ATTACHMENT I TO JPN-88-037

PROPOSED TECHNICAL SPECIFICATION CHANGES REGARDING RELOAD 8/CYCLE 9 (JPTS-88-016)

New York Power Authority

JAMES A. FITZPATRICK NUCLEAR POWER PLANT
Docket No. 50-333
DPR-59

LIST OF FIGURES

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 ≤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 uncertanties 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 < 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 x 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 x 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)

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%	1.25	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%	1.25	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 MCPR values (as a function of τ) is as given in Figure 3.1-2.

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

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

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

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 ΔW = 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.

Figure 3.1-1

K_f FACTOR

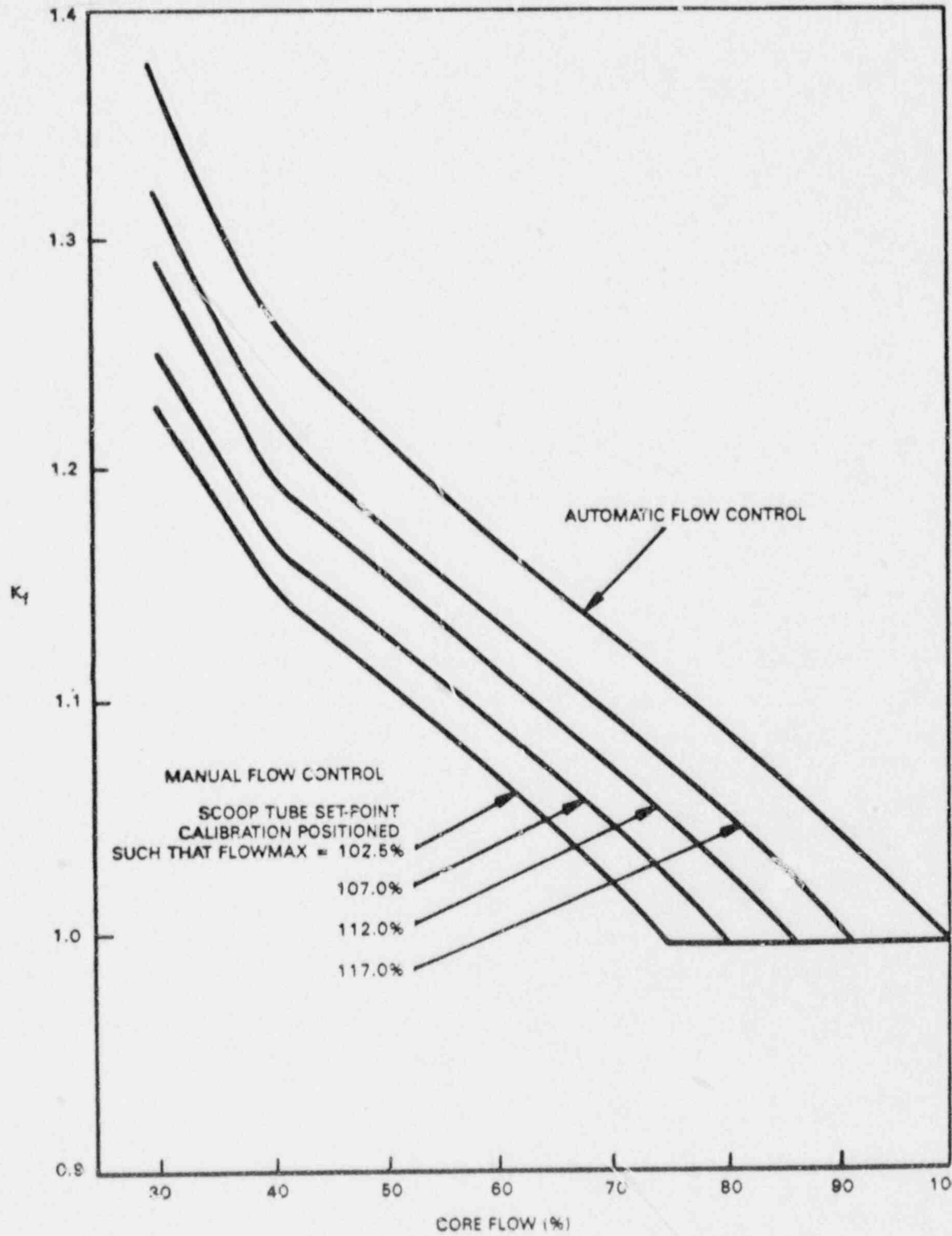

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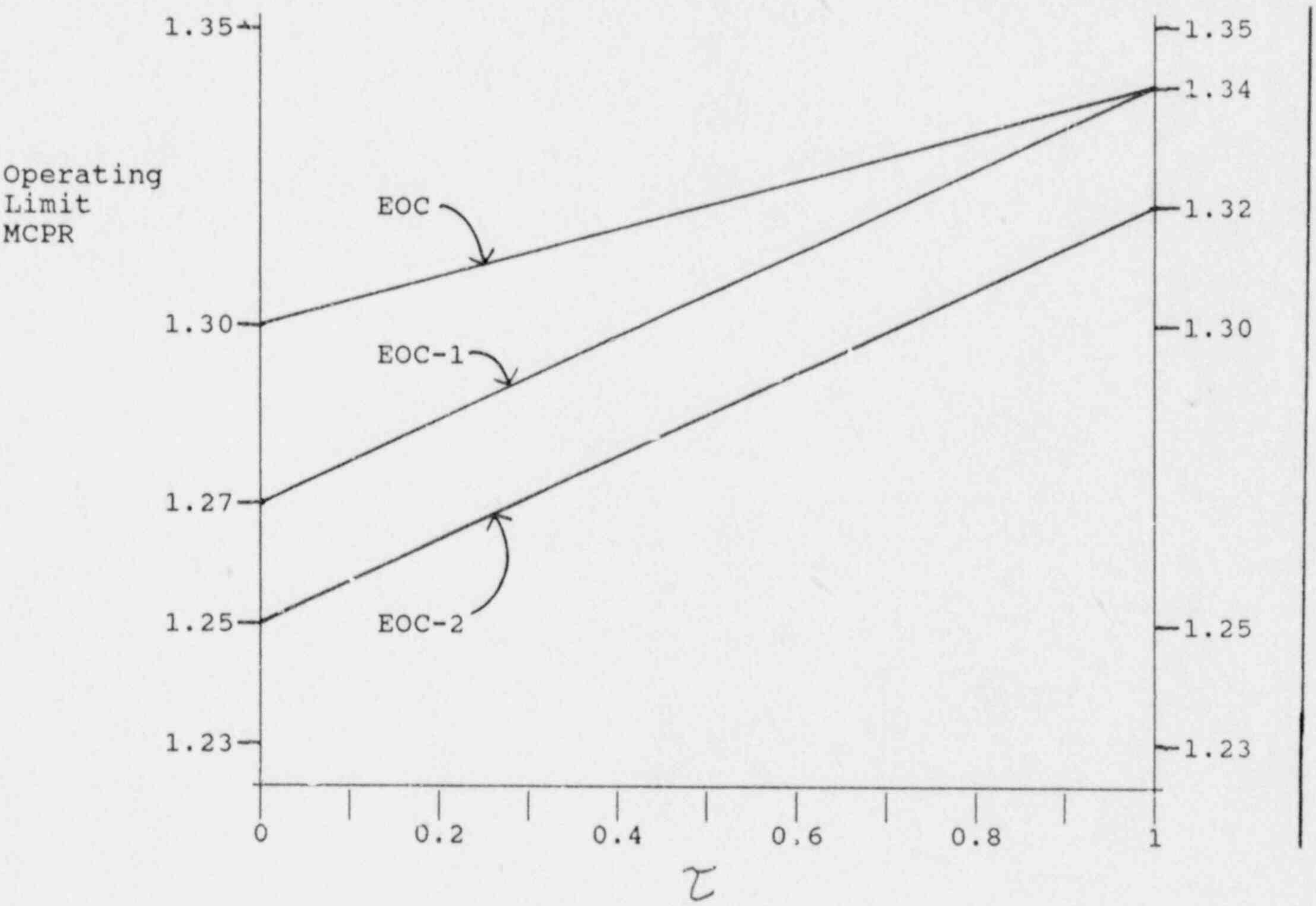
Figure 3.1-2

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

FOR ALL FUEL TYPES



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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 ± 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 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.

Figure 3.5-10

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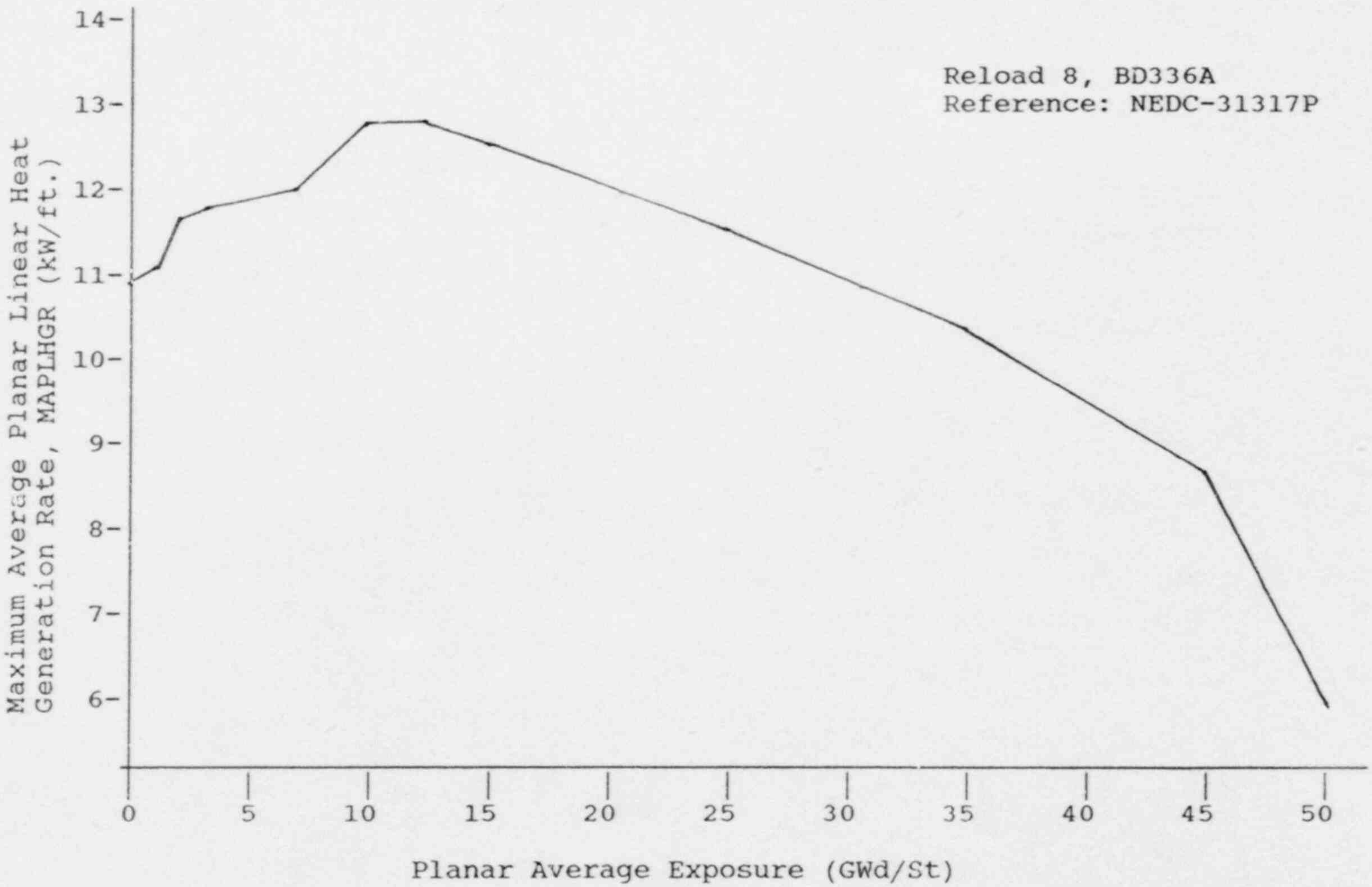
Figure 3.5-13

<u>Maximum Average Planar Linear Heat Generation Rate (MAPLHGR)</u>
<u>Versus Average Planar Exposure</u>



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

This curve represents the limiting exposure dependent MAPLHGR values.

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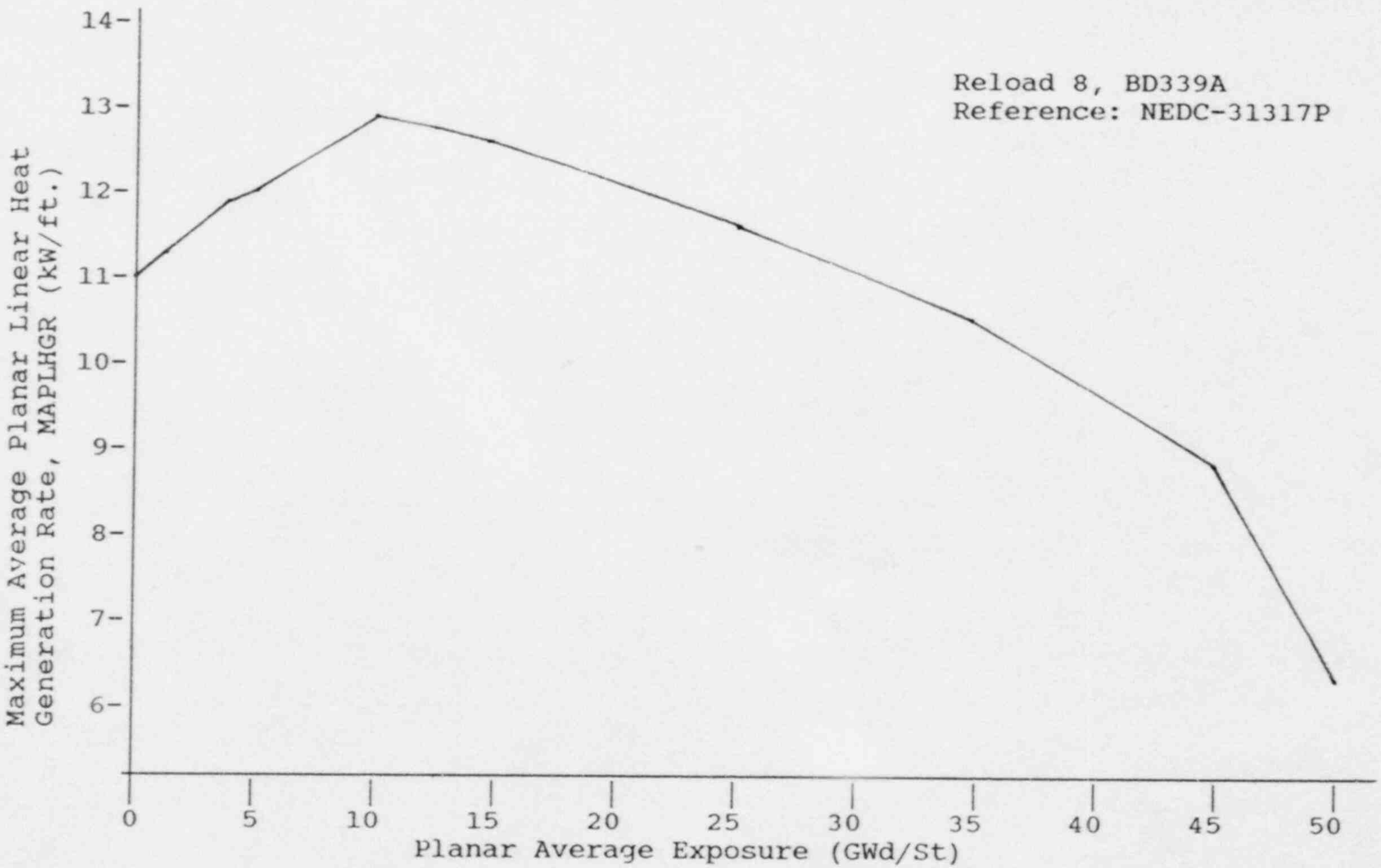
Figure 3.5-14

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.

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 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, 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.

ATTACHMENT II TO JPN-88-037

SAFETY EVALUATION FOR PROPOSED TECHNICAL SPECIFICATION CHANGES REGARDING RELOAD 8/CYCLE 9 (JPTS-88-016)

New York Power Authority

JAMES A. FITZPATRICK NUCLEAR POWER PLANT
Docket No. 50-333
DPR-59

I. DESCRIPTION OF THE PROPOSED CHANGES

The proposed changes to the James A. FitzPatrick Technical Specifications revise pages vii, 7, 12, 13, 31, 43a, 47a, 47b, 123, 130, 135h, and 245, and adds two new Figures, 3.5-13 and 14, on new pages 135k and 135L.

Page vii, List of Figure

[a] Replace entry 3.5-10 with "(Deleted)"

[b] Add two new entries 3.5-13 and 3.5-14. The figures will reside on new pages 135k and 135L. Entries 3.5-13(14) will read as follows:

MAPLHGR Versus Planar Average Exposure Reload 8, BD336A
MAPLHGR Versus Planar Average Exposure Reload 8, BD339A

Page 7, §1.1.A. Fuel Cladding Integrity

[c] In the first sentence, replace "1.07" with "1.04."

[d] In the second sentence, replace "An MCPR Limit of 1.08" with "An MCPR safety limit of 1.05."

Page 12, Bases for §1.1. Fuel Cladding Integrity

[e] In two places, replace "1.07" with 1.04."

Bases for §1.1.A.

[f] Replace "1.07" with 1.04."

Page 13, Bases for §1.1.B. Core Thermal Power Limit

[g] Replace this section in its entirety with:

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 x 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 x 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.

Page 31, §3.1.B.1. MCPR Operating Limit for Incremental Cycle Core Average Exposure

[h] Replace the values in this table with values specific for FitzPatrick Cycle 9. The new values can be found in Attachment I.

Page 43a, Notes for Table 3.1-1

[i] In note 14, correct the spelling of "signal."

Page 47a, Figure 3.1-1 K_f Factor

[j] Replace the existing figure with one approved for use with the GEXL-PLUS correlation. See Attachment I for the new figure.

Page 47b, Figure 3.1-2 Operating Limit MCPR Versus τ for All Fuel Types

[k] Replace the existing figure with one specific for Cycle 9 operation. See Attachment I for the new figure.

Page 123, §3.5.H. Average Planar Linear Heat Generation Rate (APLHGR)

[l] Replace "Figures 3.5-10 through 3.5-12" with "Figures 3.5-11 through 3.5-14"

Page 130, Bases for §3.5.H. Average Planar Linear Heat Generation Rate (APLHGR)

[m] In the top paragraph of the right column, make the following changes:

Replace "Figures 3.5-10 through 3.5-12" with "Figures 3.5-11 through 3.5-14;" replace "Reload 5 and 6 fuel" with "Reload 6 fuel;" and replace "NEDC-31317P for Reload 7 fuel" with "NEDC-31317P (as amended) for Reload 7 and 8 fuel."

Page 135h, Figure 3.5-10

[n] Delete this figure. Insert "(This page is intentionally blank.)"

Pages 135k and 135L, Figures 3.5-13 and 3.5-14
Maximum Average Planar Linear Heat eneration Rate (MAPLHGR) Versus Planar Average Exposure - Fuel Types BD336A and BD339A

[o] Add new figures prescribing the MAPLHGR limit vs. Average Planar Exposure for fuel types BD336A (page 135k) and BD339A (page 135L), added as Reload 8.

Page 245, §5.2.A. Reactor

[p] This section is revised to reflect the fuel types in the FitzPatrick Cycle 9 core. See Attachment I for the text.

II. PURPOSE OF THE PROPOSED CHANGES

The purpose of the proposed changes is to support plant start-up and operation after the Reload 8/Cycle 9 refueling outage. During this outage, 184 fuel bundles are to be removed from the reactor core and replaced by new fuel. The changes to the Technical Specifications involve deleting specifications associated with the discharged fuel and with Cycle 8 specific analyses, and replacing them with ones which are appropriate for the new fuel and are based on Cycle 9 specific analyses.

To simplify the discussion of the proposed changes, the 16 individual changes on 14 technical specification pages are grouped into three categories:

A) Changes to reflect the removal of fuel type P8DRB299 from the reactor core and its replacement with fuel types BD336A and BD339A;

B) Changes to reflect the Cycle 9 specific transient and accident analyses, use of the GEXL-PLUS critical power correlation, and the new fuel cladding integrity safety limit;

C) Miscellaneous changes (e.g., correction of a typographical error and clarification of text).

CATEGORY A

184 fuel bundles are to be removed from the reactor core and placed in the spent fuel pool for storage. Replacing these fuel bundles in the core will be 152 bundles of fuel type BD336A and 32 bundles of fuel type BD339A. The two new fuel types are General Electric's GE8X8EB design and are mechanically identical to the Reload 7 fuel. The U_{235} enrichment and gadolinium content are varied to support cycle specific needs and to improve fuel economy.

The changes to the Technical Specifications involve changes to the List of Figures, MAPLHGR figures and reactor design sections. The following changes fall into this category: a, b, l, m, n, o, and p.

CATEGORY B

Cycle 9 specific transient analyses, performed by General Electric Co. (GE), determine the operating limits for Cycle 9. The analyses were performed with GE's GEXL-PLUS thermal correlation and the revised safety limit MCPR of 1.04. The results of these analyses are contained in Reference 3 and are included in this application as Attachment II.

The changes to the Technical Specifications involve changes to the MCPR sections and Bases, MCPR operating limit table, the MCPR figure, and the K_f curves. The following changes fall into this category: c, d, e, f, h, j, and k.

CATEGORY C

One typographical error (the misspelling of the word "signal") is corrected. Change [i] to the Technical Specifications reflects this correction.

In NRC Inspection Report 50-333/78-05 (Reference 4), an NRC inspector considered the Technical Specification Bases for §1.1.B to be unclear. This section discusses the reactor thermal power limit when the reactor pressure is below 785 psig. The Authority committed to submit a revision to this bases section in this reload application submittal (Reference 5). Change [g] to the Technical Specifications is that revision.

Changed pages vii and 123, contained in Attachment I to this application, carry over errors which currently exist in the Technical Specifications. The Authority has previously submitted an application for amendment to the Technical Specifications to correct these and other administrative and typographical errors. (Reference 6)

III. IMPACT OF THE PROPOSED CHANGES

The overall impact of the proposed changes would be to allow start-up and operation of the FitzPatrick Nuclear Power Plant following the upcoming Reload 8/Cycle 9 refueling outage. This outage is currently scheduled to begin in late August 1988 and last until early November.

To simplify the discussion of the impact of each of the individual changes, this evaluation will address the three categories of changes that were previously defined in Section II above.

CATEGORY A

The 184 fuel bundles of Reload 8 are of fuel bundle type GE8X8EB. 152 of these bundles are designated BD336A, and the 32 others are designated BD339A. These fuel bundles incorporate the design features described in Reference 7.

The fuel to be added in Reload 8 is similar to that added in Reload 7 in that it contains several lattice types of varying gadolinium content. To determine the proper MAPLHGR value for a particular axial location in a fuel bundle, the MAPLHGR tables contained in Reference 8 will be programmed into the plant process computer and backup computer system. When hand calculations are necessary, the most limiting enriched uranium lattice MAPLHGR value is applicable. The exposure dependent limiting MAPLHGR values are shown in Figures 3.5-13 and 3.5-14 on pages 135k and 135l for fuel types BD336A and BD339A respectively. The tables used to generate these figures are included in Attachment III.

The MAPLHGR curve for fuel type P8DRB299 (Figure 3.5-10 on page 135h) is deleted since all fuel bundles of this type are being removed from the core.

CATEGORY B

The changes in this category reflect the Cycle 9 specific transient and accident analyses performed by General Electric Co. for FitzPatrick. The results of these analyses are contained in Reference 3 and are included in this application as Attachment III.

The analyses were performed with GE's GEXL-PLUS critical heat flux correlation. This correlation provides a more accurate determination of the critical heat flux. Implementation of the GEXL-PLUS correlation increases plant fuel cycle efficiency and operating flexibility without reducing the current safety margins (Reference 9). Use of the GEXL-PLUS correlation was approved generically for GE fueled BWRs by the NRC in Amendment 15 to Reference 7.

The analyses required for Cycle 9 were performed with the revised safety limit minimum critical power ratio (MCPR) of 1.04, instead of the previous safety limit MCPR of 1.07. The safety limit MCPR of 1.04 is a result of the statistical analyses performed by GE for reactor cores which are operated with a second successive reload of high enrichment, high R-factor, 8x8 fuel. The analyses associated with the new safety limit predict that 99.9% of the fuel rods in the core would avoid boiling transition. This is the same criteria that was previously applied to the safety limit of 1.07 and therefore, the change does not reduce any margin of safety. Use of the revised MCPR safety limit of 1.04 was generically approved by the NRC in Amendment 14 to Reference 7.

CATEGORY C

Correction of the misspelling of "signal" has no impact on plant operation and is being made to correct the error.

Clarification of the Bases for §1.1.B. has no impact on the plant since no change is being made to the safety limit power level for the Core Thermal Power Limit when the reactor pressure is below 785 psig. The Authority concurs with the NRC that the existing Bases section is unclear. The proposed replacement to this section was developed with the assistance of General Electric Co. It attempts to more clearly describe the theoretical and experimental bases for the existing power level limit than the existing paragraph.

IV. EVALUATION OF SIGNIFICANT HAZARDS CONSIDERATION

Operation of the FitzPatrick Plant in accordance with the proposed Amendment would not involve a significant hazards consideration as stated in 10 CFR 50.92 since it would not:

1. involve a significant increase in the probability or consequences of an accident previously evaluated. NRC approved methodologies and codes have been used to perform all analyses concerning the General Electric Co. fuel to be loaded at this refueling (Reference 7). The fuel design has been reviewed and approved for use at FitzPatrick under the constraints and methodologies detailed in Reference 7. There are no unique aspects of this fuel or its application which have not undergone prior NRC review and approval. The refueling of the FitzPatrick reactor and Cycle 9 operation does not increase the probability or consequences of any accident previously evaluated.

2. create the possibility of a new or different kind of accident from any accident previously evaluated. Refueling the FitzPatrick reactor is a periodic evolution performed in accordance with appropriate procedures and controlled by the Technical Specifications. The fuel bundles inserted as Reload 8 are mechanically identical to those inserted in Reload 7 and will not create the possibility of a new or different type of accident. The nuclear characteristics of the individual fuel bundles and the core loading pattern have been fully analyzed by the General Electric Co. and do not create the possibility of a new or different type of accident. The assemblies have been fully reviewed and approved for use in power reactors by the NRC (Reference 7).

3. involve a significant reduction in a margin of safety. The analyses performed in support of this reload assure maintenance of all existing margins of safety. These analyses have resulted in core wide (MCPR) and bundle specific (MAPLHGR) limits for General Electric Co. fuel which, when applied to the reloaded core, assure operation within the design criteria previously approved in Reference 7. The revised MCPR safety limit provides the same margin of safety as the previous

safety limit in preventing boiling transition. This change was previously approved by the NRC for use in GE fueled BWR reactors (Reference 7).

In the April 6, 1983, FEDERAL REGISTER (48FR14870), the NRC published examples of license amendments that are not likely to involve significant hazards considerations. Example number (iii) of that list is applicable to this proposed change and states in part:

> For a nuclear power reactor, a change resulting from a nuclear reactor core reloading, if no fuel assemblies significantly different from those found previously acceptable to the NRC for a previous core at the facility in question are involved.

V. IMPLEMENTATION OF THE PROPOSED CHANGE

Implementation of the proposed changes will not impact the ALARA or Fire Protection Programs at FitzPatrick, nor will the changes impact the environment.

VI. CONCLUSION

The change, as proposed, does not constitute an unreviewed safety question as defined in 10 CFR 50.59. That is, it:

A. will not change the probability nor the consequences of an accident or malfunction of equipment important to safety as previously evaluated in the Safety Analysis Report;

B. will not increase the possibility of an accident or malfunction of a different type from any previously evaluated 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 consideration, as defined in 10 CFR 50.92.

VII. REFERENCES

1. James A. FitzPatrick Nuclear Power Plant Updated Final Safety Analysis Report.

2. James A. FitzPatrick Nuclear Power Plant Safety Evaluation Report (SER), dated November 20, 1972, and Supplements.

3. General Electric Co. Report, "Supplemental Reload Licensing Report for James A. FitzPatrick Nuclear Power Plant - Reload 8 (Cycle 9)," 23A5898, Revision 0, June, 1988. (Included as Attachment III)

4. NRC letter, E. J. Brunner to J. D. Leonard, Jr. (PASNY), dated March 14, 1978, transmitting NRC Inspection Report 50-333/78-05: Page 9, Item 6, Technical Specifications Review - Open Item 78-05-07.

5. NRC letter, E. C. Wenzinger to R. J. Converse (NYPA), dated March 29, 1988, transmitting NRC Inspection Report 50-333/88-01: Enclosure 2, Item 2, Previous Inspection Findings - Inspection Followup Item 78-05-07.

6. NYPA letter, J. C. Brons to the NRC, JPN-88-023, dated May 27, 1988, containing an application for amendment to the FitzPatrick Technical Specifications regarding Administrative Changes.

7. General Electric Licensing Topical Report, "GESTAR-II General Electric Standard Application for Reactor Fuel," NEDE 24011-P-A-8, May 1986.

8. General Electric Co. Report, "James A. FitzPatrick Nuclear Power Plant SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis," NRDC-31317P, Errata and Addenda No. 2, June 1988. (Included as Attachment IV)

9. General Electric Co. letter, J. S. Charnley to C. O. Thomas (NRC), dated January 23, 1986, concerning Amendment 15 to GESTAR-II (Implementation of GEXL-PLUS critical power correlation).

ATTACHMENT III TO JPN-88-037

SUPPLEMENTAL RELOAD LICENSING REPORT FOR JAMES A. FITZPATRICK NUCLEAR POWER PLANT RELOAD 8 (CYCLE 9) (JPTS-88-016)

New York Power Authority

JAMES A. FITZPATRICK NUCLEAR POWER PLANT
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ATTACHMENT IV TO JPN-88-037

JAMES A. FITZPATRICK NUCLEAR POWER PLANT SAFER/GESTR-LOCA LOSS-OF-COOLANT ACCIDENT ANALYSIS ERRATA AND ADDENDA No. 2 (JPTS-88-016)

New York Power Authority

JAMES A. FITZPATRICK NUCLEAR POWER PLANT
Docket No. 50-333
DPR-59