ATTACHMENT I

PROPOSED TECHNICAL SPECIFICATION

CHANGE

RELATED TO

RELOAD 4/CYCLE 5

POWER AUTHORITY OF THE STATE OF NEW YORK
JAMES A. FITZPATRICK NUCLEAR POWER PLANT
DOCKET NO. 50-333
DPR-59

LIST OF FIGURES

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

U. Thermal Parameters

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

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

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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:

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

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.

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 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 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 operating in the shutdown cooling mode. When operati in the shutdown cooling mode, the RHRS is included in the r ctor 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 3.1-2 *

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

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

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

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

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

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

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

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

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

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

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

N_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 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 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.

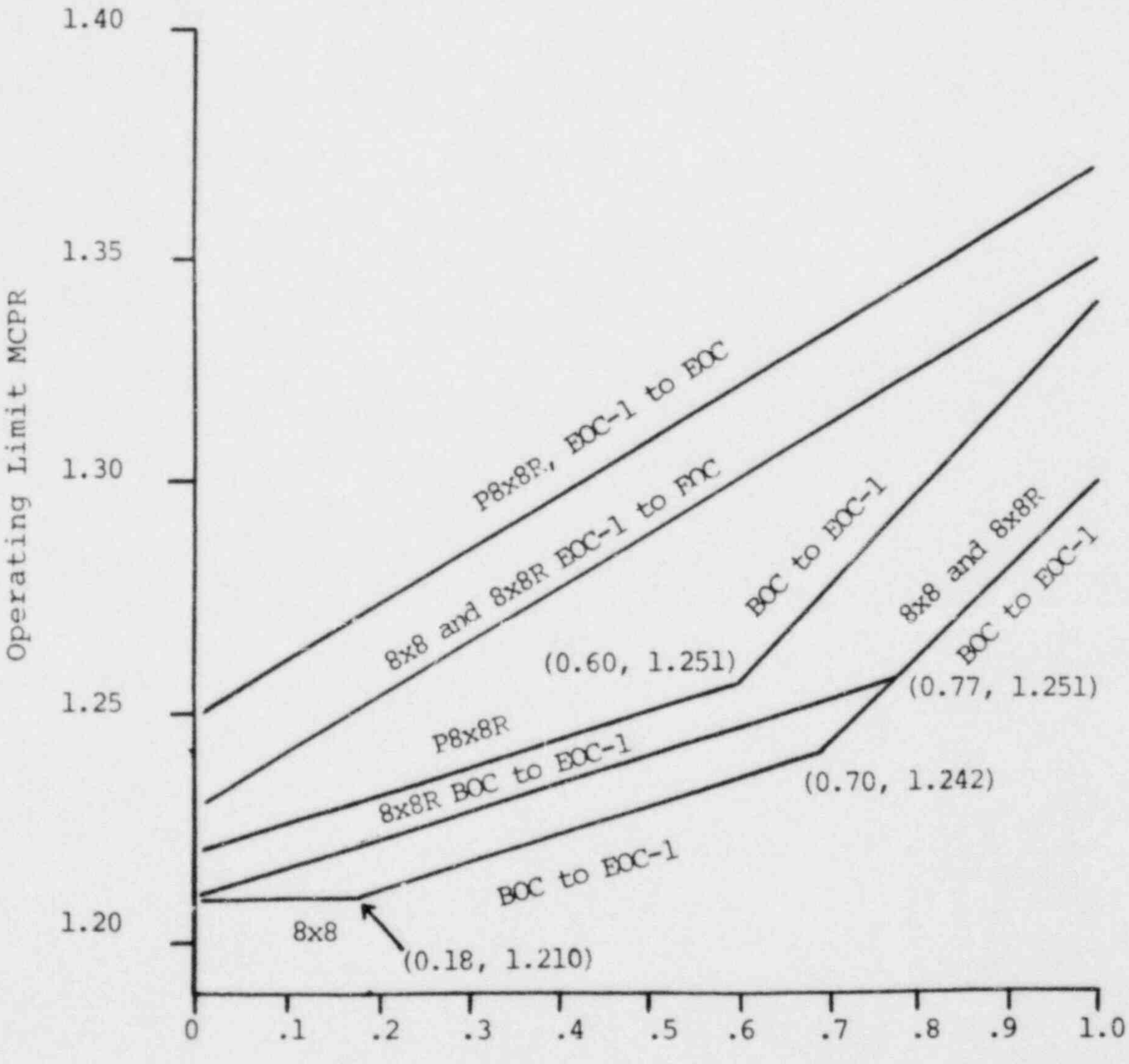
The ratio of FRP to MFLPD shall be set equal to 1.0 unless the actual operating value is less then the design value of 1.0, in which case the actual operating value will be used

W = Loop Recirculation flow in percent of rated (rated is 34.2 x 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 3.1-2 Operating Limiting MCPR vs. τ (sec. 4.1.E.)



option B = 0
option A = 1

Amendment No.

3.3 (cont'd)

2. The average of the scram insertion times for the three fastest operable control rods of all groups of four control rods in a two-by-two array shall be no greater than:

Control Rod Notch Position Observed	Average Scram Insertion Time (Sec)
46	0.361
38	0.977
24	2.112
04	3.764

4.3 (cont'd)

2. At 16-week intervals, 10 percent of the operable control rod drives shall be scram timed above 950 psig. Whenever such scram time measurements are made, an evaluation shall be made to provide reasonable assurance that proper control rod drive performance is being maintained.

3.3 and 4.3 BASES (cont'd)

rods have been withdrawn (e.g., groups A_{12} and A_{34}, it is demonstrated that the Group Notch made for the control drives is enforced. This demonstration is made by performing the hardware functional test sequence. The Group Notch restraints are automatically removed above 20% power.

During reactor shutdown, similar surveillance checks shall be made with regard to rod group availability as soon as automatic initiation of the RSCB occurs and subsequently at appropriate stages of the control rod insertion.

4. The Source Range Monitor (SRM) System performs no automatic safety system function; i.e., it has no scram function. It does provide the operator with a visual indication of neutron level. The consequences of reactivity accidents are functions of the initial neutron flux. The requirement of at least 3 counts per sec. assures that any transient, should it occur, begins at or above the initial value of 10^{-8} of rated power used in the analyses of transient cold conditions. One operable SRM channel would be adequate to monitor the approach to criticality using homogeneous patterns of squattered control rod withdrawal. A minimum of two operable SRM's are provided as an added conservatism.

5. The Rod Block Monitor (RBM) is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power level operation. Two channels are provided, and one of these may be bypassed from the console for maintenance and/or testing. Tripping of one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage. This system backs up the operator who withdraws control rods according to written sequences. The specified restrictions with one channel out of service conservatively assure that fuel damage will not occur due to rod withdrawal errors when this condition exists.

A limiting control rod pattern is a pattern which results in the core being on a thermal hydraulic limit (i.e., MCPR limits as shown in specification 3.1.B). During use of such patterns, it is judged that testing of the RBM System prior to withdrawal of such rods to assure its operability will assure that improper withdraw does not occur. It is the responsibility of the Reactor Analyst to identify these limiting patterns and the designated rods either when the patterns are initially established or as they develop due to the occurrence of inoperable control rods in other than limiting patterns. Other qualified personnel may perform this function.

C. Scram Insertion Times

The Control Rod System is designated to bring the reactor subcritical at a rate fast enough to prevent fuel damage; i.e., to prevent the MCPR from becoming less than the Safety Limit. Scram insertion time test criteria of Section 3.3.C.1 were used to generate the generic scram reactivity curve shown in NEDE-24011-P-A. This generic curve was used in analysis of non-pressurization transients to determine MCPR limits. Therefore, the required protection is provided.

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 $\geq$ 25% thermal power.

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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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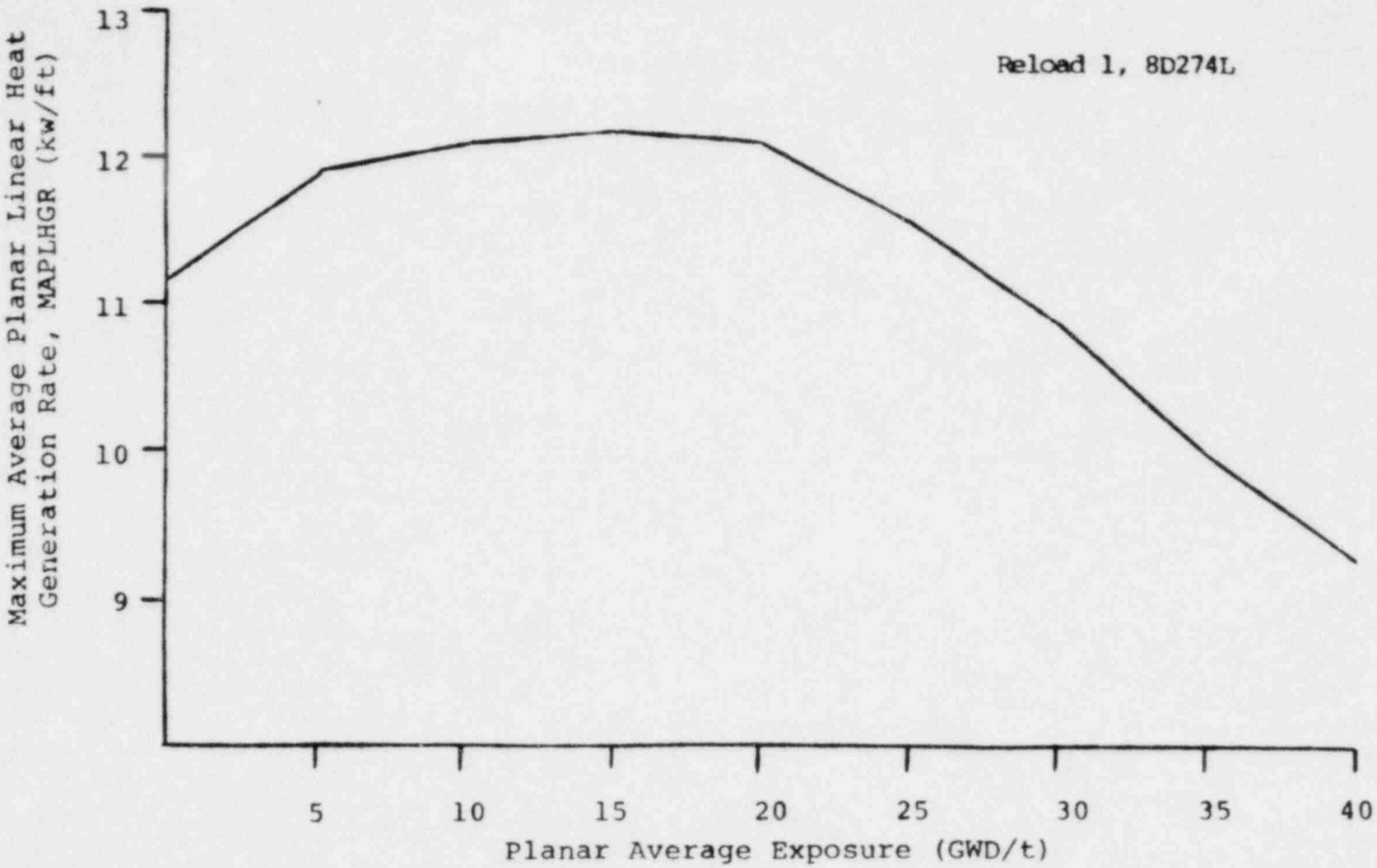
Fig 3.5-3

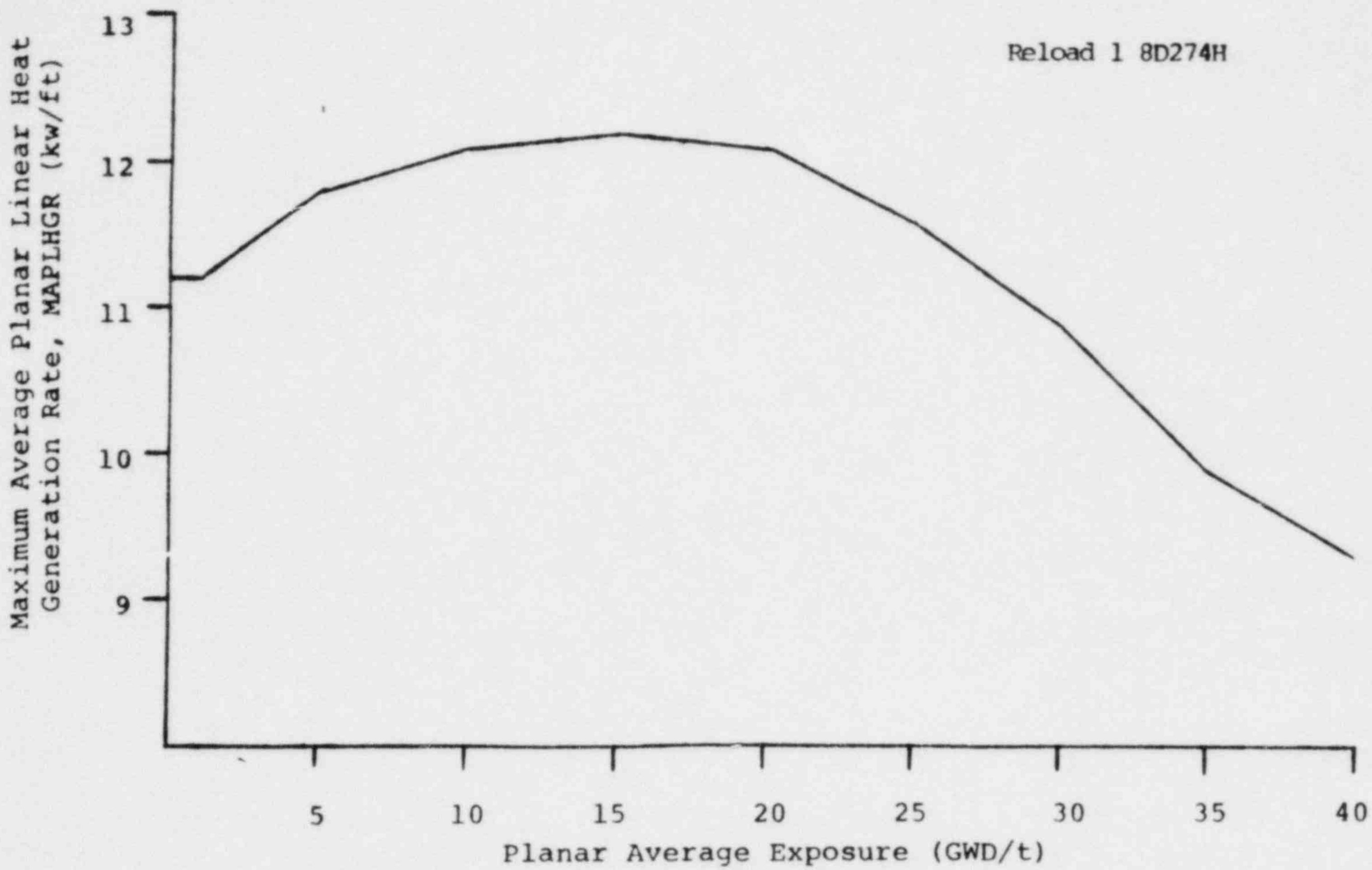


<u>Maximum Average Planar Linear Heat Generation Rate (MAPLHGR)</u>
<u>Versus Planar Average Exposure</u>

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

Amendment No. ~~30~~

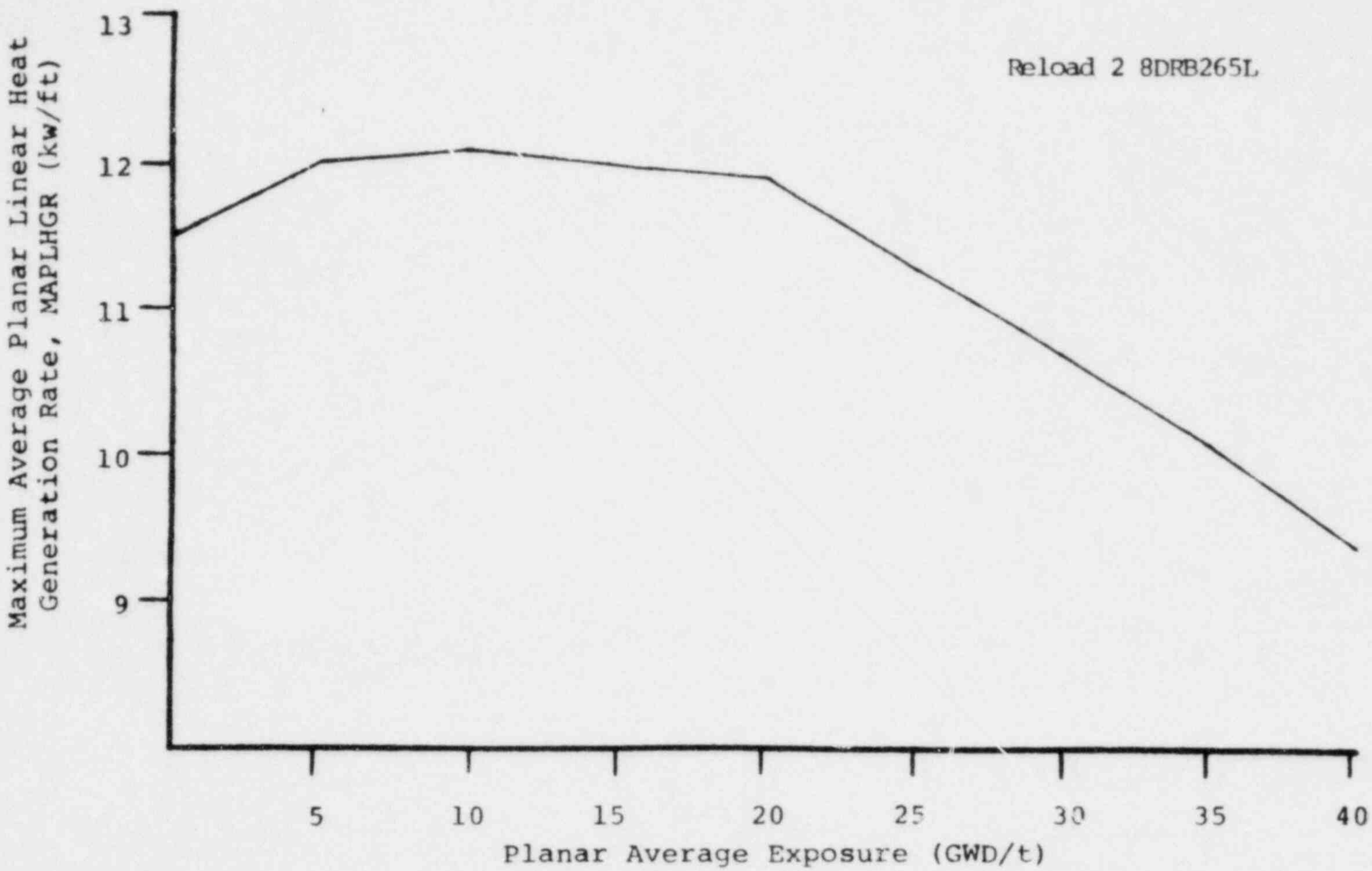
Fig 3.5-4



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

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

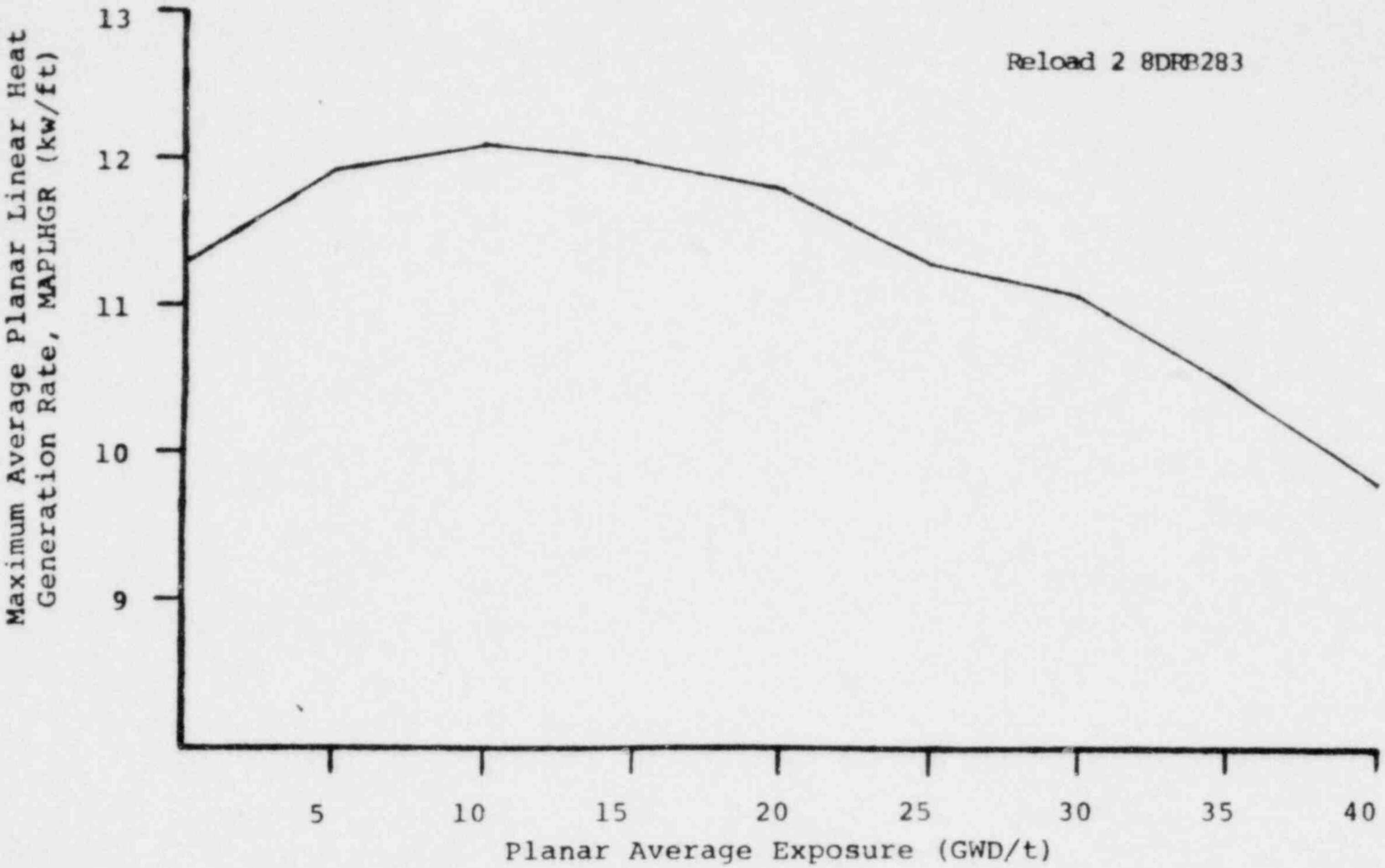
Fig 3.5-5



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

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

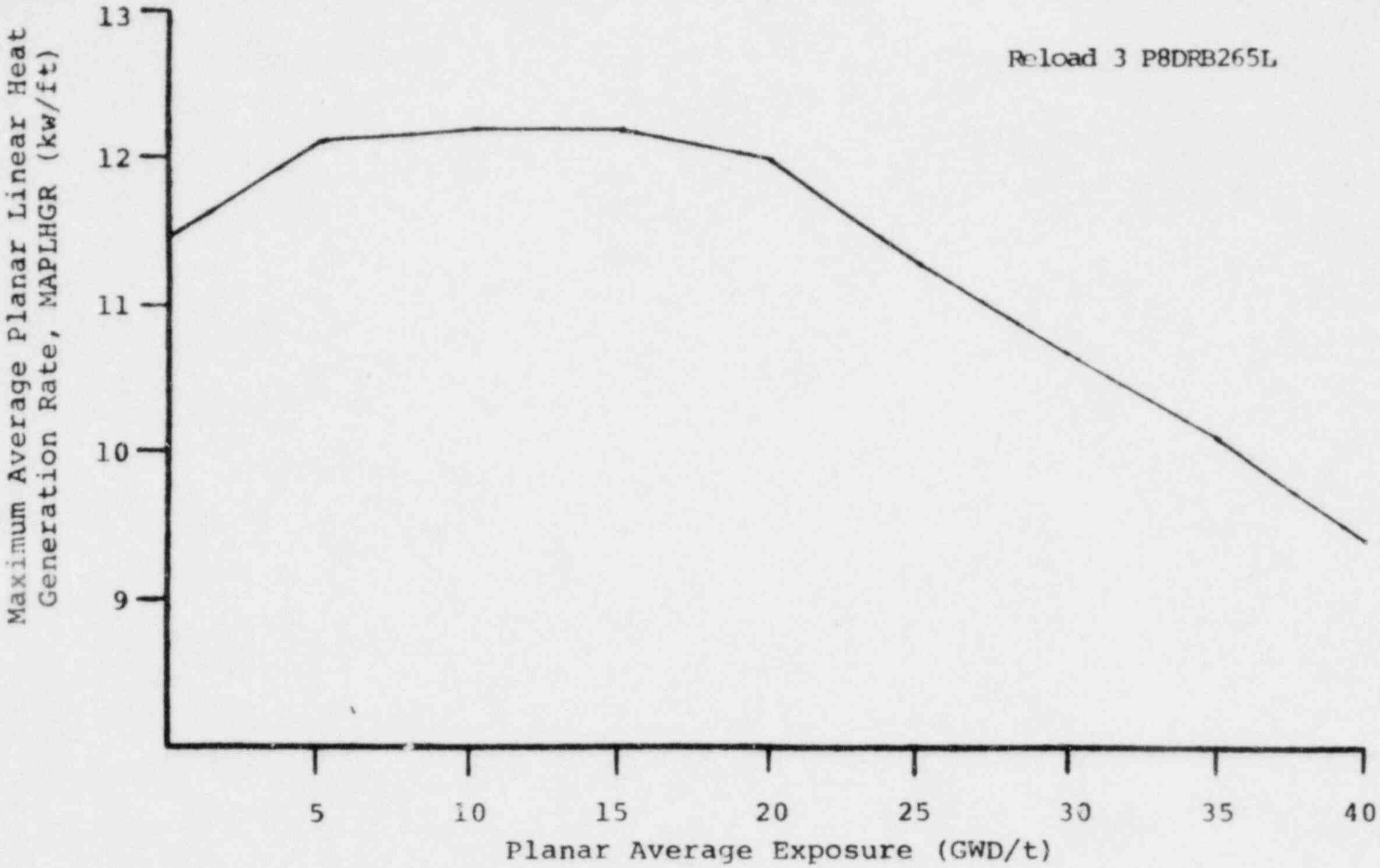
Fig 3.5-6



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

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

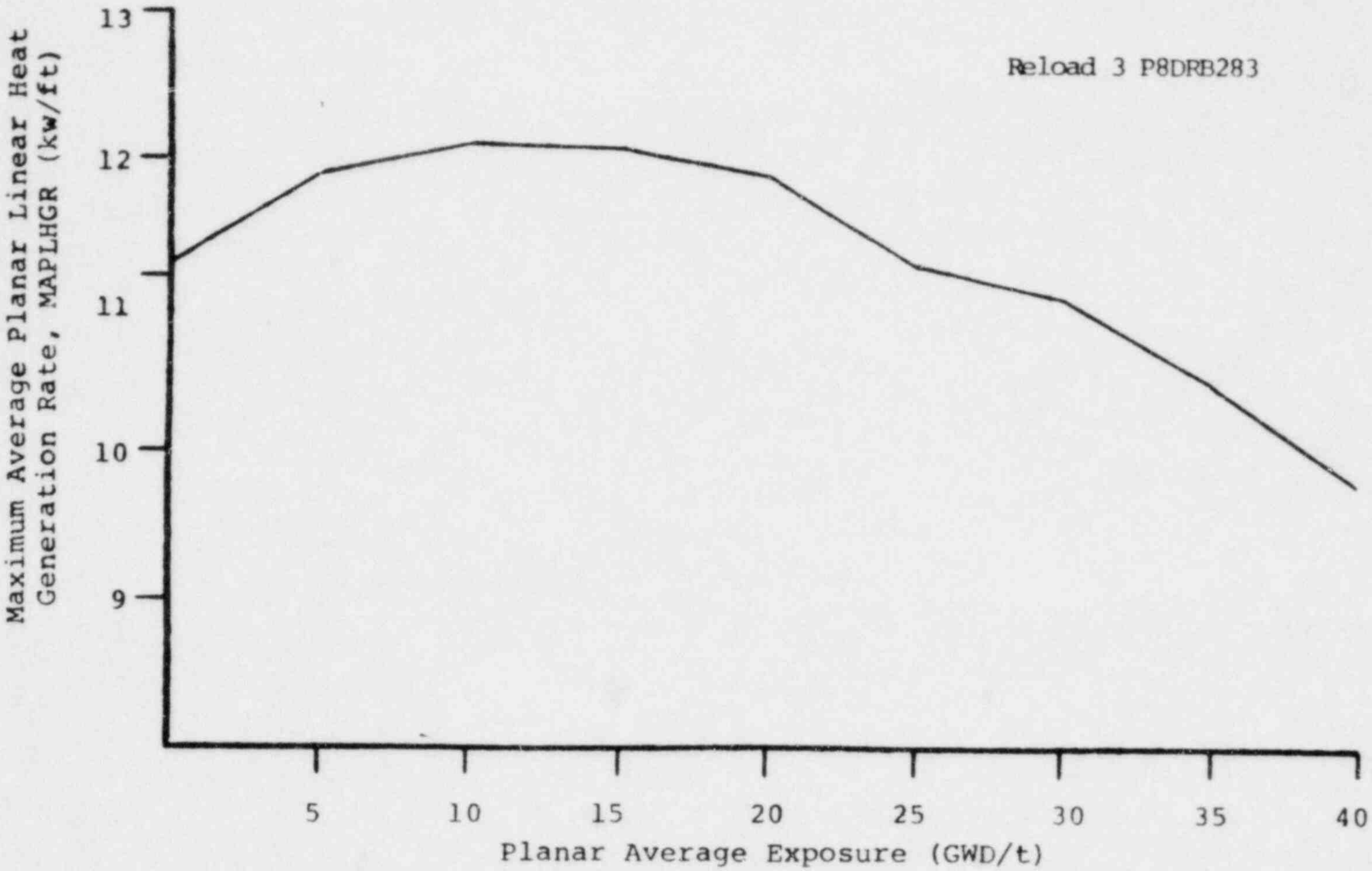
Fig 3.5-7



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

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

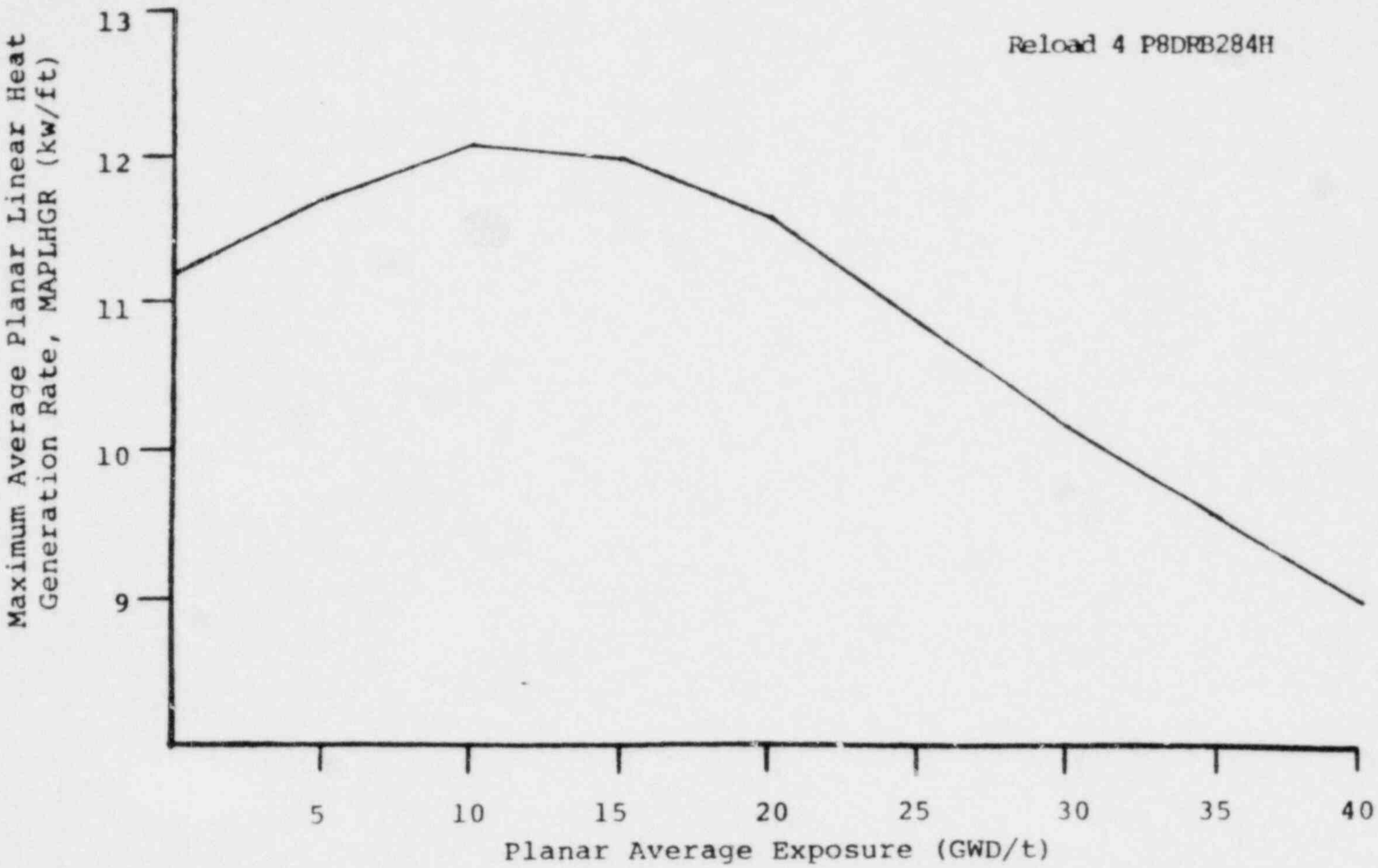
Fig 3.5-8



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

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

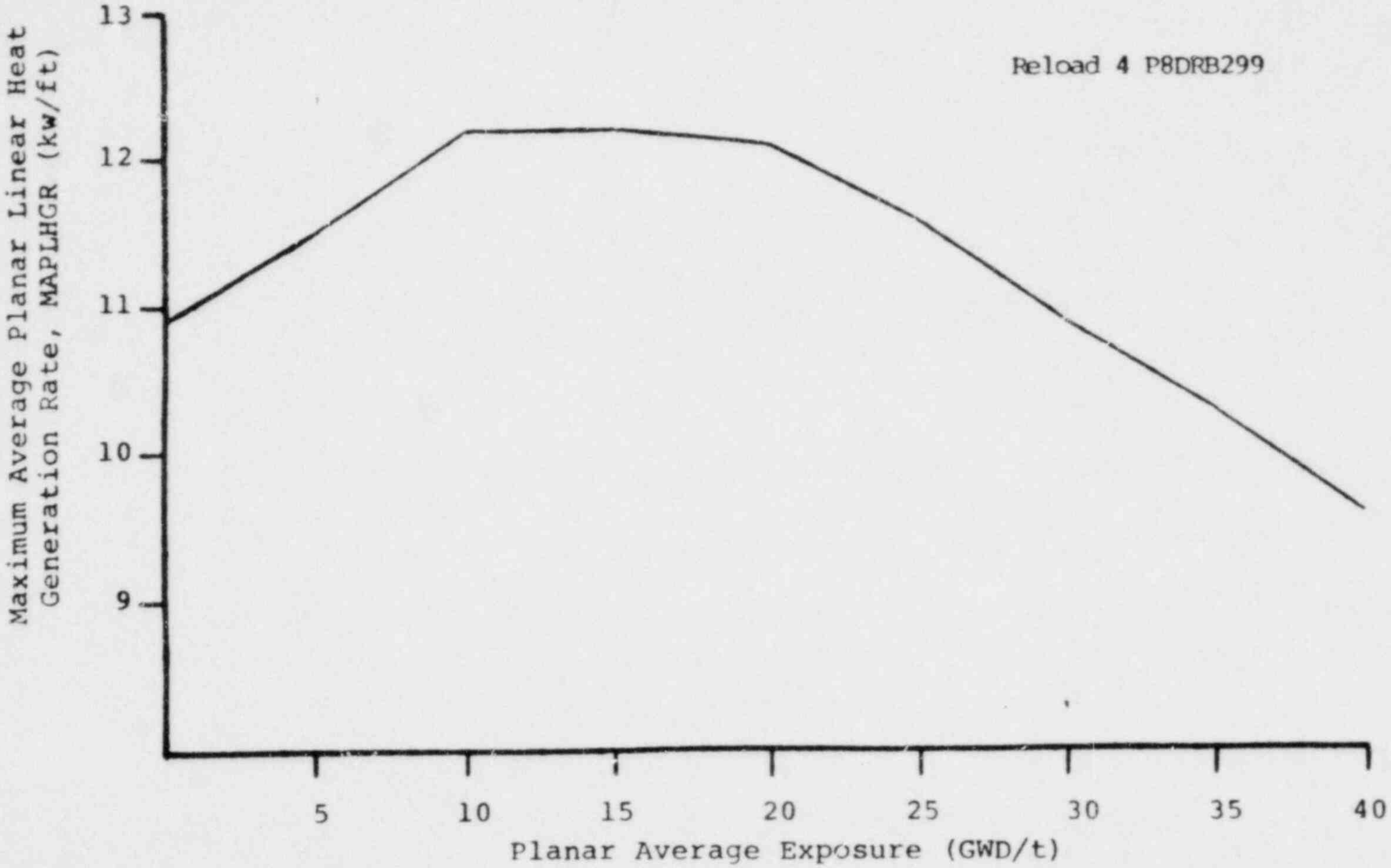
Figure 3.5-9



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)

3.7 (cont'd)

9. Primary Containment Atmosphere Monitoring Instruments

   a. Primary containment atmosphere shall be continuously monitored for hydrogen and oxygen when the containment integrity is required.

B. Standby Gas Treatment System

1. Except as specified in 3.7.B.2 below, both circuits of the Standby Gas Treatment System shall be operable at all times when secondary containment integrity is required.

4.7 (cont'd)

9. Primary Containment Atmosphere Monitoring Instruments

   a. Instrumentation shall be functionally tested and calibrated as specified in Table 4.7-1.

B. Standby Gas Treatment System

1. Standby Gas Treatment System surveillance shall be performed as indicated below:

   a. At least once per operating cycle, it shall be demonstrated that:

      (1.) Pressure drop across the combined high-efficiency and charcoal filters is less than 5.7 in. of water at 6,000 scfm and

      (2.) Each 39 KW heater shall dissipate greater than 29KW of electric power as calculated by the following expression:

$$P = \sqrt{3}\,EI$$

Where: P = Dissipated Electrical Power; E = Measured line-to-line voltage in volts (RMS); I = Average measured phase current in amperes (RMS)

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, 368.945 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.

ATTACHMENT II

SAFETY EVALUATION

RELATED TO

RELOAD 4/CYCLE 5

POWER AUTHORITY OF THE STATE OF NEW YORK
JAMES A. FITZPATRICK NUCLEAR POWER PLANT
DOCKET NO. 50-333
DPR-59

Section I - Description of Change

A. Maxiumum Average Planar Linear Heat Generation Rate

Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) data for two new fuel types is provided in Figures 3.5-9 and 3.5-10. Additionally, the MAPLHGR data for all fuel types remaining in the JAF core (Figures 3.5-3 thru 3.5-8) has been extended from a maximum nodal exposure of 30 GWD/short ton to 40 GWD/short ton. This new MAPLHGR data was developed by General Electric as a revision to the General Electric Report NEDO-21662-2 (Attachment III).

The methods used to generate the new MAPLHGR limits are the same as those used in the original LOCA Analysis which was presented in NEDO-21662-2. The effect of enhanced fission gas release at high exposures on these extended MAPLHGR limits was discussed in a letter dated May 6, 1981 from R. E. Engel of General Electric to T. A. Ippolito. In this letter, General Electric demonstrates that adequate margin to the 2200°F peak clad temperature (PCT) limit exists when General Electric's evaluation of the fission gas correction factor is applied.

B. Power Spiking

The requirement to check linear heat generation rate (LHGR) as a function of core height has been elminated from Section 3.5.I (p.124) and 4.5.I (p.124), and from the Bases 3.5.I (p.130). References related to fuel densification (p.131) have been deleted since they are not referred in the body of Appendix A.

The requirement to check LHGR as a function of core height originated with the axially varying power spiking penalty on LHGR which was instituted to account for the effect of fuel densification. In Amendment No. 49 to the JAF Technical Specifications, the power spiking penalty for 8 x 8 and 8 x 8R fuels was eliminated in accordance with the NRC safety evaluation issued June 9, 1978 to General Electric [Reference (f)]. Since the Cycle 5 core will consist only of 8 x 8 fuel types, this specification no longer applies.

C. Control Rod Drive Scram Surveillance Frequency

The proposed modification would change the control rod drive scram surveillance frequency from the present 15% every eight weeks, to what it was prior to the issuance of Amendment No. 30, namely 10% every sixteen weeks.

D. ODYN Pressurization Transient Analysis

Section 3.1.B containing Operating Limit MCPRs has been rewritten; Specification 4.1.E concerning control rod and scram time test surveillance has been added and the Bases, (Sections 2.1 and 4.3.C) have been ammended as a result of the use of the ODYN computer program (in lieu of REDY) to calculate the plant response to pressurization transients.

The amended MCPR values found in Section 3.1.B are contained in the General Electric report, "Supplemental Reload Licensing Submittal for James A. FitzPatrick Nuclear Power Plant - Reload 4," Attachment IV.

In a letter [Reference (g)] dated November 4, 1981, the Commission informed licensees that the analysis of pressurization transients in reload cores would require the use of the ODYN computer program. A safety evaluation

issued under a letter [Reference (h)] dated Janaury 29, 1981 detailed implementation procedures of the ODYN code. The changes made to Section 3.1.B are in accordance with the generic statistical adjustment factor approach (generally known as ODYN option "B") given in the supplementary safety evaluation attached to the January 29th letter. This generic statistical implementation method requires a new surveillance requirement on control rod scram time of the type contained in the new Section 4.1.E. In this test, periodic control rod scram time test data is checked against the statistical scram time test data used in the ODYN calculation; if the surveillance test data shows that the scram time exceeds that used in the analysis an adjustment is made to the operating limit MCPR as described in the revised Section 3.1.B.

The revisions to the Bases in Sections 2.1 and 4.3.C were made to account for the revised ODYN transient methodology. The previous descriptions were particular to REDY methodology and these revised sections contain a more general description applicable to the methodology used rather than the particular computer program used.

E. Test of Standby Gas Treatment Heaters

Section 4.7.B.1.a.2 (p. 181) is revised to require that the standby gas treatment heaters dissipate a minimum of 29 KW of electrical power. This power level will insure a maximum heater outlet relative humidity of 70 percent at 6000 scfm. Therefore, the existing and proposed surveillance requirements are equivalent forms of the same physical condition.

Section II - Purpose of Change

A. Maximum Average Planar Heat Generation Rate

The MAPLHGR curves are being extended in anticipation of exceeding the present 30 GWD/short ton limit in Cycle 5. Based on current predictions of Cycle 5 energy content, the maximum cumulative nodal exposure at end of Cycle 5 will be approximately 36 GWD/short ton.

B. Power Spiking

The purpose of this change is to eliminate a specification which no longer applies.

C. Control Rod Drive Scram Surveillance Testing

The increased surveillance now in effect was initiated by Amendment No. 30 to the Technical Specifications issued on September 16, 1977 and was intended to protect the drive mechanisms from possible accumulation of corrosion product particulates from the carbon steel system piping. However, a General Electric Company report entitled, "Evaluation of the Effect of Corrosion Particles on Control Rod Drive Operation at the James A. FitzPatrick Nuclear Power Plant" [Reference (c)] indicates that the presence of corrosion particles does not affect the reliability of the scram function of the control rod drive system, including operation without the use of CRD exhaust header filter. The requested modification will improve CRD system reliability by removing the additional duty on the drive imposed by the present Technical Specification

surveillance requirement, and will increase the capacity factor, since such testing causes reduction in reactor power.

D. ODYN Pressurization Transient Analysis

These changes are required to reflect the use of the ODYN computer program for pressurization transient analysis. The new surveillance requirement on control rod scram times allows the use of ODYN option "B" MCPR limits which are less restrictive than the deterministic option "A" limits.

E. Test of Standby Gas Treatment System Heaters

The Authority originally noted in OR 78-16 (and LER 78-10) that the Standby Gas Treatment System heater controls did not operate properly due to a failed transformer. The NRC reported, in Inspection Report No. 78-08, that the Authority intended to either replace the failed components or modify the heater control circuits in such a manner that the heater will operate any time the Standby Gas Treatment System operates. This is acceptable since the humidity controls and heater are intended to maintain humidity at the charcoal filters below 70% relative humidity. Full time operation of the heater will maintain relative humidity below 70%.

In support of the proposed modification and proposed Technical Specification amendment, calculations have been performed which demonstrate that dissipation of 29 KW or more will maintain the relative humidity at the charcoal filter inlet below 70%. This modification will further restore the system to proper operation condition.

Section III - Impact of Change

A. Maximum Average Planar Heat Generation Rate

The extended MAPLHGR limits are an amendment to General Electric report NEDO-21662-2, "Loss-of-Coolant Accident Analysis Report for James A. FitzPatrick Nuclear Power Plant (Lead Plant)" which was submitted to the Commission in a letter from George T. Berry to Robert W. Reid, Chief of Operating Reactor Branch, on July 29, 1977 [Reference (d)]. This report forms the basis of the ECCS analysis for the FitzPatrick plant and SER issued by the Commission on September 16, 1977. Thus the proposed modifications will not alter the conclusions reached in the FSAR accident analysis and SER since they are based on the same plant response calculated in NEDO-21662-2 report and have been calculated using the same approach as the previous MAPLHGR.

B. Power Spiking

Since these specifications no longer apply to the fuel types present in the cycle 5 core, the elimination of this specification is necessary to avoid confusion over its applicatility.

C. Control Rod Drive Scram Surveillance Frequency

The General Electric report [Reference (c)] provides technical justification for the proposed change to the Technical Specifications. Since the presence of corrosion particles does not affect the CRD scram function reliability, and since a decrease in the duty requirement upon the drives will improve their reliability, the requested change will improve the reliability of the system.

D. ODYN Pressurization Transient Analysis

Since the methodology employed is consistent with that accepted in the referenced safety evaluation [Reference (h)], there is no degradation in the margin to the safety limit MCPR.

E. Standby Gas Trreatment Heaters Surveillance Requirements

The incorporation of this alternate method for ensuring the relative humidity at the Standby Gas Treatment charcoal filters, and the completion of the associated modifications will close NRC Office of Inspection and Enforcement, open item No. 81-14-02.

Section IV - Implementation of the Modification

The modifications as proposed will not impact the ALARA or Fire Protection Program at James A. FitzPatrick Nuclear Power Plant.

Section V - Conclusion

The incorporation of these modifications: a) will not increase the probability nor the consequences of an accident or malfunction of equipment important to safety as previously evaluated in the Safety Analysis Report; b) will not increase the possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report; and c) will not reduce the margin of safety as defined in the basis for any Technical Specification, and d) does not constitute an unreviewed safety question.

Section VI - References

(a) JAF FSAR

(b) JAF SER

(c) General Electric Company Report, "Evaluation of the Effect of Corrosion Particles on Control Rod Drive Operation at the James A. FitzPatrick Nuclear Power Plant."

(d) General Electric Company Report, "Loss-of-Coolant Accident Analysis Report for James A. FitzPatrick Nuclear Power Plant (Lead Plant)," NEDO-21662-2, July 1977.

(e) Letter R. E. Engel to T. A. Ippolito, Extension of Emergency Core Cooling System Performance Limits, May 6, 1981.

(f) Letter D. G. Eisenhut to R. L. Gridley, June 9, 1978 (contains Commission SER on Power Spiking and Densification in 8 x 8 Fuels).

(g) Letter D. G. Eisenhut to BWR Licensees, November 4, 1980 (ODYN Implementation).

(h) Letter D. G. Eisenhut to BWR Licensee, January 29, 1981 (ODYN SER and Supplementary SER).