January 29, 1982

Docket No. 50-333

Mr. George T. Berry
President & Chief Operating Officer
Power Authority of the State of
New York
10 Columbus Circle
New York, New York 10019

Dear Mr. Berry:

The Commission has issued the enclosed Amendment No. 64 to Facility Operating License No. DPR-59 for the James A. FitzPatrick Nuclear Power Plant. The enclosed Amendment is in response to your submittal dated November 18, 1981 regarding Reload 4/Cycle 5.

The enclosed Amendment modifies the FitzPatrick Technical Specifications to reflect our review of the following proposed changes: (1) revised maximum average planar linear heat generation rate (MAPLHGR) data, (2) power spiking penalty, and (3) use of the ODYN computer program. Not included herein are: (1) control rod drive surveillance and (2) Standby Gas Treatment System heater surveillance. These are unrelated to the reload itself and will be the subject of a future amendment.

Copies of the related Safety Evaluation and Notice of Issuance are enclosed.

Sincerely,

Philip J. Polk, Project Manager
Operating Reactors Branch #2
Division of Licensing

Enclosures:
1. Amendment No. 64 to DPR-59
2. Safety Evaluation
3. Notice

cc: w/enclosures
See next page

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NRC FORM 318 (10-80) NRCM 0240 OFFICIAL RECORD COPY USGPO: 1981—335-960

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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. 64
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 November 18, 1981 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. 64 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

DBVassallo

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

Attachment:
Changes to the Technical Specifications

Date of Issuance: January 29, 1982

ATTACHMENT TO LICENSE AMENDMENT NO. 64

FACILITY OPERATING LICENSE NO. DPR-59

DOCKET NO. 50-333

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages:

Remove	Insert
vii	vii
6	6
9	9
10	10
13	13
15	15
20	20
29	29
30	30
31	31
--	31a
35	35
43	43
--	47b
--	47c
--	47d
123	123
124	124
130	130
131	131
134	134
135	135
135a	135a
135b	135b
135c	135c
135d	135d
--	135e
--	135f
--	135g
--	135h
245	245

LIST OF FIGURES

Amendment No. ~~14~~, ~~22~~, ~~45~~, 64 surveillance tests, checks, calibrations, and examinations shall be performed within the specified surveillance intervals. These intervals may be adjusted ± 25 percent. The interval as pertaining to instrument and electric surveillance shall never exceed one operating cycle. In cases where the elapsed interval has exceeded 100 percent of the specified interval, the next surveillance interval shall commence at the end of the original specified interval.

U. Thermal Parameters

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

2. Fraction of Limiting Power Density - The ratio of the linear heat generation rate (LHGR) existing at a given location to the design LHGR. The design LHGR is 13.4 KW/ft for 8x8, 8x8R and P8x8R bundles.

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.

1.1 (cont'd)

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

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 in. (-146.5 in. indicated level) above the top of the 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 (MPLPD) greater than the fraction of rated power (FRP), the setting shall be modified as follows:

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

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 for 8x8, 8x8R, and P8x8R fuel.

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

(2) Fixed High Neutron Flux Scram Trip Setting

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

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

1.1 (cont'd)

2.1 (cont'd)

A.1.d. 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 (rated loop recirculation flow rate equals (34.2×10^6 lb/hr))

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

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

where:

FRP = fraction of rated thermal power (2436 MWt)

MFLPD = maximum fraction of limiting power density where the limiting power density is 13.4 KW/ft for 8x8, 8x8R and P8x8R fuel

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 operating (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 = 13.4 kw/ft for 8x8, 8x8R, and P8x8R fuel. 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 28×10^3 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 28×10^3 lbs/hr. Full scale 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.

BASES

2.1 FUEL CLADDING INTEGRITY

The abnormal operational transients applicable to operation of the FitzPatrick Unit have been analyzed throughout the spectrum of planned operating conditions up to the thermal power condition of 2535 MWt. The analyses were based upon plant operation in accordance with the operating map given in Figure 3.7-1 of the FSAR. In addition, 2436 is the licensed maximum power level of FitzPatrick, and this represents the maximum steady-state power which shall not knowingly be exceeded.

Fuel cladding integrity is assured by the operating limit MCPR's for steady state conditions given in Specification 3.1.B. These operating limit MCPR's are derived from the established fuel cladding integrity Safety Limit, and an analysis of abnormal operational transients. For any abnormal operating transient analysis evaluation with the initial condition of the reactor being at the steady state operating limit, it is required that the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient.

The most limiting transients have been analyzed to determine which result in the largest reduction in CRITICAL POWER RATIO. The type of transients evaluated were increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest delta MCPR. When added to the Safety Limit, the required operating limit MCPR of Specification 3.1.B is obtained.

The evaluation of a given transient begins with the system initial parameters shown in the current reload analysis and reference 2 that are input to a core dynamic behavior transient computer program described in references 1 and 3. The output of these programs along with the initial MCPR form the input for the further analyses of the thermally limited bundle with a single channel transient thermal hydraulic code. The principal result of the evaluation is the reduction in MCPR caused by the transient.

2.1 BASES (cont'd)

C. References

1. Linford, R. B., "Analytical Methods of Plant Transient Evaluations for the General Electric Boiling Water Reactor", NEDO-10802, Feb., 1973.

2. "General Electric Fuel Application" NEDE 24011-P-A (Approved revision number applicable at time that reload fuel analyses are performed).

3. "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors" NEDO-24154, October, 1978

The reactor coolant pressure boundary integrity is an important barrier in the prevention of uncontrolled release of fission products. It is essential that the integrity of this boundary be protected by establishing a pressure limit to be observed for all operating conditions and whenever there is irradiated fuel in the reactor vessel.

The pressure safety limit of 1,325 psig as measured by the vessel steam space pressure indicator is equivalent to 1,375 psig at the lowest elevation of the Reactor Coolant System. The 1,375 psig value is derived from the design pressures of the reactor pressure vessel and reactor coolant system piping. The respective design pressures are 1250 psig at 575°F for the reactor vessel, 1148 psig at 568°F for the recirculation suction piping and 1274 psig at 575°F for the discharge piping. The pressure safety limit was chosen as the lower of the pressure transients permitted by the applicable design codes: 1965 ASME Boiler and Pressure Vessel Code, Section III for pressure vessel and 1969 ANSI B31.1 Code for the reactor coolant system piping. The ASME Boiler and Pressure Vessel Code permits pressure transients up to 10 percent over design pressure (110% x 1,250 -1,375 psig), and the ANSI Code permits pressure transients up to 20 percent over the design pressure (120% x 1,150 - 1,380 psig). The safety limit pressure of 1,375 psig is referenced to the lowest elevation of the Reactor Coolant System.

The current reload analysis shows that the main steam isolation valve closure transient, with flux scram, is the most severe event sulting directly in a reactor coolant system pressure increase. The reactor vessel pressure code limit of 1,375 psig, given in FSAR Section 4.2, is above the peak pressure produced by the event above. Thus, the pressure safety limit (1,375 psig) is well above the peak pressure that can result from reasonably expected overpressure transients. (See current reload analysis for the curve produced by this analysis.) Reactor pressure is continuously indicated in the control room during operation.

A safety limit is applied to the Residual Heat Removal System (RHRS) when it is operatin in the shutdown cooling mode. When operating in the shutdown cooling mode, the RHRS is included in the reactor coolant system.

The numerical distribution of safety/relief valve setpoints shown in 2.2.1.B (2 @ 1090 psi, 2 @ 1105 psi, 7 @ 1140 psi) is justified by analyses described in the General Electric report NEDO-24129-1, Supplement 1, and assures that the structural acceptance criteria set forth in the Mark I Containment Short Term Program are satisfied.

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 at rated power and flow, the MCPR operating limits shall not be less than those 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.

MCPR Operating Limit for Incremental Cycle Core Average Exposure

Fuel Type	BOC to 1GWD /t before EOC	EOC-1GWD/t to EOC
At RBM trip level setting S = 0.66 W + 39%		
8x8	1.22	1.23
8x8R	1.22	1.23
P8x8R	1.22	1.25
At RBM trip level setting S = 0.66W + 40%		
8x8	1.24	1.24
8x8R	1.24	1.24
P8x8R	1.24	1.25
At RBM trip level setting S = 0.66 W + 41%		
8x8	1.27	1.27
8x8R	1.27	1.27
P8x8R	1.27	1.27
At RBM trip level setting S = 0.66 W + 42%		
8x8	1.31	1.31
8x8R	1.31	1.31
P8x8R	1.31	1.31

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 3.1-2a, 3.1-2b, 3.1-2c

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

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

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

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

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

If anytime during reactor operation greater than 25% of rated power it is determined that the limiting value for MCPR is being exceeded, action shall then be initiated within fifteen (15) minutes to restore operation to within the prescribed limits. If the MCPR is not returned to within the prescribed limits within two (2) hours, an orderly reactor power reduction shall be commenced immediately. The reactor power shall be reduced to less than 25% of rated power within the next four hours, or until the MCPR is returned to within the proscribed limits. For core flows other than rated, the MCPR operating limit shall be multiplied by the appropriate k_f is as shown in figure 3.1.1.

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

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

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

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.

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 analysis assumed reactor operation will be limited to MCPR = 1.20, as described in NEDO-21662-2. 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

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 Sectión 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 pwer (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 for 8x8, 8x8R and P8x8R fuel.

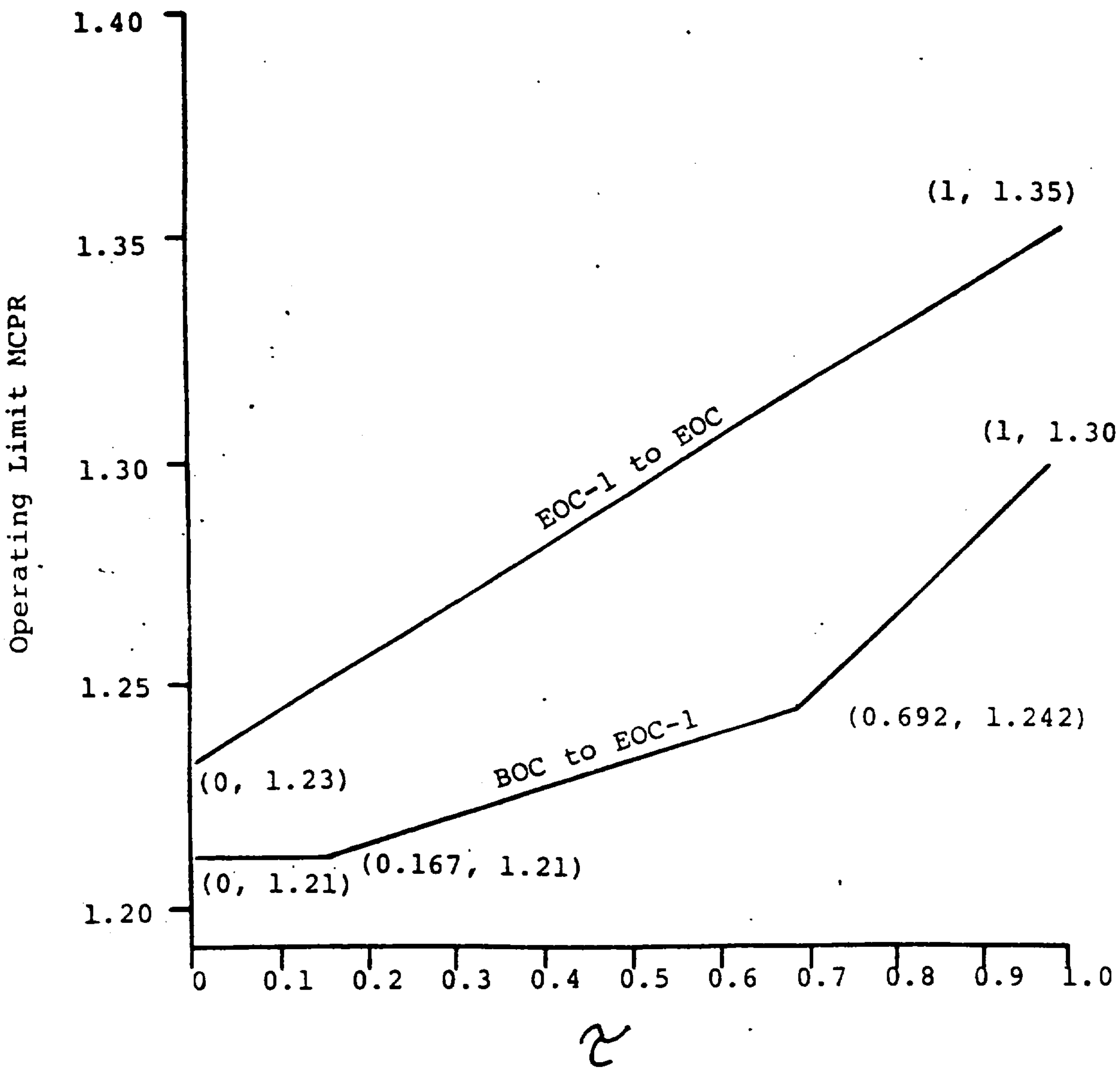
The ratio of FRP to MFLPD shall be set equal to 1.0 unless the actual operating value is less than the design value of 1.0, in which case the actual operating value will be used

W = Loop Recirculation flow in percent of rated (rated is 34.2×10^6 lb/hr)

S = Scram setting in percent of initial

13. The Average Power Range Monitor scram function is varied (Figure 1.1-1) as a function of recirculation loop flow (W). The trip setting of this function must be maintained in accordance with Specification 2.1.A.1.c.

FIGURE 3.1-2a
Operating Limit MCPR Versus τ (Section 4.1.E)
For 8x8 Fuel Types

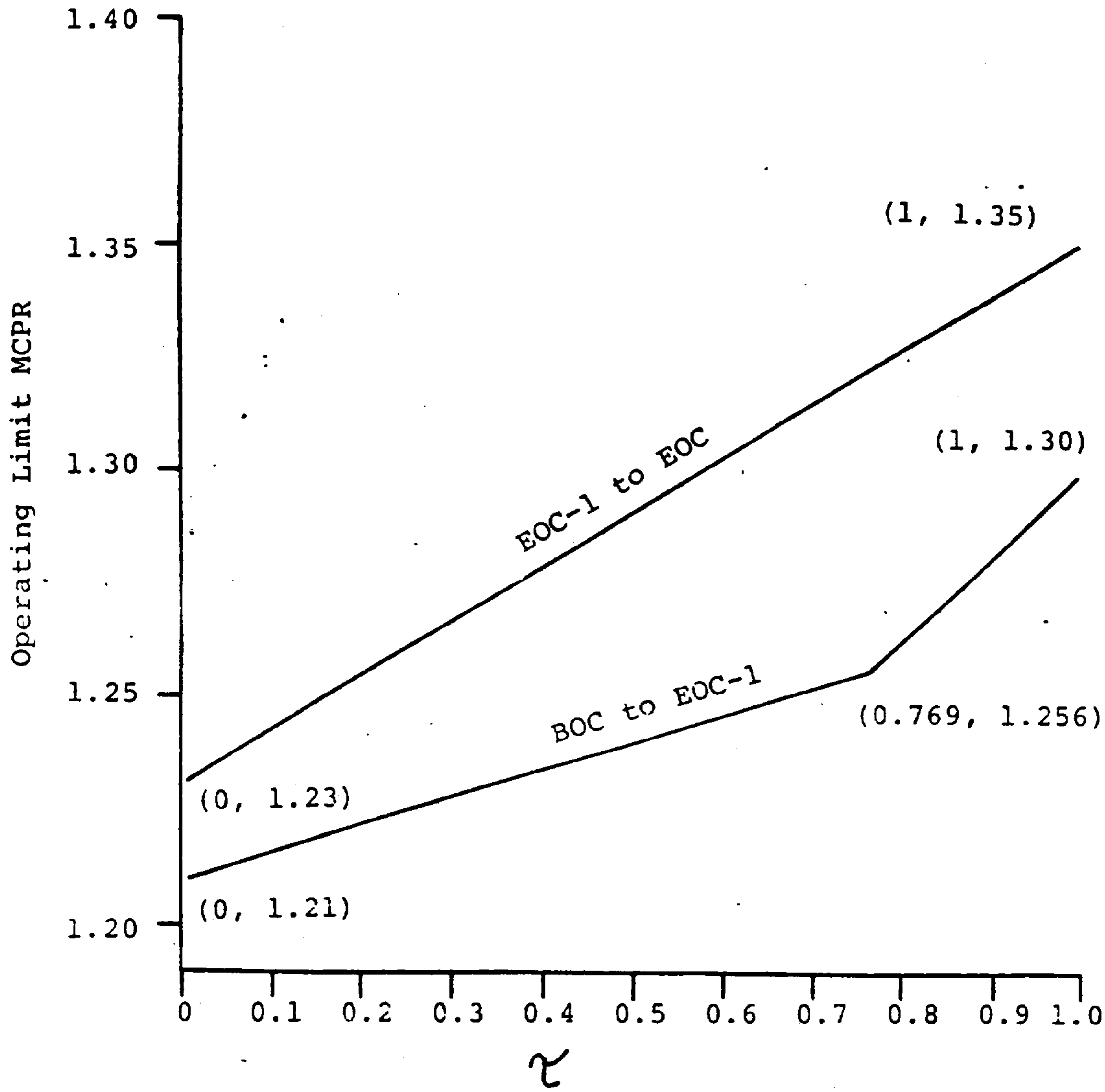


Option B τ =0
Option A τ =1

FIGURE 3.1-2b

Operating Limit MCPR Versus τ (Section 4.1.E)

For 8X8R Fuel Types



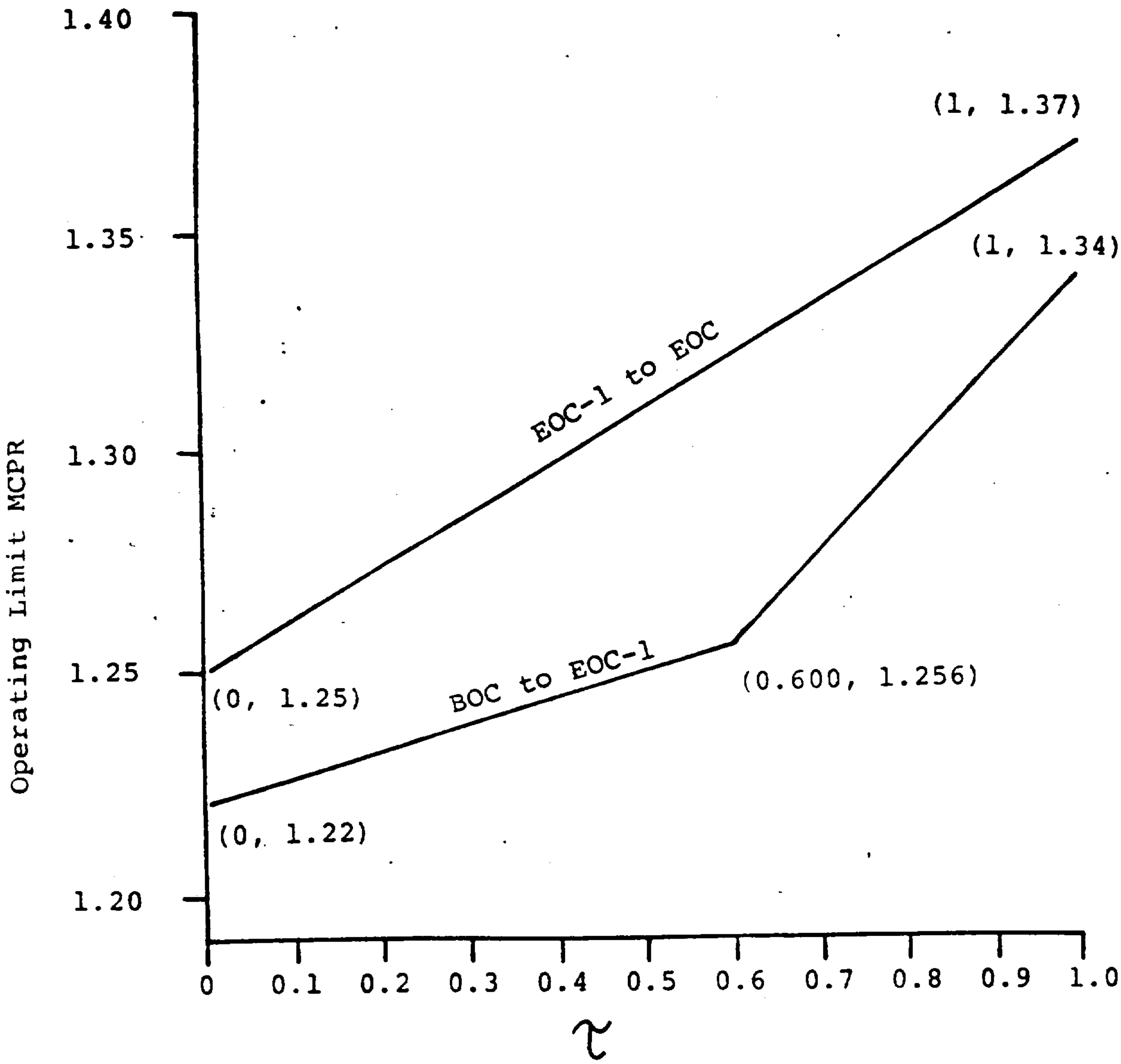
Option B τ =0
Option A τ =1

Ammendment No. 64

FIGURE 3.1-2c

Operating Limit MCPR Versus τ (Section 4.1.E)

For P8X8R Fuel Types



Option B τ =0
Option A τ =1

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 3.5.3 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 for 8x8, 8x8R and P8x8R bundles.

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% thermal power.

Amendment No. ~~49~~, 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 3.5.3 through 3.5-10.

I. Linear Heat Generation Rate (LHGR)

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

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

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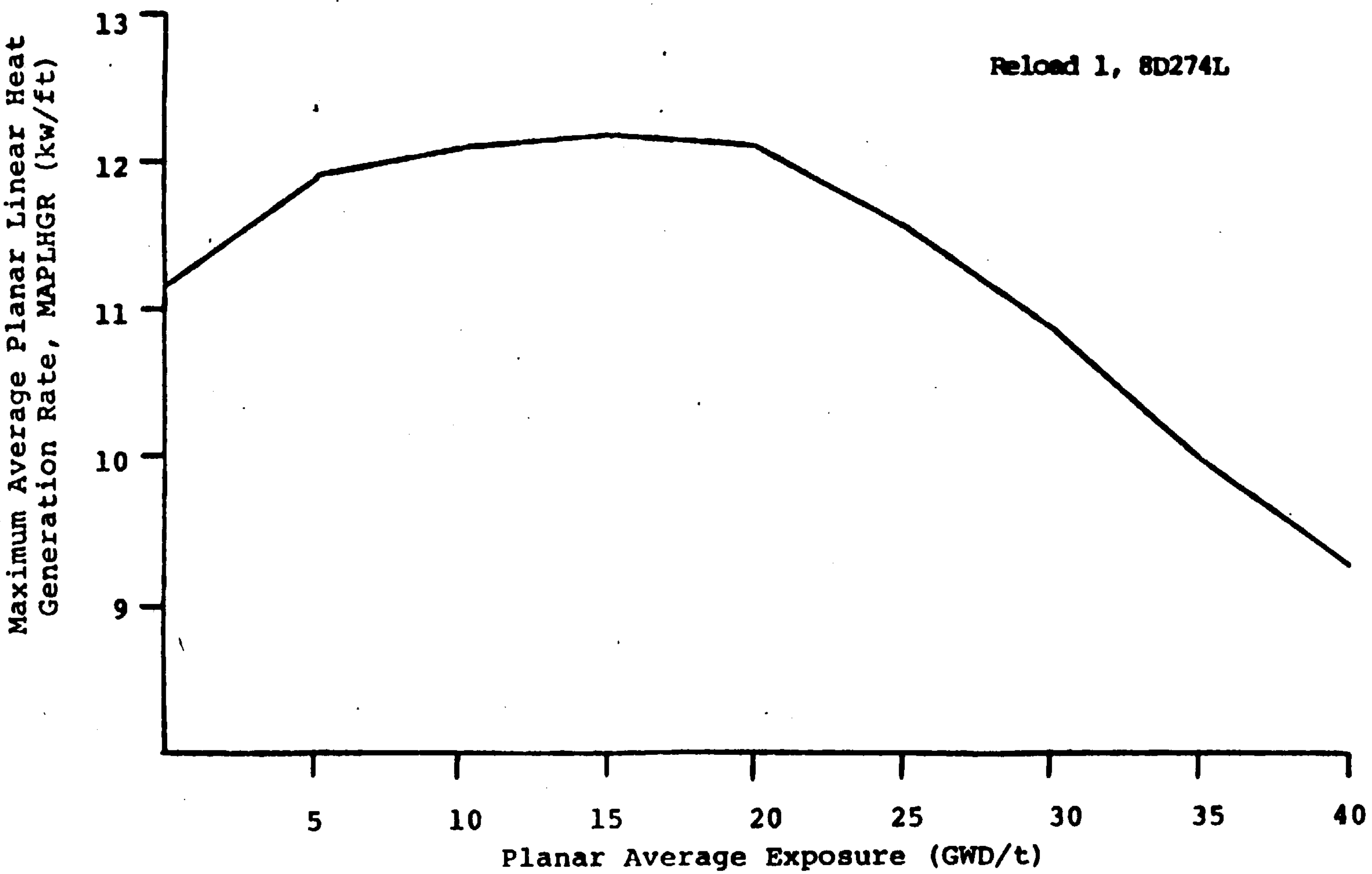
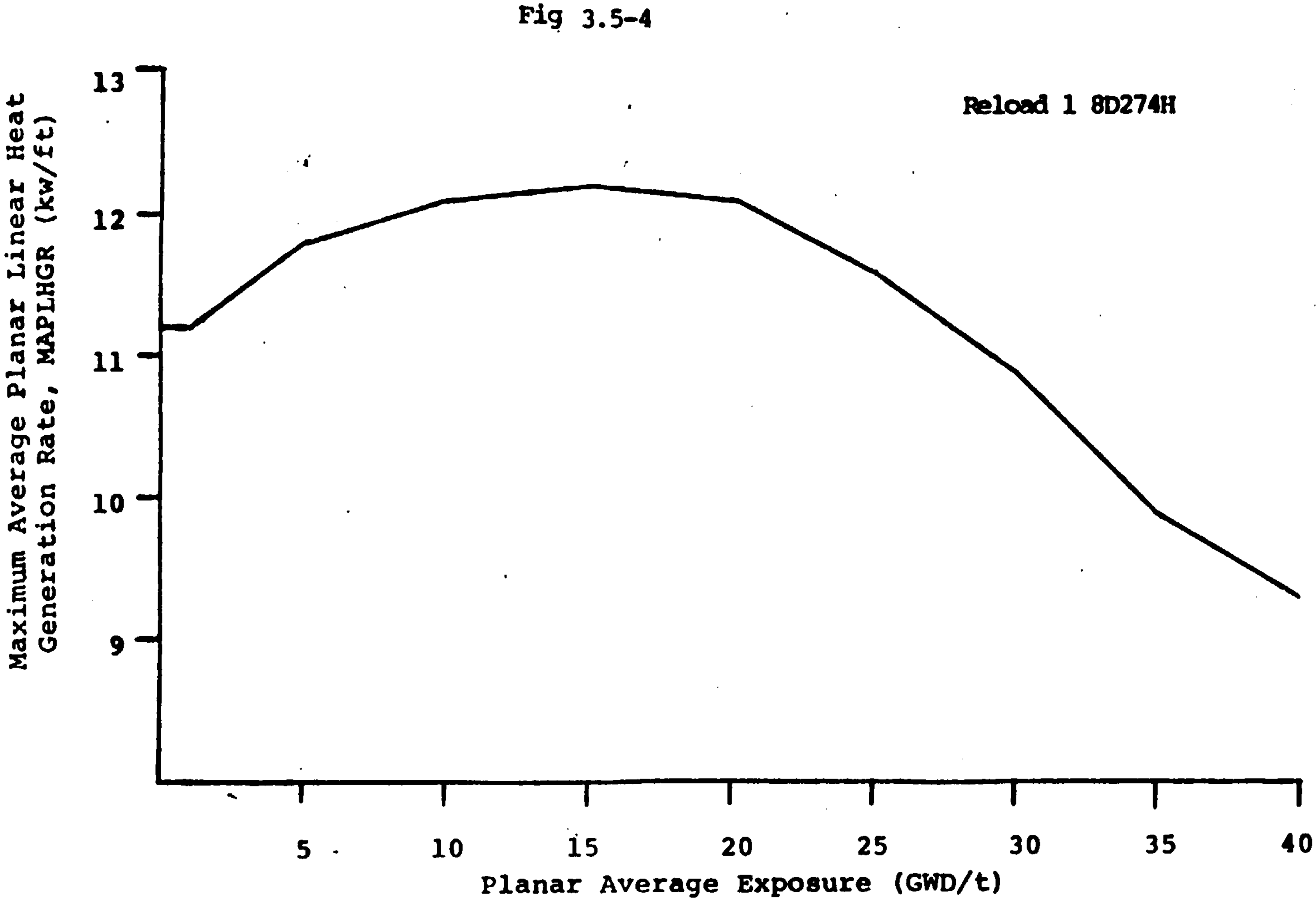
Fig 3.5-3



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

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

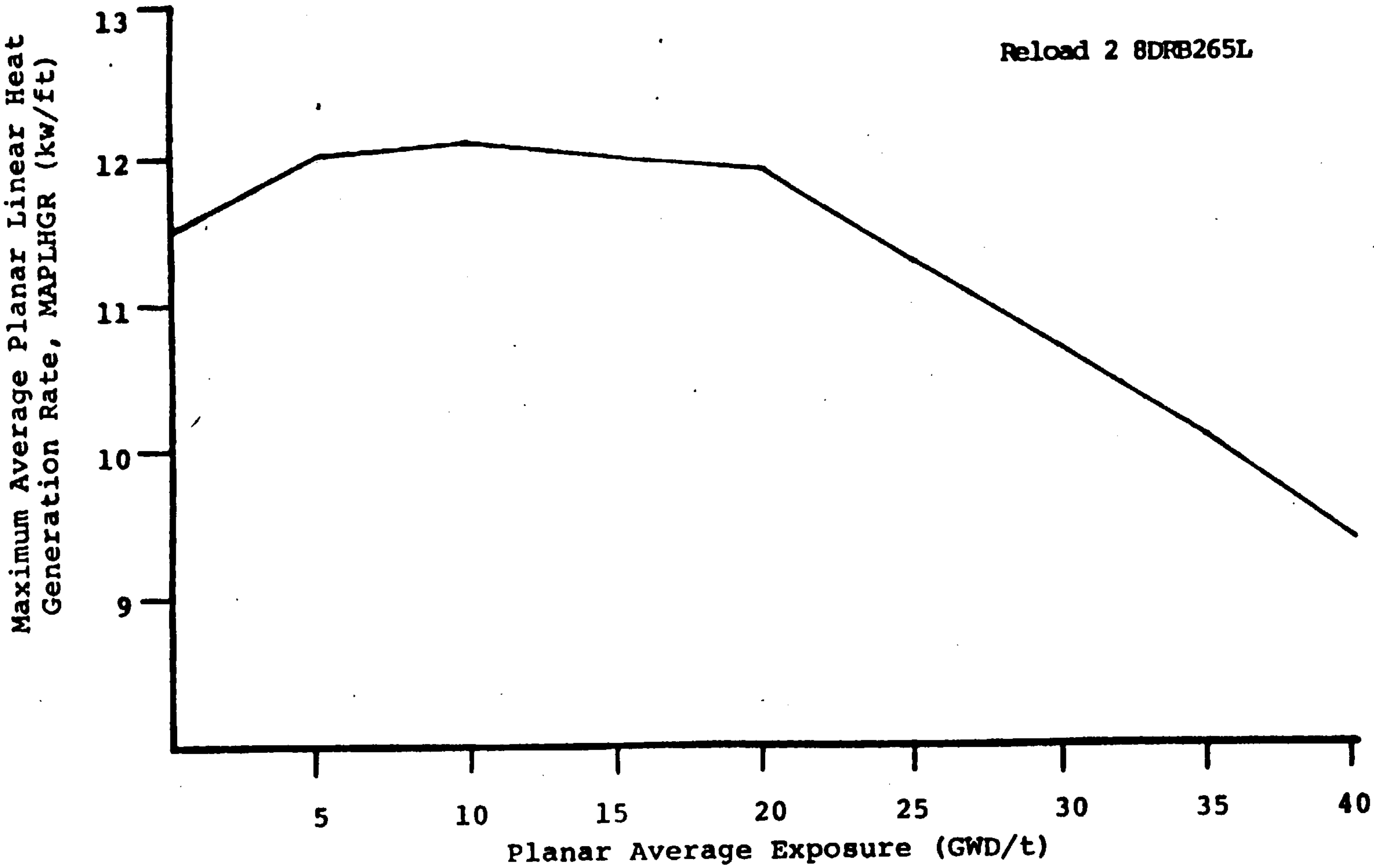
Fig 3.5-4



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

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

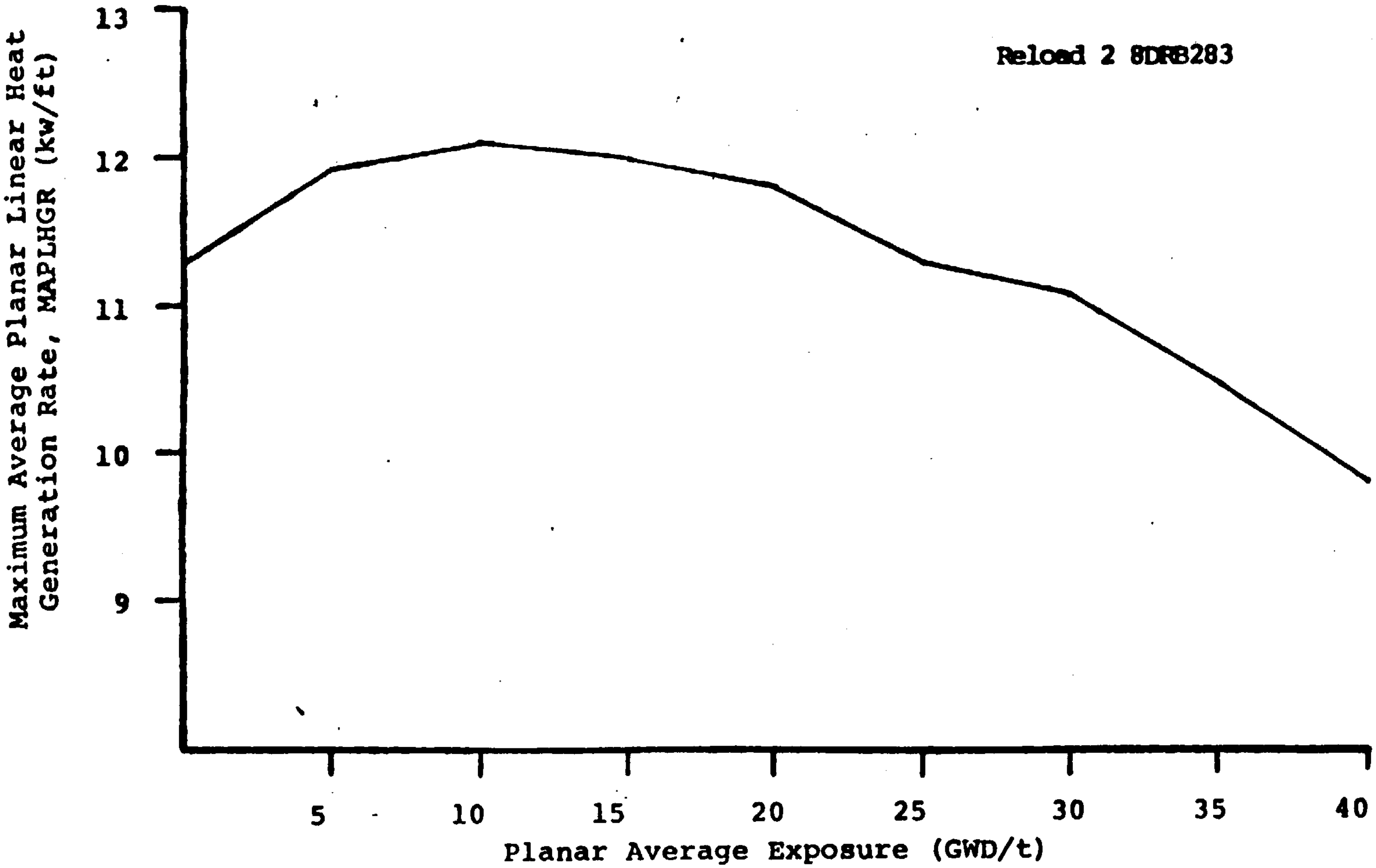
Fig 3.5-5



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

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

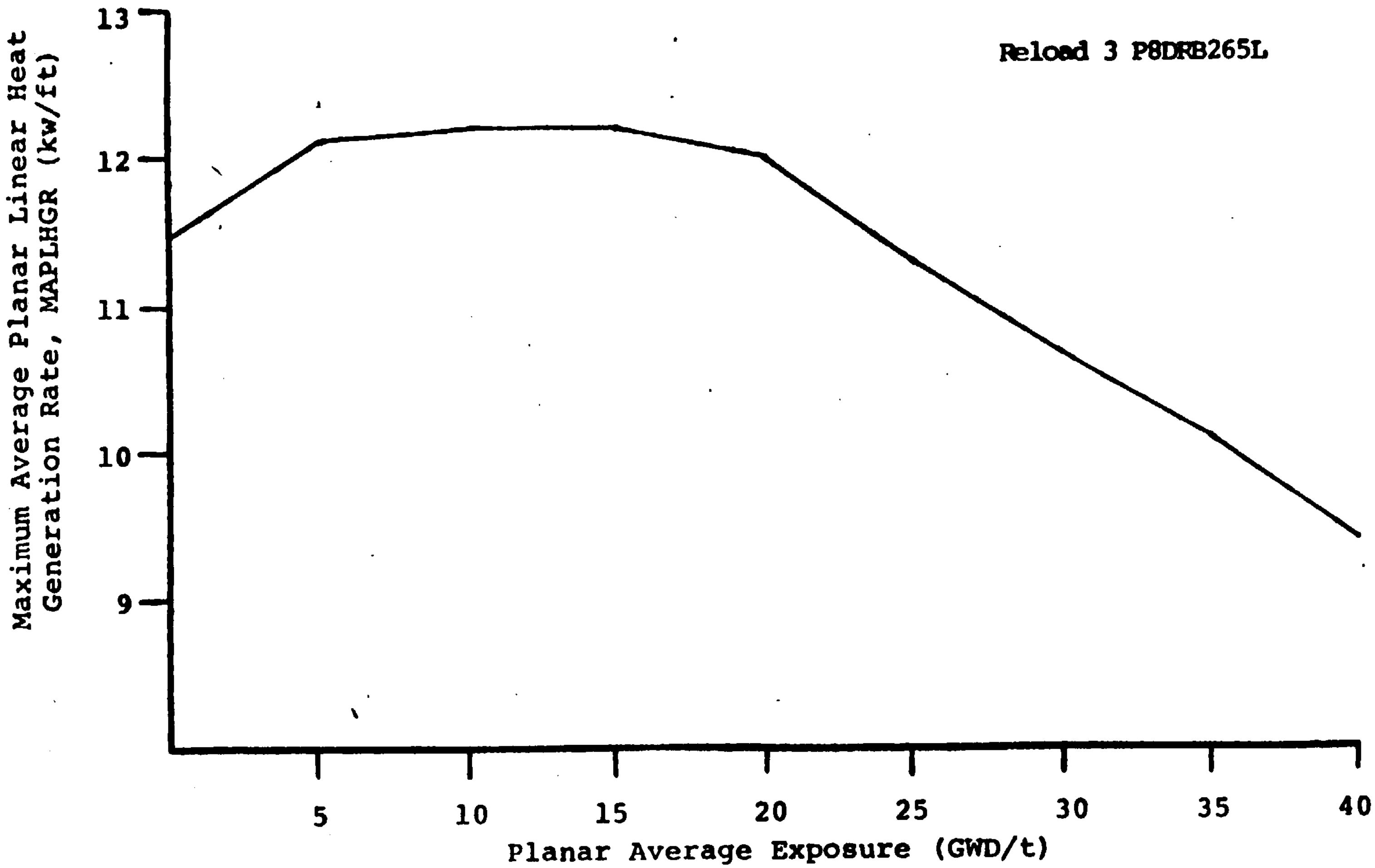
Fig 3.5-6



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

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

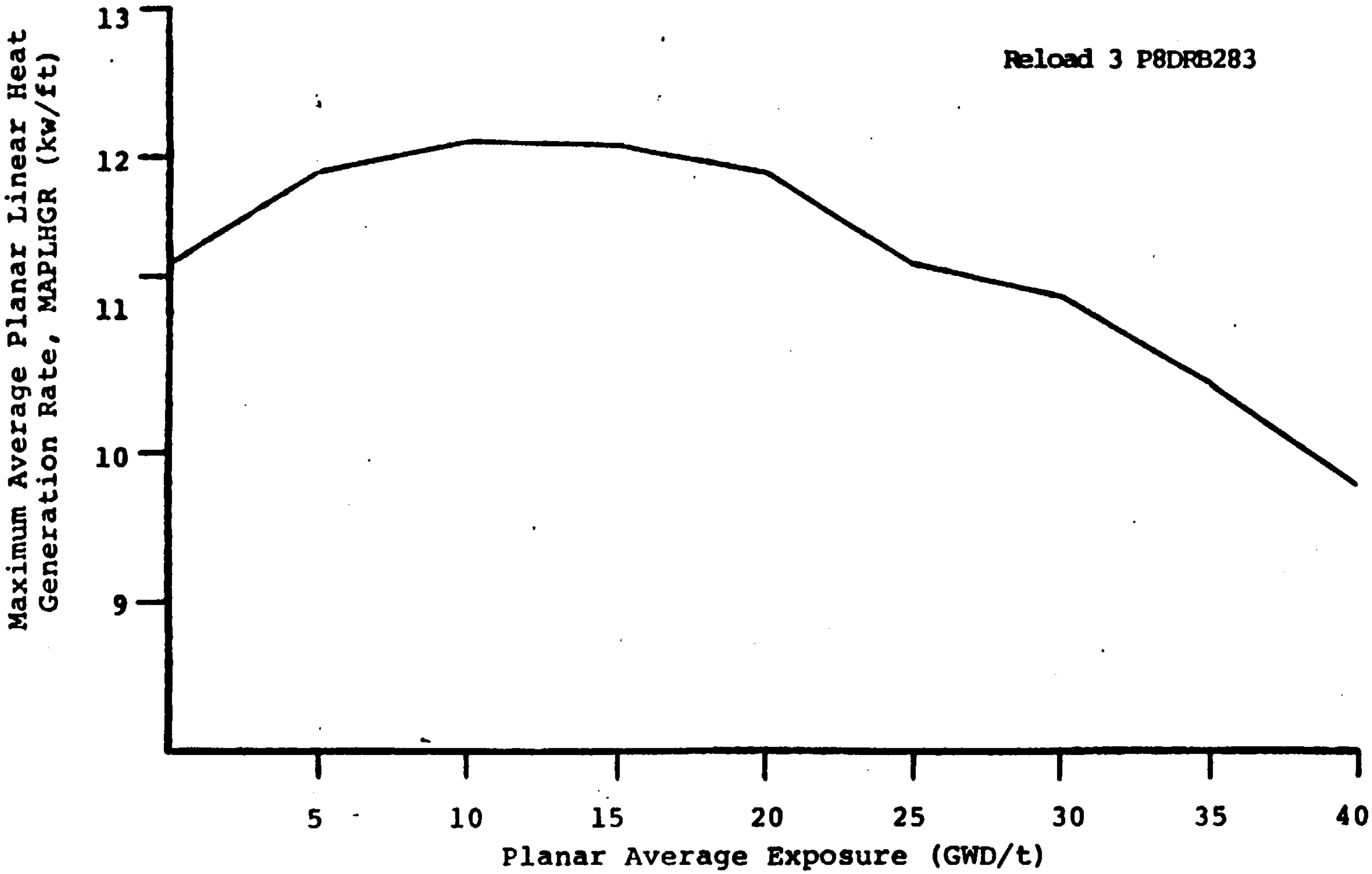
Fig 3.5-7



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

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

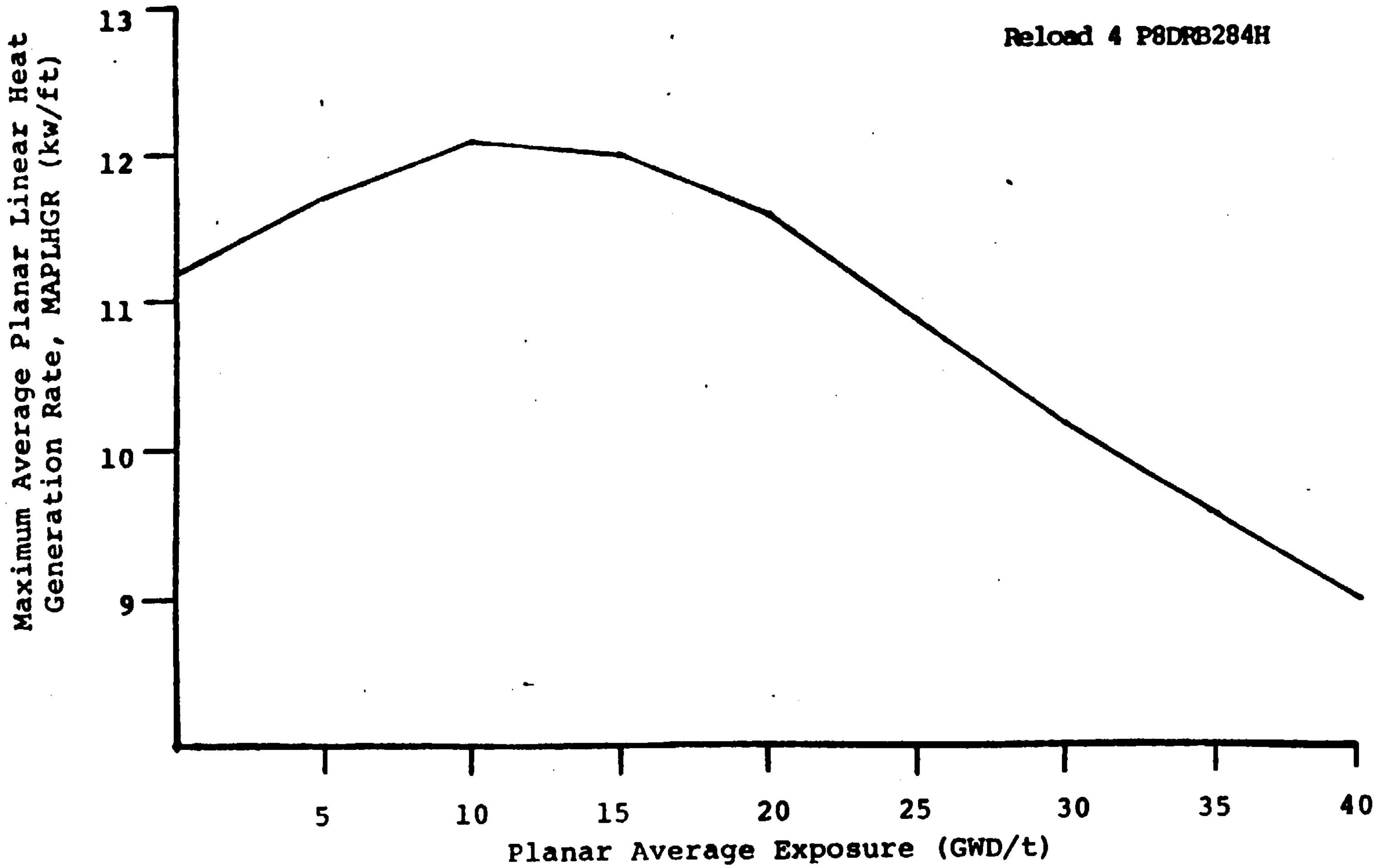
Fig 3.5-8



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

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

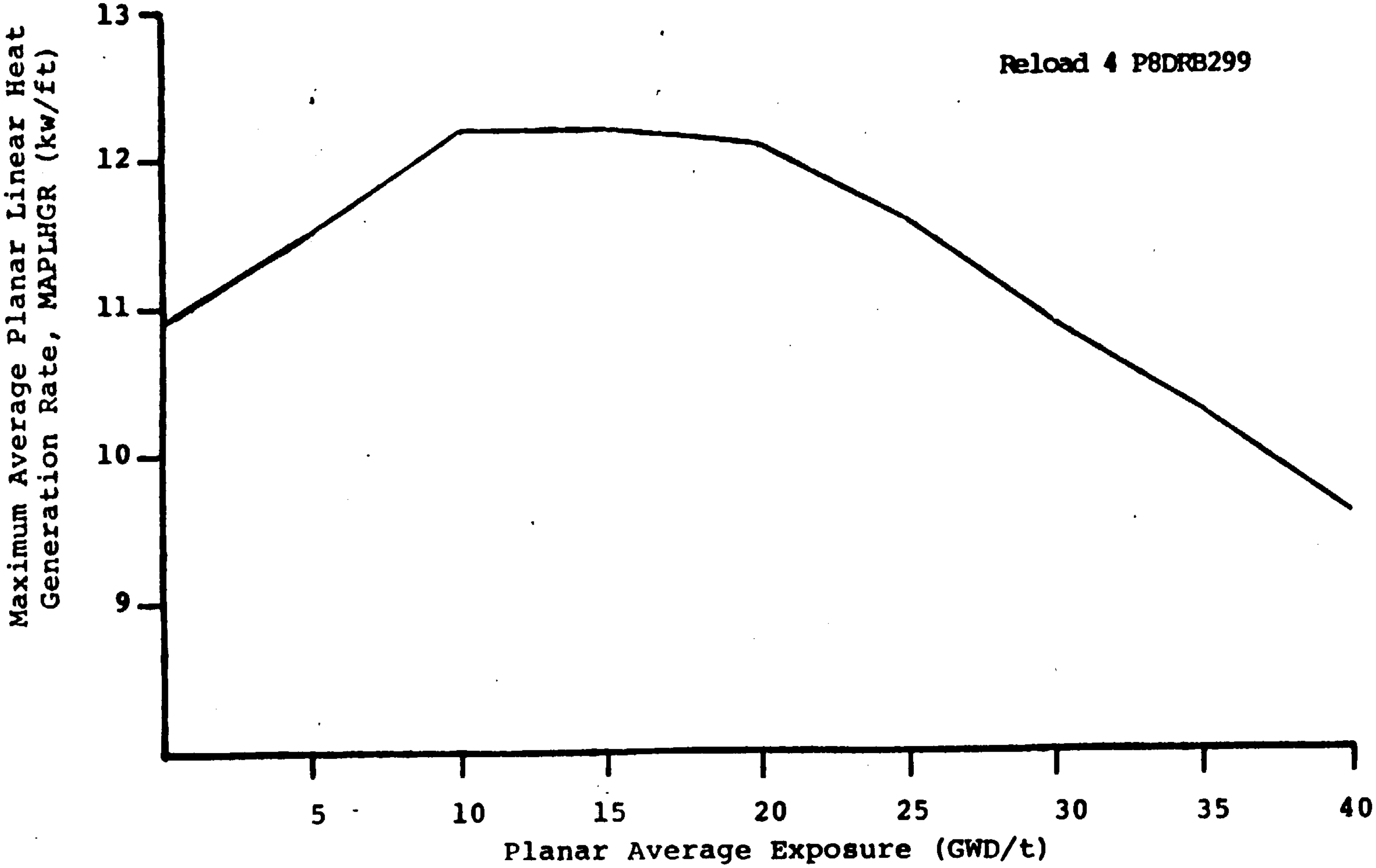
Figure 3.5-9



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

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

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 three fuel types are present in the core: 8x8, 8x8R and P8x8R. These fuel types are described in NEDO-24011. The 8x8 fuel has 63 fuel rods and 1 water rod, and the 8x8R and P8x8R fuel 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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 64 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

The Power Authority of the State of New York (the licensee) has proposed changes to the Technical Specifications of the James A. FitzPatrick Nuclear Power Plant (the facility) in Reference 1. The proposed changes relate to the core for reload 4 Cycle 5 operation at power levels up to 2436 MWt (100%) power. In support of the reload application, the licensee has enclosed proposed Technical Specification changes in Reference 1 and the GE BWR supplemental licensing submittal (Reference 2).

In addition to 8x8 and 8x8R fuel, this reload involves loading of pre-pressurized GE 8x8 retrofit (P8x8R) fuel. This is the same type of fuel as was loaded during the last reload. The description of the nuclear and mechanical designs of 8x8 retrofit fuel are contained in References 3 and 4. Reference 3 also contains a complete set of references to topical reports which describe GE's analytical methods for nuclear, thermal-hydraulic, transient and accident calculations, and information regarding the applicability of these methods to cores such as FitzPatrick which contain a mixture of fuel. The use and safety implications of prepressurized fuel have been found acceptable per Reference 4. The conclusions of Reference 5 found that the methods of Reference 3 were generally applicable to pre-pressurized fuel. Reference 6 found that the conclusions of Reference 5 are applicable for the second and subsequent fuel cycles. Therefore, unless otherwise specified, Reference 3, as supported by References 5 and 6, is adequate justification for the current application of prepressurized fuel.

2.0 Evaluation

2.1 Reactor Physics

The reload application follows the procedure described in NEDE-24011-P, "Generic Reload Fuel Application." We have reviewed this application and the consequent Technical Specification changes. The transient analysis input parameters are typical for BWRs and are acceptable. Core wide transient analysis results are given for the limiting transients and the required operating limit values for MCPR are given for each fuel type. The revised MCPR limits are required by the reload and they are acceptable.

2.2 Thermal Hydraulics

As stated in Reference 3, for BWR cores which reload with GE's retrofit 8x8R fuel, the safety limit minimum critical power ratio (SLMCPR) resulting from either core-wide or localized abnormal operational transients is equal to 1.07. When meeting this SLMCPR during a transient, at least 99.9% of the fuel rods in the core are expected to avoid boiling transition.

To assure that the fuel cladding integrity SLMCPR will not be violated during any abnormal operational transient or fuel misloading, the most limiting events have been reanalyzed for this reload by the licensee in order to determine which event results in the largest reduction in the minimum critical power ratio. These events have been analyzed for the exposed fuel and fresh fuel. Addition of the largest reductions in critical power ratio to the SLMCPR was used to establish the operating limits for each fuel type.

We have found the methods used for this analysis consistent with previously approved past practice (Reference 3). We have found the results of this analysis and the corresponding Technical Specification changes acceptable.

2.3 ECCS Appendix K

Input data and results for ECCS analysis have been given in References 1 and 2. The information presented fulfills the requirements for each analyses outlined in Reference 3.

We have reviewed the analyses and information submitted for the reload and conclude that the facility will be in conformance with all requirements to 10 CFR 50.46 and Apendix K to 10 CFR 50.46 when it is operated in accordance with the Technical Specifications we are issuing with this amendment. Supplemental calculations that address the issues of NUREG-0630 have also been given in Reference 2.

2.4 Maximum Average Planar Linear Heat Generation Rate (MAPLHGR)

In References 7 and 8 GE requested that credit for calculated peak cladding temperature margin as well as credit for recently approved, but unapplied, ECCS evaluation model changes be used to offset any operating penalties due to high burnup fission gas release. This proposal was found acceptable (Ref. 9), provided the generic analysis was found to be applicable to each plant citing the GE position. In Attachment II of Reference 1 the licensee stated that the generic analysis is applicable to the FitzPatrick reload. On this basis we find the proposed Technical Specification changes (MAPLHGR limits) given in Attachment II of Reference 1 acceptable.

3.0 Environmental Considerations

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact, and pursuant to 10 CFR Section 51.5(d)(4) that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of the amendment.

4.0 Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) 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.

Dated: January 29, 1982

Author: Philip J. Polk

References

1. Letter, J. P. Bayne (PASNY) to Office of Nuclear Reactor Regulation (USNRC) dated November 18, 1981.

2. "Supplemental Reload Licensing Submittal for James A. FitzPatrick Nuclear Power Plant, Reload 4," dated August 1981.

3. "General Electric Boiling Water Reactor Generic Reload Application," NEDE-24011-P-A, May 1977.

4. Letter, R. E. Engel (GE) to U.S. Nuclear Regulatory Commission dated January 30, 1979.

5. Letter, T. A. Ippolito (USNRC) to R. Gridley (GE), April 16, 1979 and enclosed SER.

6. Letter, T. A. Ippolito (USNRC), to all operating BWR licensees, May 28, 1981 and enclosed SER.

7. R. E. Engel (GE) letter to T. A. Ippolito (NRC) dated May 6, 1981.

8. R. E. Engel (GE) letter to T. A. Ippolito (NRC) dated May 28, 1981.

9. L. S. Rubenstein (NRC) memorandum for T. M. Novak (NRC) on "Extension of General Electric Emergency Core Cooling Systems Performance Limits" dated June 25, 1981

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-333

POWER AUTHORITY OF THE STATE OF NEW YORK

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U.S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 64 to Operating License No. DPR-59 issued to the Power Authority of the State of New York which revises the Technical Specifications for operation of the James A. FitzPatrick Nuclear Power Plant (the facility) located in Oswego County, New York. The amendment is effective as of the date of its issuance.

The amendment modifies the Technical Specifications to reflect the following changes: (1) revised maximum average planar linear heat generation rate (MAPLHGR) data, (2) power spiking penalty, and (3) use of the ODYN computer program.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of the amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of the amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of the amendment.

For further details with respect to this action, see (1) the application for amendment dated November 18, 1981 (2) Amendment No. 64 to License No. DPR-59, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C., and at the Penfield Library, State University College at Oswego, Oswego, New York 13126. A copy of items (2) and (3) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 29th day of January 1982.

FOR THE NUCLEAR REGULATORY COMMISSION

DBVassallo

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