Attachment I

PROPOSED TECHNICAL SPECIFICATIONS

REGARDING RELOAD 5/CYCLE 6

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

POWER AUTHORITY OF THE STATE OF NEW YORK

Docket No. 50-333

LIST OF FIGURES

surveillance tests, checks, calibrations, and examinations shall be performed within the specified surveillance intervals. These intervals may be adjusted $\pm$ 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 77.0×10^6 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.)

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:

$S \leq 120\%$ Power

1.1 (cont'd)

2.1 (cont'd)

A.1.d. APRM Rod Block Trip Setting

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 fractio 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 position 38 shall be: $\tau_{AVE} \leq \tau_B$
2. The average scram time to notch position 38 is determined as follows:

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

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:

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

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_1 = 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

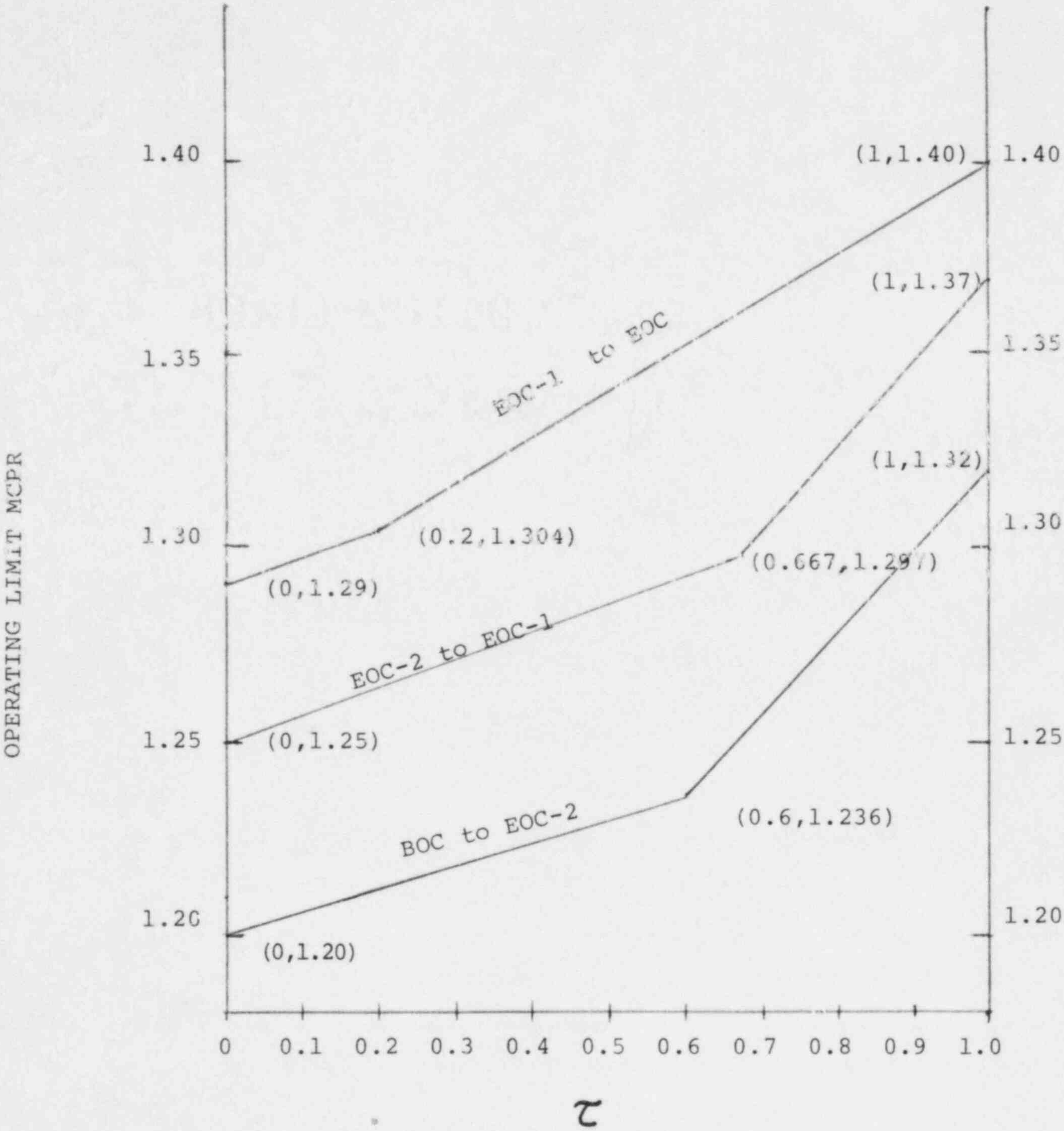
13. The Average Power Range Monitor scram function is varied as a function of recirculation flow (W). The trip setting of this function must be maintained in accordance with Specification 2.1.A.1.c.

Amendment No. ~~49~~, ~~62~~, ~~64~~, ~~67~~, ~~69~~, ~~72~~

Figure 3.1-2

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

FOR ALL FUEL TYPES



This page deleted

This page deleted

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 $\geq$ 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 $\leq$ 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 $\geq$ 25% rated thermal power.

Amendment No. ~~48~~, ~~64~~

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.

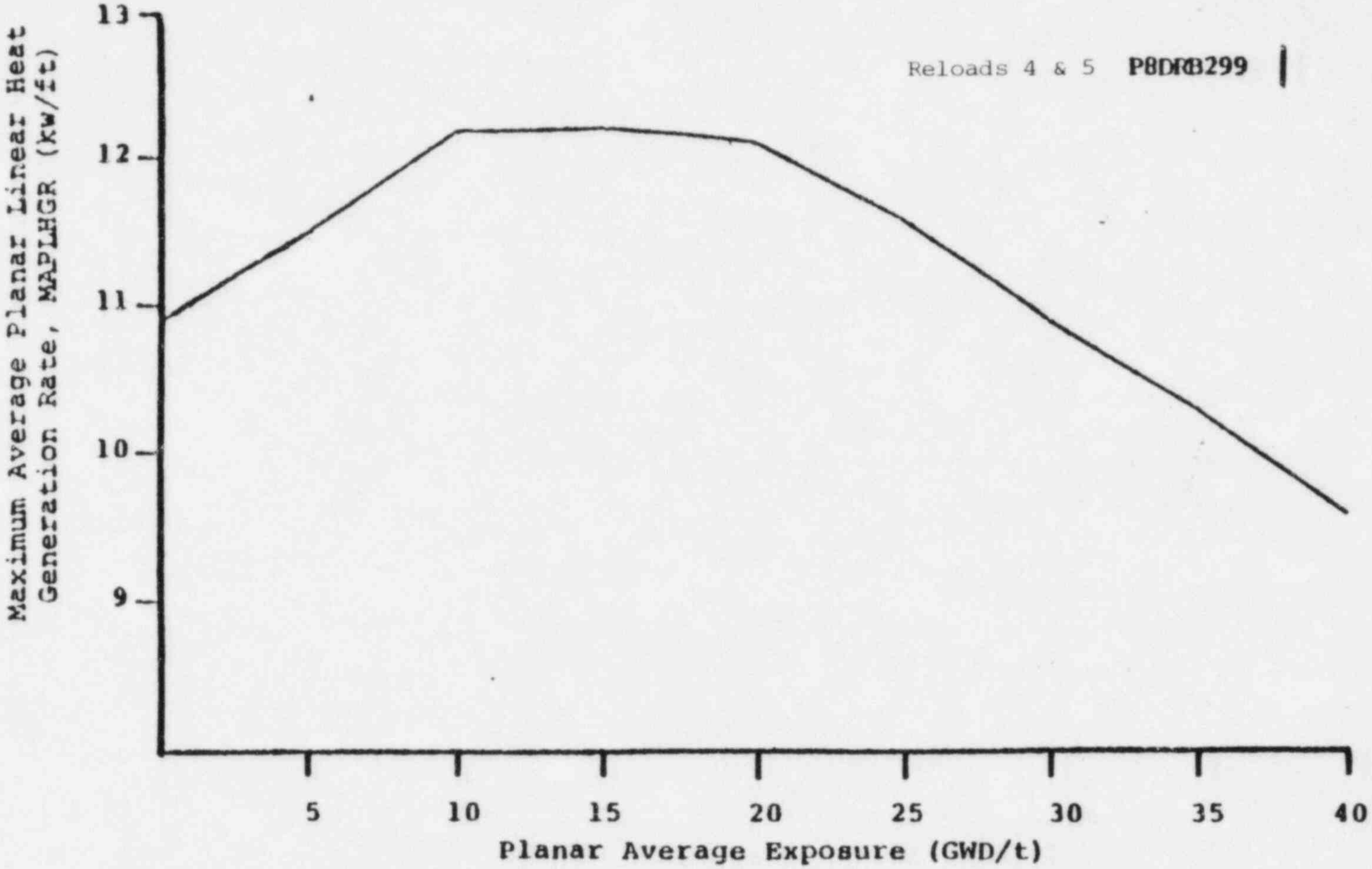
The LHGR shall be checked daily during reactor operation at ≥ 25% rated thermal power to determine if fuel burnup, or control rod movement, has caused changes in power distribution. For LHGR to be a limiting value below 25% rated thermal power, the ratio of local LHGR to average LHGR would have to be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

This page deleted

This page deleted

This page deleted

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.

Attachment II

PROPOSED TECHNICAL SPECIFICATIONS

REGARDING RELOAD 5/CYCLE 6

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

POWER AUTHORITY OF THE STATE OF NEW YORK

Docket No. 50-333

I. Description of the Changes

All changes described herein refer to Appendix A of the FitzPatrick Technical Specifications. Most of the changes involve deletions associated with discharged fuel types and new transient analyses for the forthcoming Reload 5/Cycle 6 core. This core will consist of 548 P8x8R bundles and 12 8x8R bundles. The core will contain no 8x8 bundles.

Specific changes are described below.

1. In the List of Figures on page vii, references to Figures 3.1-2a, 3.1-2b and 3.1-2c are deleted. These figures, specifying MCPR Operating Limit versus scram time ratio, are incorporated into a single new Figure 3.1-2 reflecting Cycle 6 transient analyses for all fuel types. References to Figures 3.5-3, 3.5-4 and 3.5-5, specifying thermal limits for discharged fuel types, are also deleted.

2. All references to specific fuel types are deleted on pages 6, 9, 10, 13, 43 and 124. These references occur in sections discussing thermal limits, which will be core-wide limits for Cycle 6.

3. On a new page 6a, a definition for Top of Active Fuel is added to the definitions section, since there are numerous references to this term in the Technical Specifications.

4. In Section 3.1.B on page 31, the Table of MCPR Operating Limits for Incremental Cycle Core Average Exposure is revised to reflect transient analyses for the Reload 5/ Cycle 6 core. Again, all references to specific fuel types are deleted, since the P8x8R limits are core-wide. Furthermore, MCPR limits at two additional Rod Block Monitor Trip Level Settings (S=.66W + 43% and S=.66W + 44%) are added to the table.

5. On pages 31a, 123 and 130, references to deleted figures are removed.

6. In section 1.1.D on page 9, the parenthetical reference to indicated water level is removed. Thus, water level remains defined in relation to the Top of Active Fuel to avoid confusion with other level indications in the reactor vessel.

7. In the first line of the notes on page 73, a typographical error, reading RRM, is corrected to read RBM, for Rod Block Monitor.

8. On pages 124 and 130, references to core power are prefaced with the words "rated thermal," to clarify the references.

9. In the Notes for Table 3.1-1 on page 43, a statement is added to note 6D, "Scram Discharge Volume High Level." The note should read, "Scram Discharge Volume High Level when any control rod in a control cell containing fuel is not fully inserted." The latter part of this statement was inadvertently omitted in the retyping and processing of page 43 for Amendment No. 69.

10. A new Figure 3.1-2 on page 47b replaces Figures 3.1-2a b and c, which specify Operating Limit MCPR versus scram time ratio. Since limits for P8x8R fuel can be applied to the entire core, only a single Figure, 3.1-2, is required.

11. In the Notes for Table 3.2-3 on page 73, two intercept values, 43% and 44%, are added to those specified in the definition for K. These values reflect the two additional Rod Block Monitor Trip Level Settings described in No. 4 above.

12. Figures 3.5-3, 3.5-4 and 3.5-5 on pages 135a, b and c, respectively, are deleted. These figures specify Maximum Average Planar Linear Heat Generation Rate versus Planar Average Exposure for those fuel types which are being discharged prior to Cycle 6.

13. Figure 3.5-10, specifying Maximum Average Planar Linear Heat Generation Rate versus Planar Average Exposure, is relabeled to include fuel added for Reload 5. The fuel added in both Reload 4 and Reload 5 is of type P8DRB299.

14. On page 245, the reference to discharged 8x8 fuel is deleted.

15. Lastly, in Section 5.5.B on page 246, a parenthetical sentence is added to the description of axial loading limits for the spent fuel storage pool. In order to clarify axial loading requirements, the statement specifies the average enrichment (3.3 w/o U-235) that corresponds to the maximum allowed axial loading in the pool.

II. Purpose of the Changes

The proposed changes revise the Appendix A Technical Specifications to account for the discharge of old fuel and the addition of new fuel for the Reload 5/Cycle 6 core. As noted above, most of the changes simply involve deletions of references to specific fuel types due to the discharge of all 8x8 fuel bundles. Substantive changes are described below.

1. MCPR Operating Limits

The Cycle 6 core will consist of 548 P8x8R bundles and 12 8x8R bundles. Since all remaining 8x8R bundles will be placed on the periphery of the core, where local power levels will never be limiting vis-à-vis all other radial locations, Minimum Critical Power Ratio (MCPR) operating limits for P8x8R fuel are applied on a core-wide basis. As noted in the supporting General Electric document (Attachment III and Reference 1), both initial MCPR's and MCPR's calculated for postulated transients at all exposures are, for P8x8R fuel, greater than or equal to those for 8x8R fuel. Hence, MCPR operating limits for P8x8R fuel are limiting for Cycle 6 at all exposures.

2. Operating Limit MCPR Versus Scram Time Ratio (τ)

As noted in Section I above, Figure 3.1-2 replaces Figures 3.1-2a, b, and c due to the applicability of P8x8R MCPR limits to the entire core. For Cycle 6, the new figure contains curves for three different exposure ranges (EOC-1 to EOC, EOC-2 to EOC-1, BOC to EOC-2).

3. Rod Block Monitor Trip Level Settings

As noted in Section I above, MCPR operating limits were specified at two additional Rod Block Monitor (RBM) Hi-Trip Level Settings:

$$S = .66W + 43\% \text{ ; and}$$

$$S = .66W + 44\%$$

Where: S = RBM Hi-Trip Level Setting, and
W = total recirculation loop flow

The RBM is designed to prevent fuel damage from a rod withdrawal error transient by comparing local power levels around a selected rod to average powers in the core. Rod movements are blocked if local power exceeds average power by a preset margin. The incorporation of two additional RBM Hi-Trip Level Settings in the Technical Specifications gives operators more latitude for performing rod pulls while maintaining existing safety margins. At any given time, only one of the allowed Hi-Trip Level Settings is programmed into the plant's process computer and the corresponding MCPR operating limit applied.

As noted on page 4 of the accompanying General Electric document (Attachment III and Reference 1), the two additional RBM Hi-Trip Level Settings, corresponding to rod block readings of 109% and 110%, allow longer rod withdrawals than did the previous maximum setting of 108%. Consequently, to guarantee that the MCPR safety limit of 1.07 is not transcended for any rod withdrawal error, MCPR operating limits at these settings must be maintained at the higher values shown in the proposed new table of MCPR Operating Limits for Incremental Cycle Core Average Exposure in Section 3.1.B.

Since transient analyses have been performed for the two additional RBM Hi-Trip Level Settings and appropriately higher MCPR operating limits established for them, operation under either of the two additional settings would provide the same conservatism and safety margins as would operation under any of the existing settings.

4. MAPLHGR Versus Planar Average Exposure

Figure 3.5-10, specifying maximum Average Planar Linear Heat Generation Rate versus Planar Average Exposure, applies to P8DRB299 fuel. Since this is the fuel type added to the core in both Reload 4 and Reload 5, the same curve is used to represent both reloads.

5. Spent Fuel Pool Average Lattice Enrichment

The parenthetical statement added to page 246, section 5.5B, is designed to clarify spent fuel pool axial loading requirements by specifying an average enrichment corresponding to the axial loading limit.

III. Impact of the Changes

Because existing safety limits and conservatisms are maintained with the incorporation of the above changes, approval of the proposed amendment will not have negative safety implications. The two additional Rod Block Monitor Trip Level Settings will potentially allow more latitude in control rod withdrawal maneuvers while maintaining existing core safety margins.

IV. Implementation of the Changes

Implementation of the changes, as proposed, will not impact the ALARA or Fire Protection programs at FitzPatrick. Moreover, the changes will not impact the environment.

V. Conclusion

The incorporation of these changes: a) will not increase the probability or the consequences of an accident or malfunction of equipment important to safety as evaluated previously in the Safety Analysis Report; b) will not increase the possibility of an accident or malfunction of a type other than that evaluated previously in the Safety Analysis Report; c) will not reduce the margin of safety as defined in the basis for any Technical Specification; d) does not constitute an unreviewed safety question, and e) involves no Significant Hazards Considerations, as defined in 10 CFR 50.92.

VI. References

1. "Supplemental Reload Licensing Submittal for James A. FitzPatrick Nuclear Power Plant Reload 5," General Electric report Y1003J01A56, Rev. 0, March 1983.

2. "General Electric Standard Application for Reactor Fuel (GESTAR), "NEDE-24011-P-A-4, January 1982.

3. "Loss-of-Coolant Accident Analysis Report for James A. FitzPatrick Nuclear Power Plant (Lead Plant)," July 1977, NEDO-21662 (as amended).