the reactor power is ≤ 30%.

7. This function is bypassed when the Mode Switch is placed in Run.

8. S = Rod Block Monitor Setting in percent of initial.

   W = Recirculation flow in percent of rated

   K = Intercept values of 39%, 40%, 41%, 42%, 43% and 44% can be used with appropriate MCPR Limits from Section 3.1.B.

9. When the reactor is subcritical and the reactor water temperature is less than 212°F, the control rod block is required to be operable only if any control rod in a control cell containing fuel is not fully inserted.

10. When the control rod block function associated with scram discharge instrument volume high water level is not operable when required to be operable, the trip system shall be tripped.

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)

The APLHGR for each type of fuel as a function of average planar exposure shall not exceed the limiting value shown in Figures 3.5-6 through 3.5-10. 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 (cont'd)

I. Linear Heat Generation Rate (LHGR)

The linear heat generation rate (LHGR) of any rod in any fuel assembly at any axial location shall not exceed the maximum allowable LHGR of 13.4 KW/ft.

If anytime during reactor power operation greater than 25% of rated power it is determined that the limiting value for LHGR is being exceeded, action shall then be initiated within 15 minutes to restore operation to within the prescribed limits. If the LHGR 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 LHGR is returned to within the prescribed limits.

4.5 (cont'd)

I. Linear Heat Generation Rate (LHGR)

The LHGR shall be checked daily during reactor operation at ≥ 25% rated thermal power.

Amendment No. ~~45~~, ~~64~~, 74

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 that 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°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 value for APLHGR is shown in Figure 3.5-6 through 3.5-10.

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 $\geq$ 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.

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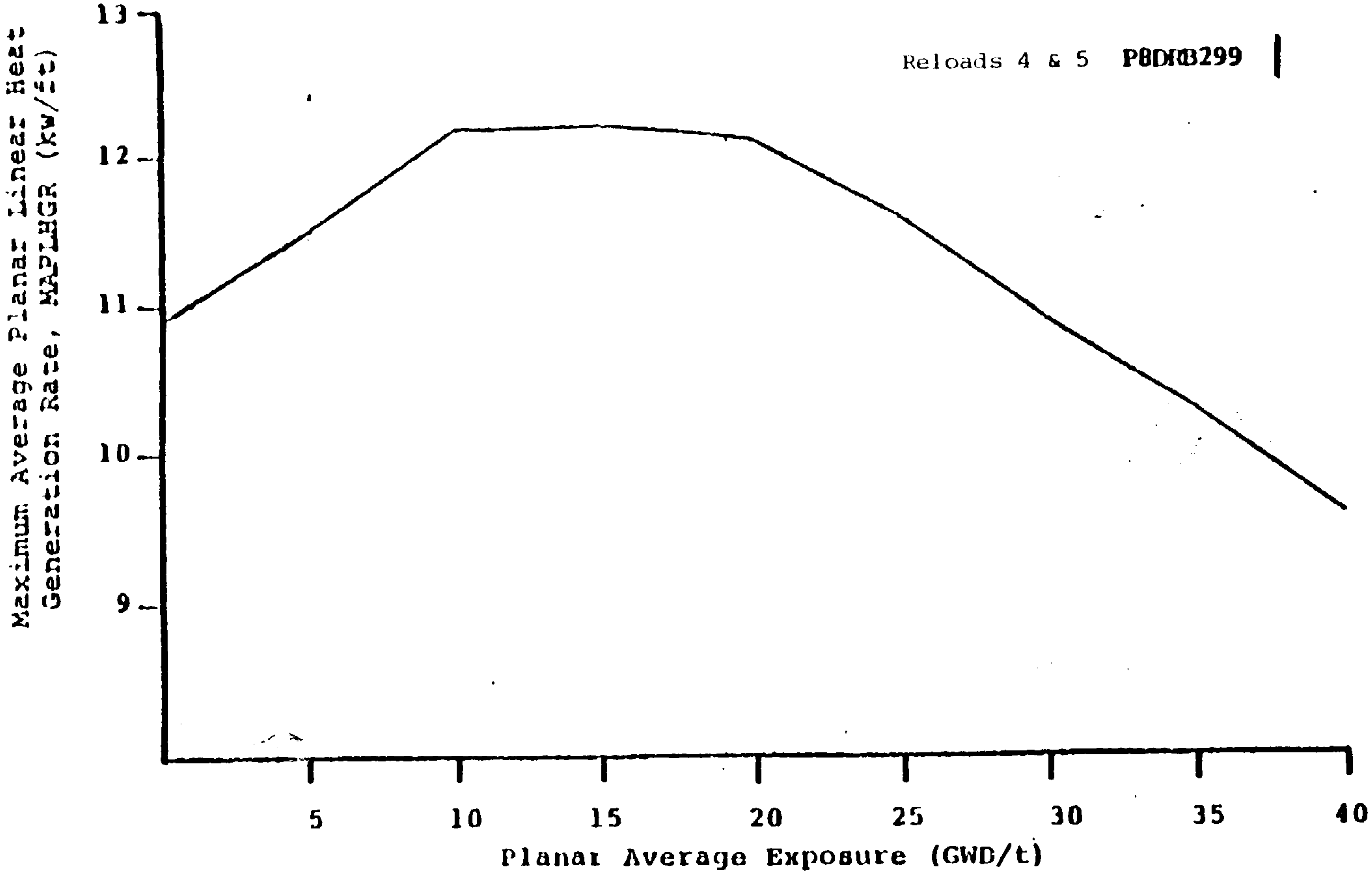
Figure 3.5-10



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

Reference: NEDO-21662-2
(As Ammended
August 1981)

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 NMP-JAF site is on Lake Ontario in Oswego Country, 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, two fuel types are present in the core: 8x8R and P8x8R. These fuel types are described in NEDO-24011. Both 8x8R and P8x8R fuel types have 62 fuel rods and 2 water 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.

5.5 (cont'd)

B. The spent fuel storage pool is designed to maintain K_{eff} less than 0.95 under all conditions as described in the Authority's application for spent fuel storage modification transmitted to the NRC July 26, 1978. In order to assure that the criterion is met, new fuel will be limited to an axial loading of 16.28 gm U-235/axial cm or equivalent. (For the present fuel design, described in NEDO-24011, this axial loading is equivalent to an average lattice enrichment of 3.3 w/o U-235.) The number of spent fuel assemblies stored in the spent fuel pool shall not exceed 2244.

5.6 Seismic Design

The reactor building and all engineered safeguards are designed on a basis of dynamic analysis using acceleration response spectrum curves which are normalized to a ground motion of 0.08 g for the Operating Basis Earthquake and 0.15 g for the Design Basis Earthquake.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 74 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

In its application for amendment dated May 25, 1983, the Power Authority of the State of New York (the licensee) proposed changes to the Technical Specifications for the James A. FitzPatrick Nuclear Power Plant (the facility) as a result of new fuel loaded during the current refueling outage (Reload 5) to support operation in Cycle 6. The reload involves removing depleted fuel assemblies in about one-third of the nuclear reactor core and replacing them with new fuel at the same type previously loaded in the core. The proposed changes pertain to revisions in the Minimum Critical Power Ratio (MCPR) Operating Limits to accommodate the new fuel; add additional Rod Block Monitor Trip Level Settings to facilitate control rod withdrawals; and, delete referenced to the fuel types and supporting analyses for the fuel removed from the core. In support of the of the reload application, the licensee has also enclosed the GE BWR supplemental licensing submittal for the facility in Reference 1. Our evaluation of the licensee's submittal and proposed Technical Specification change is provided below.

2.0 Evaluation

2.1 Fuel Mechanical Design

The Cycle 6 core consists of 12 GE 8x8R bundles and 548 GE P8x8R bundles. Two hundred of the GE P8x8R bundles contain fresh fuel. The P8x8R fuel is of the current GE standard design as described in Reference 3 which has been approved by the staff in Reference 5. Both 8x8R and P8x8R fuel types have 62 fuel rods and two water rods.

2.2 MAPLHGR Limit

The maximum average planar linear heat generation rate (MAPLHGR) for the fresh fuel labeled Type P8DRB299 is identical to the previously approved MAPLHGR limit for the existing Cycle 5 fuel of the same type. We thus find the MAPLHGR limit acceptable for Cycle 6.

2.3 Nuclear Design

The nuclear design analysis was performed with the methods and procedures described in Reference 3 which has been approved by the staff in Reference 5 for reload applications. The nuclear parameters for the reload core are within the range of those normally obtained and are acceptable.

2.4 Thermal-Hydraulic Design

The objective of the review is to confirm the thermal-hydraulic design of the core has been accomplished using acceptable methods, and provides an acceptable margin of safety from conditions which could lead to fuel damage during normal operation and anticipated operational transients, and is not susceptible to thermal-hydraulic instability.

The review includes the following areas: (1) safety limit minimum critical power ratio (MCPR), (2) operating limit MCPR, and (3) thermal-hydraulic stability.

The licensee has submitted the analysis report for Cycle 6 operation at rated core flow conditions (Ref. 2). Discussion of the review concerning the thermal-hydraulic design for Cycle 6 operation follows.

Safety Limit MCPR

A safety limit MCPR has been imposed to assure that 99.9 percent of the fuel rods in the core are not expected to experience boiling transition during normal and anticipated operational transients. As stated in Reference 3, the approved safety limit MCPR is 1.07. The safety limit MCPR of 1.07 is used for FitzPatrick Cycle 6 operation.

Operating Limit MCPR

The most limiting events have been analyzed by the licensee to determine which event could potentially induce the largest reduction in the initial critical power ratio (ΔCPR). The ΔCPR values given in Section 9 of Reference 2 are plant-specific values calculated by using approved methods including ODYN methods. The calculated ΔCPRs are adjusted to reflect either Option A or Option B ΔCPR by employing the conversion methods described in Reference 4, which was approved by the staff in Reference 6. The MCPR values are determined by adding the adjusted ΔCPRs to the safety limit MCPR. Section 11 of Reference 2 presents both the cycle MCPR values of the pressurization and non-pressurization transients. The maximum cycle MCPR values (Options A and B) in Section 11 are specified as the operating limit MCPRs and incorporated into the Technical Specifications. We found that the approved method was used to determine the operating limit MCPRs to avoid violation of the safety limit MCPR in the event of any anticipated transients. Therefore, we conclude that these limits are acceptable.

Thermal-Hydraulic Stability

The results of thermal-hydraulic analyses (Ref. 2) show that the maximum core stability decay ratio is 0.93, as compared to 0.87 for Cycle 5 core, which has been previously approved. Since the calculated maximum core stability decay ratio is less than some of the operating plants (for example, Peach Bottom Units 2 and 3 have decay ratio of 0.98) and since operation in the natural circulation mode is prohibited by Technical Specification 3.5.J, there will be additional margin to the stability limit. Therefore, we conclude that the thermal-hydraulic stability results are acceptable for Cycle 6 operation.

2.5 Transient and Accident Analyses

A cycle specific analysis of the rod drop accident was performed for Cycle 6 because the accident reactivity shape function was not bounded by the generic curve for both cold and hot standby conditions. In both cases the calculated peak fuel enthalpy was less than our acceptance criterion of 280 calories per gram. We conclude that the analysis of the rod drop accident is acceptable.

The analysis of the rod withdrawal error at power was extended to include larger values of the rod block monitor trip setting with accompanying larger changes in the critical power ratio. This permits greater freedom of rod motion at times when other events are limiting with respect to the operating value of MCPR. This is an acceptable procedure.

2.6 Technical Specifications

With the exceptions noted below, the proposed Technical Specification changes are related to the discharge of the last 8x8 fuel from the core or are typographical error corrections or clarifications. The only fuel remaining in the core is of the 8x8R type. These changes are acceptable.

The two additional equations for the Rod Block Monitor setpoints are added in Specification 3.1.B and in the Notes for Table 3.2-3. These are acceptable. Figure 3.1-2 has been altered to reflect the MCPR operating limit as a function of τ for the new cycle and MAPLHGR limits for the new fuel (Type P8DRB299) are added in Figure 3.5-10. These changes are acceptable.

3.0 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.

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4.0 References

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2. "Supplemental Reload Licensing Submittal for James A. FitzPatrick Nuclear Power Plant, Reload 5," GE Report Y 1003J01A56, March, 1983.

3. "General Electric Standard Application for Reactor Fuel," GE Report NEDE-24011-P-A-4, January, 1982.

4. "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," GE Report NEDE-24154-P, October, 1978.

5. D. Eisenhut (NRC) to R. Gridley (GE), May 12, 1978, Safety Evaluation for General Electric Standard Application for Reactor Fuel.

6. R. Tedesco (NRC) to G. Sherwood (GE), February 14, 1981, Acceptance for Referencing GE Report NEDE-24154-P, in reload submittals.