April 3, 1987

Docket No. 50-333

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

Dear Mr. Brons:

The Commission has issued the enclosed Amendment No. 109 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 application dated December 23, 1986, as supplemented March 13, 1987.

The amendment changes the Technical Specifications to permit fuel reloading and Cycle 8 operation. Included in the Cycle 8 core will be four Westinghouse QUAD+ demonstration fuel assemblies.

A copy of our Safety Evaluation supporting the revisions to the Technical Specifications is provided in Enclosure 1. Enclosure 2 contains our Safety Evaluation regarding the inclusion of the four QUAD+ demonstration assemblies in the Cycle 8 core. Notice of Issuance will be included in the Commission's Bi-Weekly Federal Register Notice.

Sincerely,

Original Signed By

Harvey I. Abelson, Project Manager
BWR Project Directorate #2
Division of BWR Licensing

Enclosures:
1. Amendment No. 109 to
   License No. DPR-59
2. Safety Evaluation Supporting Amendment
3. Safety Evaluation Regarding QUAD+

cc w/enclosures:
See next page

DISTRIBUTION:

Docket File	SNorris	BGrimes	OPA
NRC PDR	HAbelson	TBarnhart (4)	Plant File
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ELJordan	ACRS (10)	EButcher	

DBL:PD#2	DBL:PD#2	OGC - Bethesda	DBL:PD#2:D
SNorris:	HAbelson		DMuller
3/26/87	3/26/87	3/27/87	4/2/87

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

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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. 109
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 December 23, 1986, as supplemented March 13, 1987, 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. 109 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Daniel R. Muller

Daniel R. Muller, Director
BWR Project Directorate #2
Division of BWR Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 3, 1987

ATTACHMENT TO LICENSE AMENDMENT NO. 109

FACILITY OPERATING LICENSE NO DPR-59

DOCKET NO. 50-333

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised areas are indicated by marginal lines.

Pages

vii
6
9
10a
13
30f
31
31a
35
43
47b
123
124
130
135g
135h
135i
135j-added
245

LIST OF FIGURES

Amendment No. ~~14~~, ~~22~~, ~~43~~, ~~64~~, ~~72~~, ~~74~~, ~~88~~, ~~98~~ 109 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 14.4 KW/ft for GE8x8EB fuel and 13.4 KW/ft for the remainder.

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.

1.1 (cont'd)

D. <u>Reactor Water Level (Hot or Cold Shutdown Conditions)</u>

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:

$$S \leq (0.66\ W + 54\%) \times (FRP/MFLPD)$$

for two loop operation or,

$$S \leq (0.66\ W + 54\% - 0.66\ W)(FRP/MFLPD)$$

for single loop operation

Where:

FRP = fraction of rated thermal power (2436 MWt)

MFLPD = maximum fraction of limiting power density where the limiting power density is 14.4 KW/ft for GE8X8EB fuel and 13.4 KW/ft for the remainder.

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:

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

2.1 (cont'd)

MFLPD = maximum fraction of limiting power density where the limiting power density is 14.4 KW/ft for GE8X8EB fuel and 13.4 KW/ft for the remainder.

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.

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 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. <u>Core Thermal Power Limit (Reactor Pressure < 785 psig)</u>

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 by 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 LIMITING CONDITIONS FOR OPERATION

3.1 REACTOR PROTECTION SYSTEM

Applicability:

Applies to the instrumentation and associated devices which initiate the reactor scram.

Objective:

To assure the operability of the Reactor Protection System.

Specification:

A. The setpoints, minimum number of trip systems, minimum number of instrument channels that must be operable for each position of the reactor mode switch shall be as shown on Table 3.1-1. The design system response time from the opening of the sensor contact to and including the opening of the trip actuator contacts shall not exceed 50 msec.

B. Minimum Critical Power Ratio (MCPR)

During reactor power operation, the MCPR operating limit shall not be less than that shown below:

1. When surveillance requirement 4.1.E is met. ($\tau_{AVE} \leq \tau_B$)

4.1 SURVEILLANCE REQUIREMENTS

4.1 REACTOR PROTECTION SYSTEM

Applicability:

Applies to the surveillance of the instrumentation and associated devices which initiate reactor scram.

Objective:

To specify the type of frequency of surveillance to be applied to the protection instrumentation.

Specification:

A. Instrumentation systems shall be functionally tested and calibrated as indicated in Tables 4.1-1 and 4.1-2 respectively.

B. Maximum Fraction of Limiting Power Density (MFLPD)

The MFLPD shall be determined daily during reactor power operation at $\geq$ 25% rated thermal power and the APRM high flux scram and Rod Block trip settings adjusted if necessary as required by Specifications 2.1.A.1.c and 2.1.A.1.d, respectively.

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%	1.23	1.29	1.30
S = .66W + 40%	1.23	1.29	1.30
S = .66W + 41%	1.23	1.29	1.30
S = .66W + 42%	1.27	1.29	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

*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.

3. During single loop operation, the operating limit MCPR shall be increased by 0.01 as specified in Specification 3.1.B.1 or 3.1.B.2 to reflect the increase in safety limit MCPR. (See Specification 1.1.A).

4. During Reactor power operation with core flow less than 100% of rated, the MCPR operating limit shall be multiplied by the appropriate K_f as shown in Figure 3.1.1.

5. If anytime during reactor operation at 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 begin 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 prescribed limits.

the i^{th} surveillance and τ_i = average scram time to notch position 38 of all rods measured in the i^{th} surveillance test.

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

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

where μ = mean of the distribution for the average scram insertion time to the pickup of notch position 38 = 0.706 sec.

σ = standard deviation of the distribution for average scram insertion time to the pickup of notch position 38 = 0.016 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.

3.1 BASES (cont'd)

Turbine control valves fast closure initiates a scram based on pressure switches sensing electro-hydraulic control (EHC) system oil pressure. The switches are located between fast closure solenoids and the disc dump valves, and are set relative (500 < P < 850 psig) to the normal (EHC) oil pressure of 1,600 psig so that based on the small system volume, they can rapidly detect valve closure or loss of hydraulic pressure.

The requirement that the IRM's be inserted in the core when the APRM's read 2.5 indicated on the scale in the start-up and refuel modes assures that there is proper overlap in the neutron monitoring system functions and thus, that adequate coverage is provided for all ranges of reactor operation.

B. The limiting transient which determines the required steady state MCPR limit depends on cycle exposure. The operating limit MCPR values as determined from the transient analysis in the current reload submittal for various core exposures are given in Specification 3.1.B.

The ECCS performance analyses assumed reactor operation will be limited to MCPR = 1.20, as described in NEDO-21662 and NEDC-31317P. The Technical Specifications limit operation of the reactor to the more conservative MCPR based on consideration of the limiting transient as given in Specification 3.1.B.

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 14.4 KW/ft for GE8X8EB fuel and 13.4 KW/ft for the remainder.

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 rated thermal power.

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.

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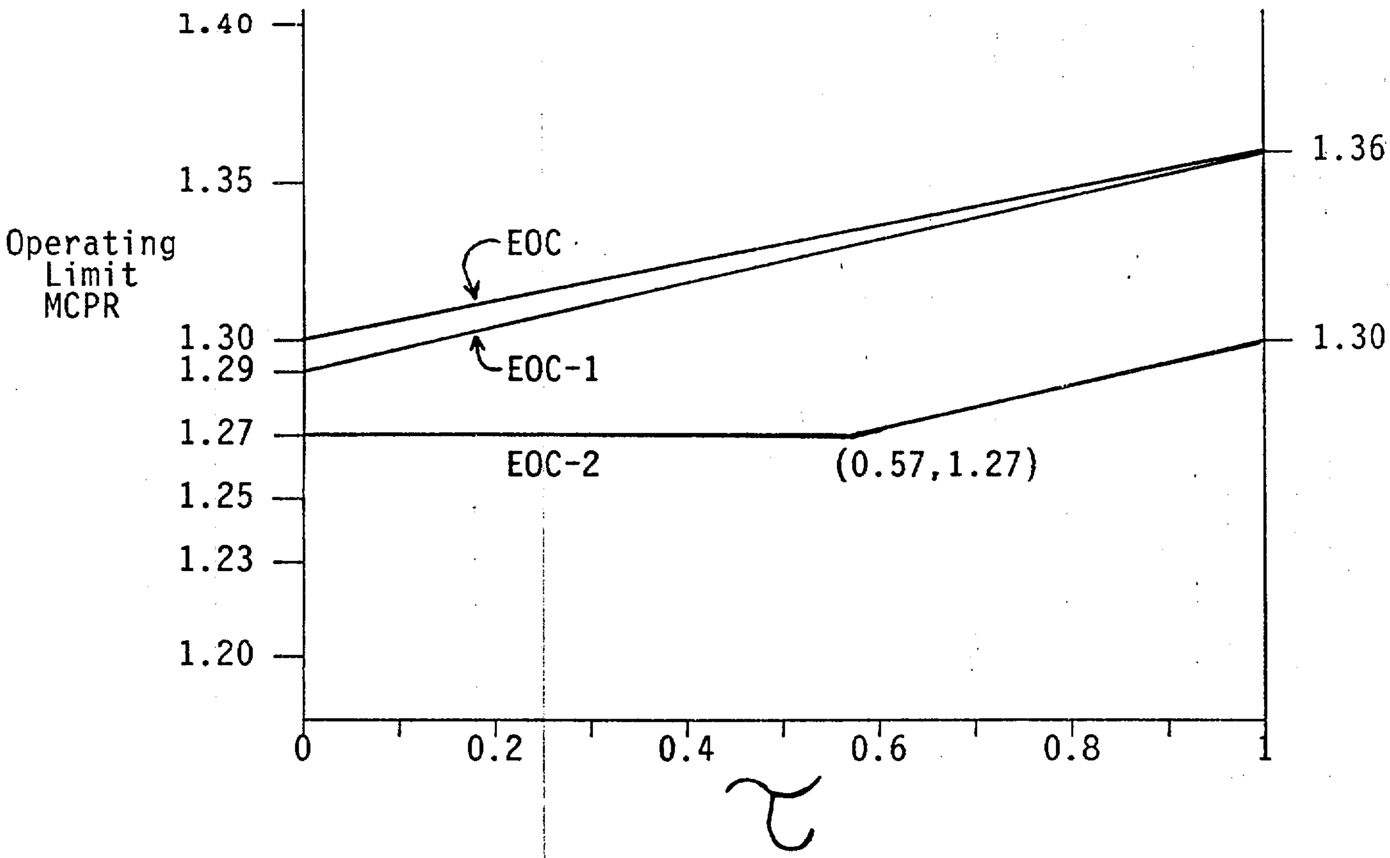
Figure 3.1-2

Operating Limit MCPR

Versus τ (Defined in Section 3.1.B.2)

FOR ALL FUEL TYPES



Amendment No. ~~64~~, ~~74~~, ~~79~~, ~~88~~, 109

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 3.5-10 through 3.5-12 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 ≥ 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 14.4KW/ft for GE8x8EB fuel and 13.4 kW/ft for the remainder of the fuel.

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.

3.5 BASES (cont'd)

requirements for the emergency diesel generators.

G. Maintenance of Filled Discharge Pipe

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

H. Average Planar Linear Heat Generation Rate (APLHGR)

This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the limit specified in 10 CFR 50 Appendix K.

The peak cladding temperature following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is only dependent secondarily on the rod to rod power distribution within an assembly. Since expected local variations in power distribution within a fuel assembly affect the calculated peak clad temperature by less than $\pm$ 20°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 3.5-10 through 3.5-12. Approved limiting values of APLHGR as a function of fuel type are given in NEDO-21662-2 (as amended) for Reload 5 and 6 fuel. Approved limiting values of APLHGR as a function of fuel and lattice types are given in NEDC-31317P for Reload 7 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 3.5-9

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Figure 3.5-10

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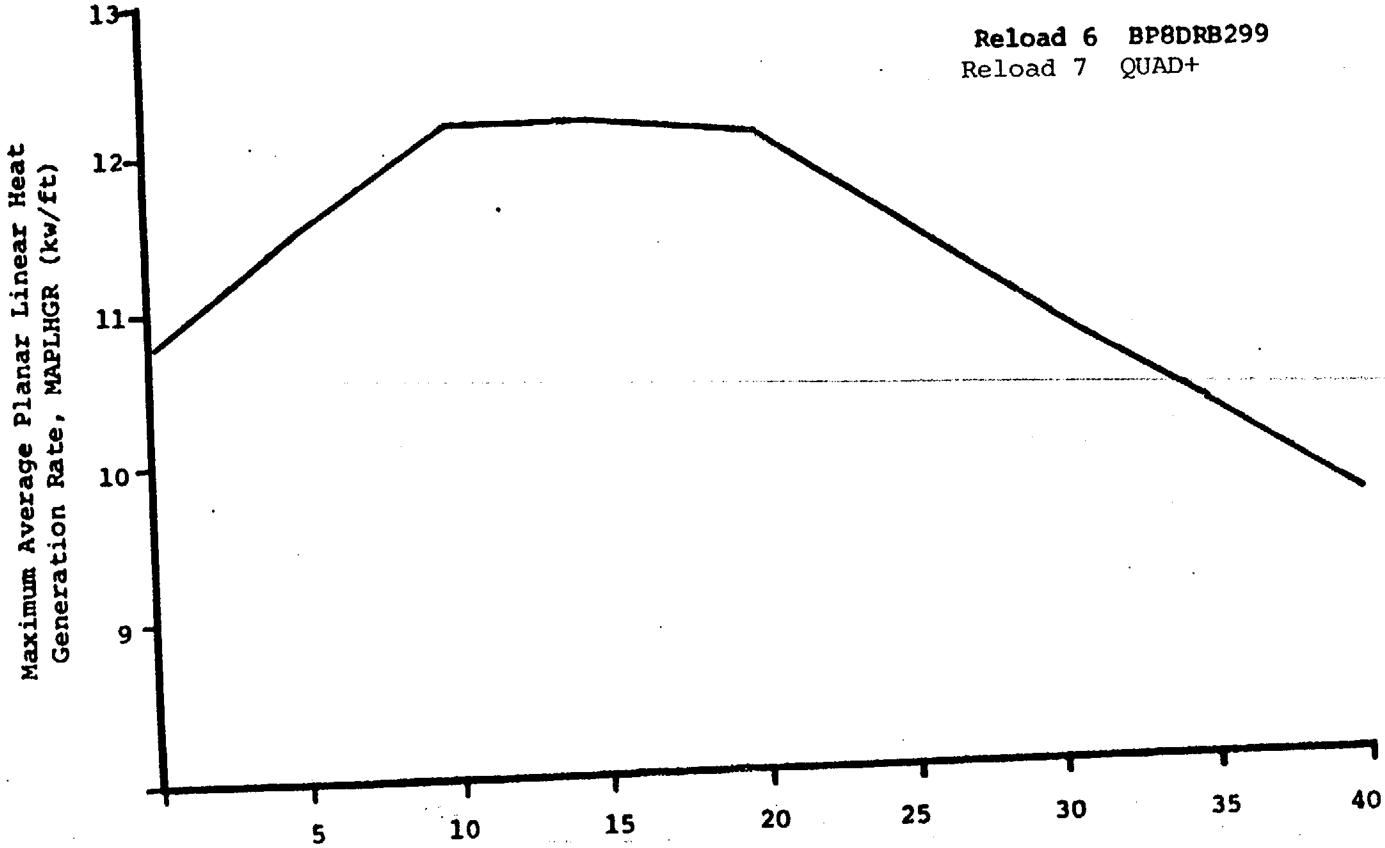
Figure 3.5-11



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

Reference: NEDO-21662-2
(As amended December 1984)
WCAP-11159

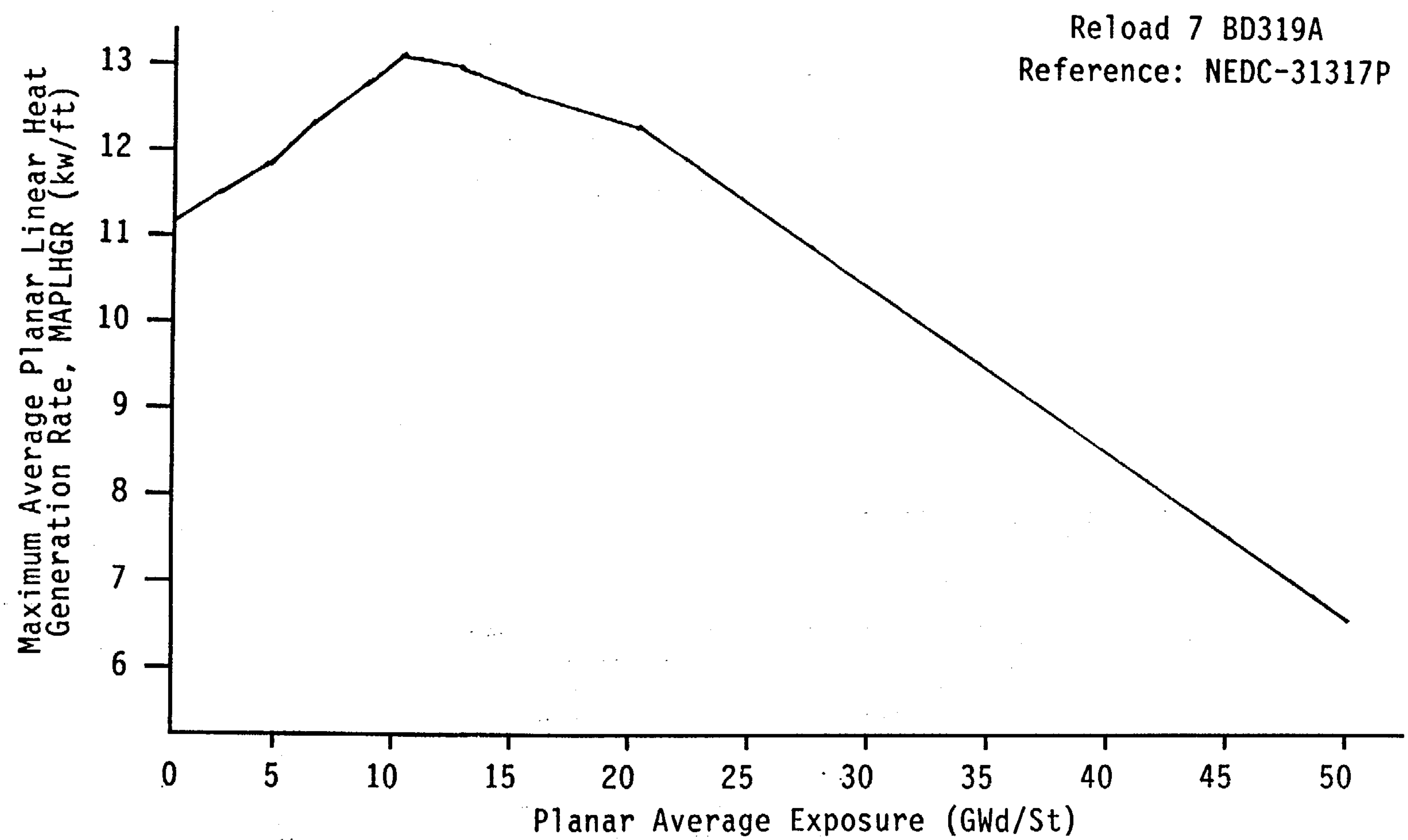
For single-loop operation, these MAPLHGR values are multiplied by 0.84.

JAFNPP

Figure 3.5-12

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.

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, four fuel types are present in the core: P8X8R, BP8X8R, GE8X8EB, and QUAD+. The GE fuel types are described in NEDO-24011. Both P8X8R and BP8X8R fuel types have 62 fuel rods and 2 water rods and GE8X8EB fuel type has 60 fuel rods and 4 water rods. The QUAD+ fuel type is described in WCAP-11159 and has 64 fuel rods.

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

5.3 REACTOR PRESSURE VESSEL

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

5.4 CONTAINMENT

A. The principal design parameters and characteristics for the primary containment are given in Table 5.2-1 of the FSAR.

B. The secondary containment is as described in Section 5.3 and the applicable codes are as described in Section 12.4 of the FSAR.

C. Penetrations of the primary containment and piping passing through such penetrations are designed in accordance with standards set forth in Section 5.2 of the FSAR.

5.5 FUEL STORAGE

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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 109 TO FACILITY OPERATING LICENSE NO. DPR-59

POWER AUTHORITY OF THE STATE OF NEW YORK

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

DOCKET NO. 50-333

1.0 INTRODUCTION

By letters dated December 23, 1986 and March 13, 1987 (Ref. 1 and 2), the Power Authority of the State of New York submitted proposed changes to the Technical Specifications for the James A. Fitzpatrick Nuclear Power Plant to permit reloading and operation for Reload 7/Cycle 8. In support of these changes, the submittal included a Safety Evaluation, as well as the General Electric (GE) Reports, "Supplemental Reload Licensing Submittal for the James A. Fitzpatrick Nuclear Power Plant Reload 7" (Ref. 3), and the GE Report, "Loss-of-Coolant Analysis for James A. Fitzpatrick Nuclear Power Plant" (Ref. 4). The staff has reviewed the submittals and has prepared the following evaluation.

2.0 EVALUATION

2.1 Reload Description

For Reload 7/Cycle 8, 188 irradiated fuel assemblies will be removed from the reactor core and replaced by 184 General Electric 8x8EB assemblies and 4 Westinghouse designed QUAD+ demonstration assemblies. These 4 QUAD+ assemblies will be placed at quadrant symmetric positions near the core periphery. The QUAD+ demonstration assemblies are designed to be compatible with the GE fuel. The staff has also reviewed use of the QUAD+ fuel assemblies. The results of the QUAD+ fuel evaluation are included in a separate evaluation.

2.2 Fuel Mechanical Design

The fuel (GE8x8EB) to be inserted into the core for Cycle 8 is similar to that customarily used for BWR reloads and is described in Reference 3. The mechanical design methodology is described in Reference 5 and was used in this design for the GE8x8EB fuel. Reference 5 has been approved by the staff (Ref. 6). We conclude that the fuel mechanical design for the GE8x8EB fuel is acceptable.

2.3 Nuclear Design

The nuclear design and analysis of the Cycle 8 reload was performed with methods and techniques which are described in Reference 5 and which are used in all reload analyses performed by GE. The results of the FitzPatrick analyses are within the range of those reload cores previously reviewed by the staff and found to be acceptable. We therefore conclude that the nuclear design and analysis of the Cycle 8 reload is acceptable.

2.4 Thermal-Hydraulic Design

The methods and procedures employed in the thermal-hydraulic (T-H) design and analysis of the Cycle 8 core are described in Reference 5. The value of 1.07 for the safety limit MCPR, approved in that reference, is used for Cycle 8. The methods and procedures used to obtain the operating limit MCPR are those described in Reference 5 and are acceptable.

2.5 Thermal-Hydraulic Stability

The issue regarding thermal-hydraulic stability has been resolved during the staff's review of one loop operation (Ref. 7). The licensee has changed the Technical Specifications which provide operating limits and surveillance requirements for thermal-hydraulic stability. As a result of our review, the staff finds that these revised Technical Specifications implement the recommendations of GE SIL-380 and are acceptable for both one and two loop operation.

2.6 Loss-of-Coolant Accident Analyses

The LOCA analyses in the reload were performed using the SAFER/GESTR code package and the application methodology described in Reference 8. In Reference 8, the staff has specified the necessary conditions for demonstrating applicability of the SAFER/GESTR methodology. These conditions are:

1. Calculation of a sufficient number of plant specific PCT points based on both nominal input values and Appendix K values to verify the shape of the PCT curves versus break size.

2. Confirmation that plant specific operating parameters have been bounded by the models and inputs used in the generic calculations.

3. Confirmation that the plant specific ECCS configuration is consistent with the referenced plant class ECCS configuration.

The licensee has reported the results of those analyses (Ref. 4) which are required to meet these conditions. Specifically, the analyses include break sizes from 0.05 ft^2 to the DBA recirculation suction line break (4.17 ft^2). Seven different break sizes were analyzed in conjunction with ECCS failure combinations. A total of 19 cases were evaluated to establish the trend of PCT curves (nominal and Appendix K) versus break size.

The input parameter values used for both the nominal and Appendix K cases are consistent with those used in the approved generic analyses. The ECCS configuration of FitzPatrick (4 LPCI, 2CS and 1 HPCI) is typical for a BWR 4. The results show that the DBA recirculation line break with battery failure is the limiting case. The calculated PCT is 1036° F when nominal input values are used and 1568° F when Appendix K input values are used. The input parameters, the ECCS configuration and the cases analyzed to establish the trend of PCT verse break size meet the requirements of Reference 8. Because the accident analyses have been performed using approved methods, and the results meet the staff's acceptance criteria, we conclude that these analyses are acceptable.

2.7 MCPR and MAPLHGR Limits

A safety limit MCPR has been imposed to assure that 99.9 percent of the fuel rods in the core will not experience boiling transition during normal operation and anticipated operational transients. As stated previously, the safety limit of 1.07 was used for Cycle 8.

To assure that the fuel cladding integrity safety limit MCPR will not be violated during any anticipated transient, the most limiting events were reanalyzed for this reload (Ref. 3) to determine which events result in the largest reduction in CPR. The operating limit MCPR was then established by adding the largest reduction factor in the CPR to the safety limit MCPR. Since acceptable methods (Ref. 5) have been used, we find the MCPR Technical Specification changes to be acceptable.

The MAPLHGR limits specified in the proposed Technical Specification changes are less than or equal to the bounding MAPLHGR used in the SAFER/GESTR-LOCA analysis (Ref. 4) and are, therefore, acceptable.

2.8 Technical Specification Changes

The Technical Specification changes proposed by the licensee reflect the new fuel for Cycle 8. These changes include LHGR limit, MCPR operating limit and MAPLHGR curve for the GE8x8EB fuel. These proposed changes are acceptable since they are based upon approved analytical methods as discussed above.

2.9 Evaluation Summary

Based on the review described above, we conclude that James A. FitzPatrick Nuclear Power Plant may be loaded and operated for Cycle 8. Our evaluation includes the presence of four QUAD+ bundles as lead test assemblies. This conclusion is based on the following:

1. The safety analyses have been performed by previously approved methods and procedures, except for those directly relating to the demonstration assemblies.

2. The use of the demonstration assemblies has been approved (see attached evaluation) subject to certain conditions. These conditions have been met for FitzPatrick Cycle 8 operation.

The Cycle 8 core meets all of the staff's acceptance criteria.

3.0 ENVIRONMENTAL CONSIDERATIONS

This amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

4.0 CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

REFERENCES

1. Letter, John C. Brons (Power Authority of the State of New York) to D. R. Muller (NRC), December 23, 1986.

2. Letter, John C. Brons (Power Authority of the State of New York) to NRC Document Control Desk, March 13, 1987.

3. Supplemental Reload Licensing Submittal for James A. FitzPatrick Nuclear Power Plant Reload 6, General Electric, 23A 4825, November 1986.

4. General Electric Company, NEDO-31317P, James A. FitzPatrick Nuclear Power Plant, "SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis," October 1986.

5. GESTAR II - "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A-6, July 1986.

6. Approval letter, D. G. Eisenhut (NRC) to R. Gridley (GE) dated May 12, 1978 and supplements thereto, forming Appendix C to Reference 4.

7. Letter, H. I. Abelson (NRC) to J. C. Brons (Power Authority of the State of New York), May 6, 1986.

8. NEDE-23785-1-PA, "The GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident" Volume I, II and III, General Electric Company, June 1984.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

"QUAD+" DEMONSTRATION ASSEMBLIES TO BE INCLUDED IN CORE FOR CYCLE 8

POWER AUTHORITY OF THE STATE OF NEW YORK

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

DOCKET NO. 50-333

1.0 INTRODUCTION

Westinghouse Nuclear Energy Systems (W) has prepared a report, WCAP-11159 (Ref. 1), on the demonstration "QUAD+" assemblies to be included in the James A. FitzPatrick Nuclear Power Plant (FitzPatrick) core for Cycle 8. This report describes the physical and operating characteristics of the assemblies, the restrictions to be placed on their operation at FitzPatrick, and the safety analyses relating to their use.

This report is similar to, and is in fact a supplement to, the W report WCAP-10507, "QUAD+ Demonstrations Assembly Report" (Ref. 2), which was reviewed by the staff in connection with the Browns Ferry 2 (BF2) Cycle 2 reload (Ref. 3). That review approved the use of the QUAD+ demonstration assemblies at BF2, specified that the guidelines stated in the report be followed and also specified that a 20 percent margin in power between the QUAD+ fuel and the lead assembly in the core (at full power) be maintained.

The QUAD+ assemblies and their operation proposed for FitzPatrick are very similar to that approved for BF2. There will be four assemblies, located in quadrant symmetric positions near the core periphery. Their operation will follow the approved guidelines, and calculations indicate that the required power margin will be maintained.

The FitzPatrick QUAD+ assemblies differ in some details from the BF2 assemblies. These differences will be discussed in the following evaluation. However, most characteristics of the fuel, its analysis, use, loading and operating limitations, and its safety evaluation, are essentially the same as that discussed in the BF2 safety evaluation (Ref. 3) and will not be repeated in detail here. Only a brief outline of that review will be presented here, with significant differences noted.

2.0 EVALUATION

The FitzPatrick Cycle 8 reload has been designed by the reload vendor, General Electric (GE), to have a GE 2.99 percent enriched fuel assembly at the location at which the four QUAD+ assemblies will be placed. Therefore, the QUAD+ fuel has been designed to match the relevant hydraulic and nuclear characteristics of such an assembly. The location of the QUAD+ assemblies has been determined based on the guidelines (mentioned above) provided for the BF2 demonstration and approved in the staff review of that proposal. These guidelines, which are also applicable to the FitzPatrick demonstration, are as follows:

1. QUAD+ demonstration assembly will not become a lead assembly.

2. QUAD+ demonstration assembly will not become limiting under transient conditions.

3. One QUAD+ demonstration assembly should be placed adjacent to a Local Power Range Monitor (LPRM) string.

4. QUAD+ demonstration assemblies should be loaded quarter-core symmetric.

5. QUAD+ demonstration assembly will not be loaded in the core cell with the analytically strongest rod for the rod drop accident.

6. QUAD+ demonstration assembly should not be next to a control rod at power during the first cycle.

2.1 Mechanical Design

The mechanical design is similar to that of the QUAD+ fuel approved for BF2, and similar design analyses were done, using similar methodologies. The fuel has been designed to be mechanically compatible with the other FitzPatrick fuel and components. There are some differences in design detail from BF2 fuel, intended to provide improved reliability and hydraulic performance. These include:

- Bottom nozzle changes to decrease leakage paths and improve flow to the water cross.

- Fuel tube processing changes and liner addition to improve PCI performance.

- Assembly changes to improve handling and replacement.

- Water cross thickness increase to improve strength.

- Two assemblies will have a revised spacer design to improve mechanical characteristics and coolant communication between flow channels.

These are straightforward mechanical changes, within the normal scope of a demonstration program, and have been analyzed by methodologies used in previously approved stages of the program. We find that these changes are acceptable.

2.2 Nuclear Design

The FitzPatrick core for Cycle 8 has been designed by GE with 2.99 percent average U235 enrichment assemblies in the four QUAD+ locations. Westinghouse has designed the nuclear characteristics of the QUAD+ to be compatible with that design. That approach is consistent with the approved approach for BF2.

The QUAD+ nuclear design for FitzPatrick is described in Ref. 1. The methodology used is the same as used for BF2 and approved for that limited type of application. As in the Browns Ferry 2 review report (Ref. 3), a wide range of QUAD+ and GE assembly nuclear characteristics were calculated and compared over relevant parameter ranges. These comparisons demonstrate the similarity or conservatism of this QUAD+ fuel as a replacement for the GE assemblies, and confirm that the replacement has no significant effect on the core nuclear behavior. It is concluded that the nuclear aspects of the replacement are acceptable. The calculated peak k (infinity) of the assemblies is such that the criteria for storage in the spent fuel pool are met.

The nuclear compatibility between the QUAD+ and the replaced GE fuel is such that the power distribution determined during operation for the core and the QUAD+ fuel in particular, via modeling with GE fuel, should be reasonably accurate. Thus the monitoring of the QUAD+ power level should be suitable to maintain the required 20 percent margin between the QUAD+ fuel and the lead assembly in the core at full power.

2.3 Thermal-Hydraulic Analysis

As discussed in the BF2 review (Ref. 3), the acceptability of the thermal-hydraulic design is based on hydraulic compatibility of the QUAD+ fuel and the replaced GE fuel, and on an acceptable Core Power Ratio (CPR) performance. The hydraulic compatibility and CPR analysis are discussed in the BF2 review. That review and its conclusions are also applicable for FitzPatrick. The FitzPatrick QUAD+ hydraulic characteristics are essentially the same. The pressure drop characteristics have been matched to the GE replaced fuel as with BF2. Similar hydraulic compatibility tests of the fuel were run. The CPR considerations discussed in the BF2 apply to FitzPatrick. The BF2 review of the information available on the CPR aspects of the QUAD+ fuel, along with the results of transient analyses, concluded that a margin of at least 20 percent in power should be maintained between the QUAD+ fuel and the lead assembly in the core when operating at full power. This margin requirement is also applicable for the FitzPatrick QUAD+ fuel. The licensee has indicated that this margin will be maintained and nuclear calculations indicate that this is feasible.

2.4 Transient and Accident Analyses

The transient and accident response of FitzPatrick with respect to the QUAD+ fuel is the same as that for BF2. The BF2 review (Ref. 3) and its conclusions are applicable to FitzPatrick. The core-wide transients and accidents, which depend on core-wide neutronic and thermal-hydraulic parameters, are not significantly changed by the presence of the four QUAD+ assemblies, and thus there is negligible affect on these events. However, the response of the QUAD+ fuel must be considered. The review of these events for BF2, considering the neutronic and thermal-hydraulic characteristics of the fuel, concluded that the 20 percent margin in assembly margin discussed above should be maintained, and that MAPLHGR limits obtained from LOCA calculations for the replaced GE assemblies may be conservatively applied to the QUAD+ fuel. These conclusions are also applicable for the FitzPatrick demonstration fuel.

2.5 Evaluation Summary

Based on the review described above, we conclude that WCAP-11159 (Ref. 1) presents sufficient information to support the use of up to four QUAD+ assemblies in the FitzPatrick core provided that the guidelines presented in Table 1 of that report (and summarized above) are adhered to and that a margin of at least 20 percent in power exists between the QUAD+ assembly and the lead assembly when the core is operating near full power. Any more extensive loading of QUAD+ assemblies into BWRs or changes in the limiting conditions for loading and operating the QUAD+ assemblies will be subject to review in considerably greater depth than is described in this evaluation.

REFERENCES

1. WCAP-11159 "Supplemental QUAD+ Demonstration Assembly Report for James A. FitzPatrick Nuclear Power Plant," December 1986, J. Foley and L. Mayhue.

2. WCAP-10507 "QUAD+ Demonstration Assembly Report," March 1984, L. Mayhue.

3. Letter (and enclosure) from M. Grotenhuis, NRC, to S. White, TVA, August 19, 1986, "Amendment No. 125 to Facility Operating License No. DPR-52 for the Browns Ferry Nuclear Plant, Unit 2."

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