ATTACHMENT I

PROPOSED TECHNICAL SPECIFICATION CHANGES

RELATED TO

SINGLE RECIRCULATION LOOP OPERATION

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

1.1 FUEL CLADDING INTEGRITY

Applicability:

The Safety Limits established to preserve the fuel cladding integrity apply to those variables which monitor the fuel thermal behavior.

Objective:

The objective of the Safety Limits is to establish limits below which the integrity of the fuel cladding is preserved.

Specificaitons

A. Reactor Pressure > 785 psig and Core Flow > 10% of Rated

The existence of a minimum critical power ratio (MCPR) less than 1.07 during two recirculation loop operation shall constitute violation of the fuel cladding integrity safety limit, hereafter called the Safety Limit. An MCPR Limit of 1.08 shall apply during single-loop operation.

2.1 FUEL CLADDING INTEGRITY

Applicability:

The Limiting Safety System Settings apply to trip settings of the instruments and devices which are provided to prevent the fuel cladding integrity Safety Limits from being exceeded.

Objective:

The objective of the Limiting Safety System Settings is to define the level of the process variables at which automatic protective action is initiated to prevent the fuel cladding integrity Safety Limits from being exceeded.

Specificaitons

A. Trip Settings

The limiting safety system trip settings shall be as specified below:

1. Neutron Flux Trip Settings

A. IRM - The IRM flux scram setting shall be set at ≤ 120/125 of full scale.

1.1 (cont'd)

B. Core Thermal Power Limit (Reactor Pressure $\leq$ 785 psig)

When the reactor pressure is $\leq$ 785 psig or core flow is less than 10% of rated, the core thermal power shall not exceed 25 percent of rated thermal power.

C. Power Transient

To ensure that the Safety Limit established in Specification 1.1.A. and 1.1.B is not exceeded, each required scram shall be initiated by its expected scram signal. The Safety Limit shall be assumed to be exceeded when scram is accomplished by a means other than the expected scram signal

2.1 (cont'd)

b. APRM Flux Scram Trip Setting (Refuel or Start & Hot Standby Mode)

APRM - The APRM flux scram setting shall be $\leq$ 15 percent of rated neutron flux, with the Reactor Mode Switch in Startup/Hot Standby or Refuel.

c. APRM Flux Scram Trip Settings (Run Mode)

(1) Flow Referenced Neutron Flux Scram Trip Setting

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

$S \leq (0.66W + 54\%)$ for two-loop operation or:
$S \leq (0.66W + 54\% - 0.66\Delta W)$ for single-loop operation.

where:

S = Setting in percent of rated thermal power (2436 MWT)

W = Loop recirculation flow rate in percent of rated (rated loop recirculation flow rate equals 34.2 x 10^6 lb/hr)

ΔW = Difference between two-loop and single-loop effective drive flow at the same core flow.

= 0 for two recirculation loop operation.

= for one recirculation loop operation. (to be determined upon implementation of single loop operation)

For no combination of loop recirculation flow rate and core thermal power shall the APRM flux scram trip setting be allowed to exceed 117% of rated thermal power.

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 (MFLPD) greater than the fraction of rated power (FRP), the setting shall be modified as follows:

$$S \leq (0.66W + 54\%)\left[\frac{FRP}{MFLPD}\right] \text{ for two-loop operation or,}$$

$$S \leq (0.66W + 54\% - 0.66\Delta W)\left[\frac{FRP}{MFLPD}\right] \text{ for single-loop operation}$$

where:

FRP = fraction of rated thermal power (2436 MWt)

MFLPD = maximum fraction of limiting power density where the limiting power density is 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:

$S \leq 120\%$ Power

A.1.d APRM Rod Block Trip Setting

The APRM Rod block trip setting shall be:

$S \leq (0.66W + 42\%)$ for two-loop operation or:

$S \leq (0.66W + 42\% - 0.66\Delta W)$ for single-loop operation.

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)

ΔW = Difference between two-loop and single-loop effective drive flow at the same core flow.
= 0 for two recirculation loop operation.
= for one recirculation loop operation. (to be determined upon implementation of single-loop operation.)

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:

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

$$S \leq (0.66W + 42\% - 0.66\Delta W)\left[\frac{FRP}{MFLPD}\right] \text{ for single-loop operation.}$$

where:

MFLPD = maximum fraction of limiting power density where the limiting power density is 13.4 KW/ft for 8X8, 8X8R and P8X8R Fuel.

FRP = fraction of rated thermal power (2436 MWT)

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 the actual operating value will be used.

1.1 BASES

1.1 FUEL CLADDING INTEGRITY

The fuel cladding integrity limit is set such that no calculated fuel damage would occur as a result of an abnormal operational transient. Because fuel damage is not directly observable, a step-back approach is used to establish a Safety Limit such that the minimum critical power ratio (MCPR) is no less than 1.07. MCPR > 1.07 represents a conservative margin relative to the conditions required to maintain fuel cladding integrity. The fuel cladding is one of the physical barriers which separate radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding, perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions and the protection system safety settings. While fission product migration from cladding perforation is just as measurable as that from use related cracking, the thermally caused cladding perforations signal a threshold, beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding Safety Limit is defined with margin to the conditions which would produce onset of transition boiling, (MCPR of 1.0). These conditions represent a significant departure from the condition intended by design for planned operation.

A. Reactor Pressure > 785 psig and Core Flow > 10% of Rated

Onset of transition boiling results in a decrease in heat transfer from the clad and, therefore, elevated clad temperature and the possibility of clad failure. However, the existence of critical power, or boiling transition, is not a directly observable parameter in an operating reactor. Therefore, the margin to boiling transition is calculated from plant operating parameters such as core power, core flow, feedwater temperature, and core power distribution. The margin for each fuel assembly is characterized by the critical power ratio (CPR) which is the ratio of the bundle power which would produce onset of transition boiling divided by the actual bundle power. The minimum value of this ratio for any bundle in the core is the minimum critical power ratio (MCPR). It is assumed that the plant operation is controlled to the nominal protective setpoints via the instrumented variables, i.e., normal plant operation presented on Figure 1.1-1 by the nominal expected flow control line. The Safety Limit (MCPR of 1.07) has sufficient conservatism to assure that in the event of an abnormal operational transient initiated from the MCPR operating limits specified for the normal operating conditions in specification 3.1.B, more than 99.9% of the fuel rods in the core are expected to avoid boiling transition. The MCPR fuel cladding safety limit is increased by 0.01 for single loop operation as discussed in Reference 2. The margin between MCPR of 1.0 (onset of transition boiling) and the Safety Limit is derived from a detailed statistical analysis considering all of the uncertainties in monitoring the core operating state including uncertainty in the boiling transition correlation as described in Reference 1. The uncertainties employed in deriving the Safety Limit are

1.1 BASES (Cont'd.)

C. Power Transient

Plant safety analyses have shown that the scrams caused by exceeding any safety system setting will assure that the Safety Limit of 1.1.A or 1.1.B will not be exceeded. Scram times are checked periodically to assure the insertion times are adequate. The thermal power transient resulting when a scram is accomplished other than by the expected scram signal (e.g., scram from neutron flux following closure of the main turbine stop valves) does not necessarily cause fuel damage. However, for this specification a Safety Limit violation will be assumed when a scram is only accomplished by means of a backup feature of the plant design. The concept of not approaching a Safety Limit provided scram signals are operable is supported by the extensive plant safety analysis.

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

During periods when the reactor is shut down, consideration must also be given to water level requirements due to the effect of decay heat. If reactor water level should drop below the top of the active fuel during this time, the ability to cool the core is reduced. This reduction in core cooling capability could lead to elevated cladding temperatures and clad perforation. The core will be cooled sufficiently to prevent clad melting should the water level be reduced to two-thirds the core height. Establishment of the safety limit at 18 in. above the top of the fuel provides adequate margin. This level will be continuously monitered whenever the recirculation pumps are not operating.

E. References

1. Generic Reload Fuel Application General Electric BWR Thermal Analysis Basis (GETAB) Data, Correlation and Design Application, NEDO 10958 and NEDE 10958. NEDE - 24011-P-A and Appendices

2. FitzPatrick Nuclear Power Plant Single-Loop Operation NEDO - 24281, August 1980

3. Generic Reload Fuel Application, NEDE-24011-P-A and Appendices

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.

The transient analyses performed for each reload are given in Reference 2. Models and model conservatism are also described in this reference. As discussed in Reference 4, the core wide transient analysis for one recirculation pump operation is conservatively bounded by two-loop operation analysis, and the flow-dependent rod block and scram setpoint equations are adjusted for one-pump operation.

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' 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

4. "FitzPatrick Nuclear Power Plant Single-Loop Operation "NEDO-24281, August 1980.

MCPR Operating Limit for Incremental Cycle Core Average Exposure

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

For one recirculation loop operation the MCPR limits at rated flow are 0.01 higher than the comparable two-loop values.

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:

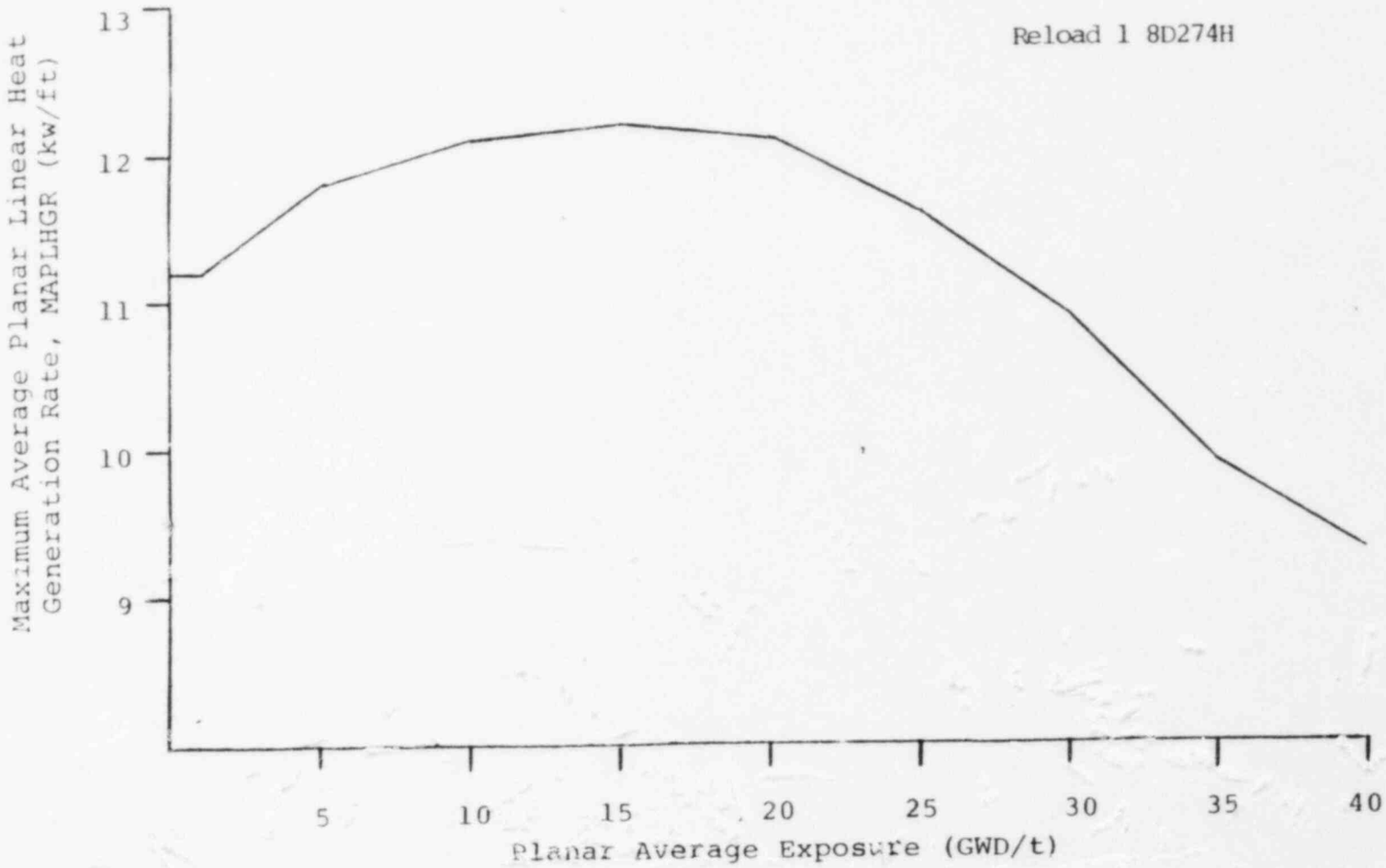
1. The average scram time to notch position 38 shall be: $\tau_{AVE} \leq \tau_B$

2. The average scram time to notch position 38 is determined as follows:

$$\tau_{AVE} = \sum_{i=1}^{n} Ni\ \tau i \Big/ \sum_{i=1}^{n} Ni$$

where: n = number of surveillance tests performed to date in the cycle, Ni = number of active rods measured in

Fig 3.5-4



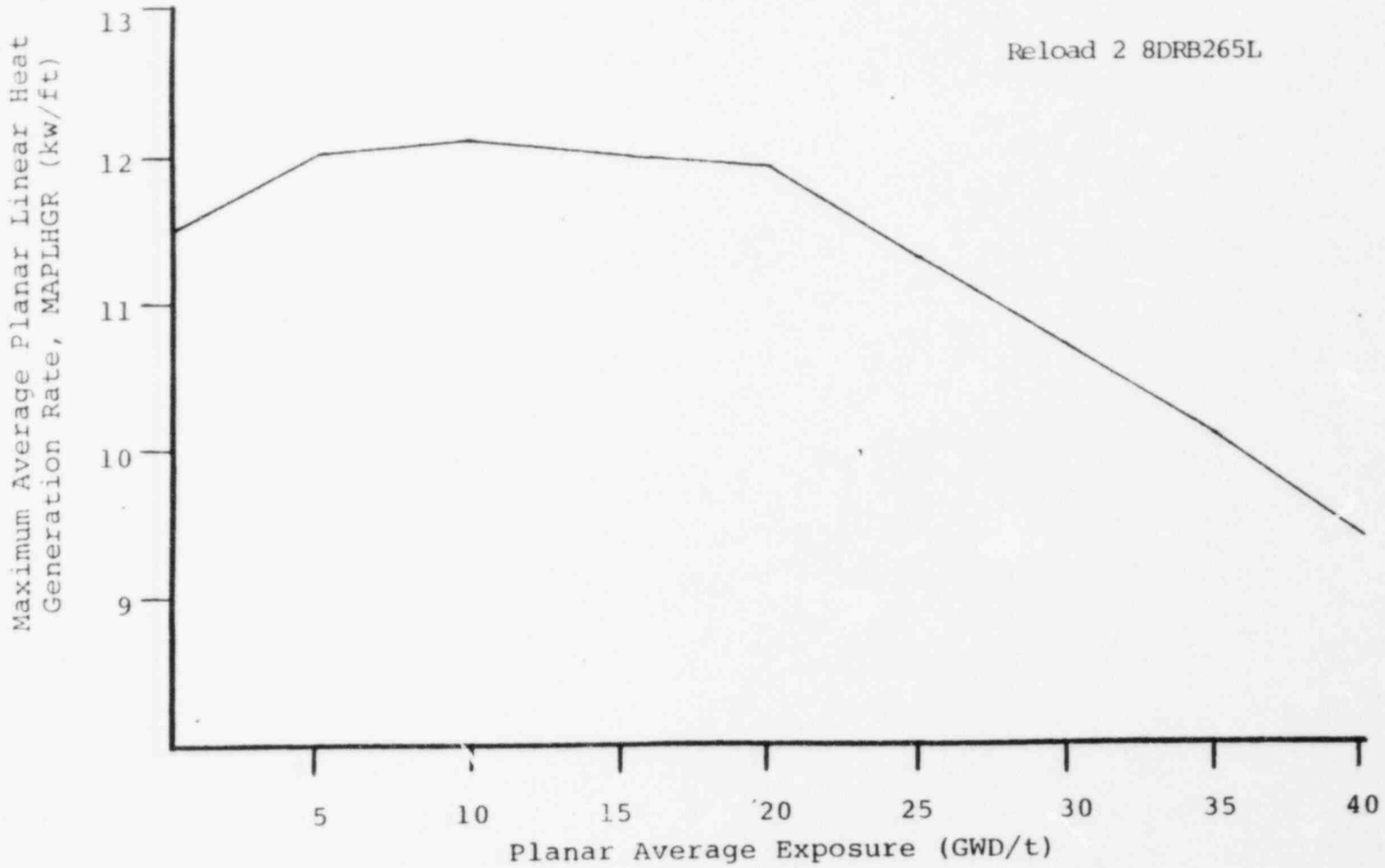
Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) Versus Planar Average Exposure

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

For single recirculation loop operation, these MAPLHGR values should be multiplied 0.84.

Amendment No. 36, 64

Fig 3.5-5



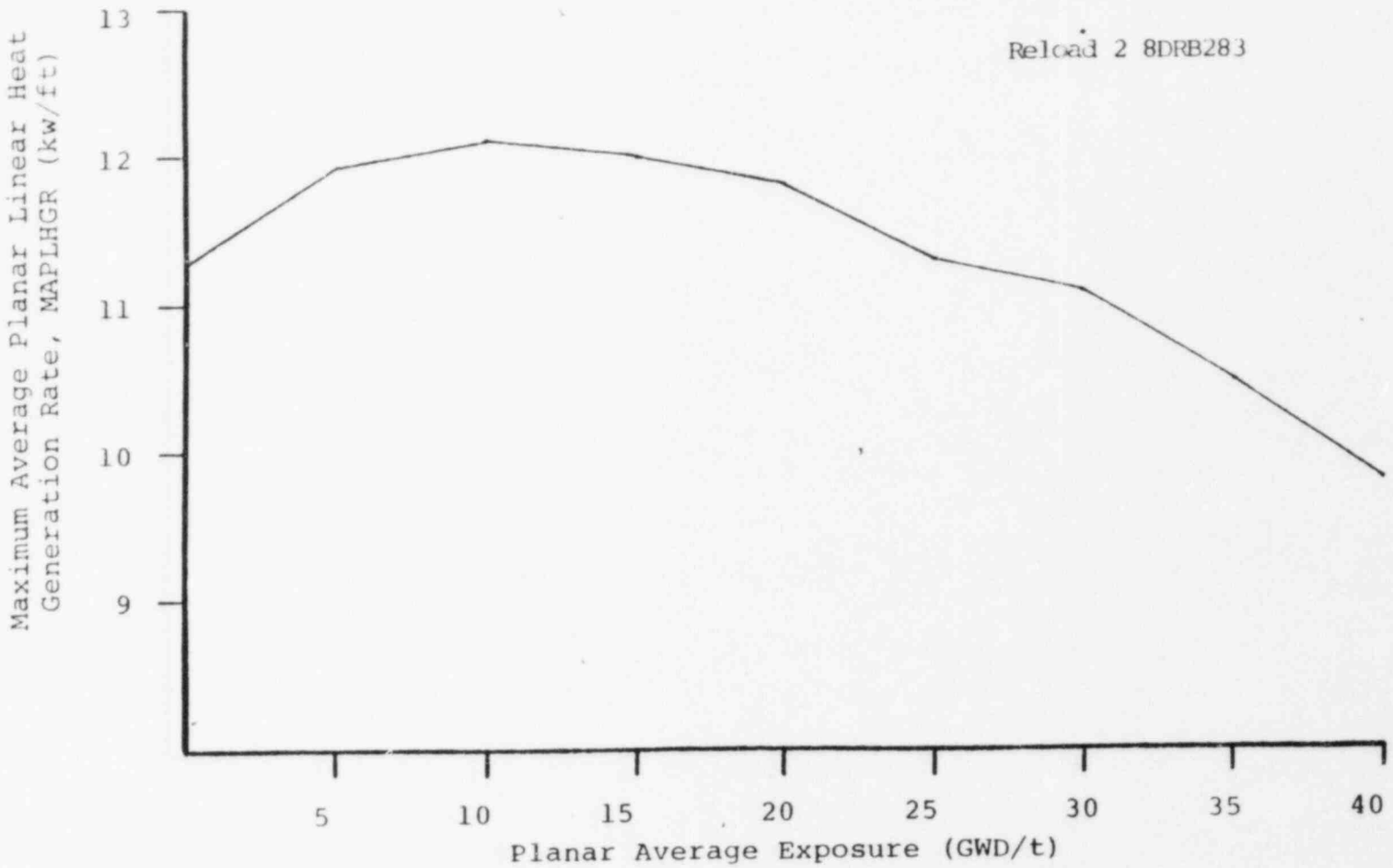
Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) Versus Planar Average Exposure

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

For single recirculation loop operation, these MAPLHGR values should be multiplied 0.84.

Amendment No. 45, 64

Fig 3.5-6



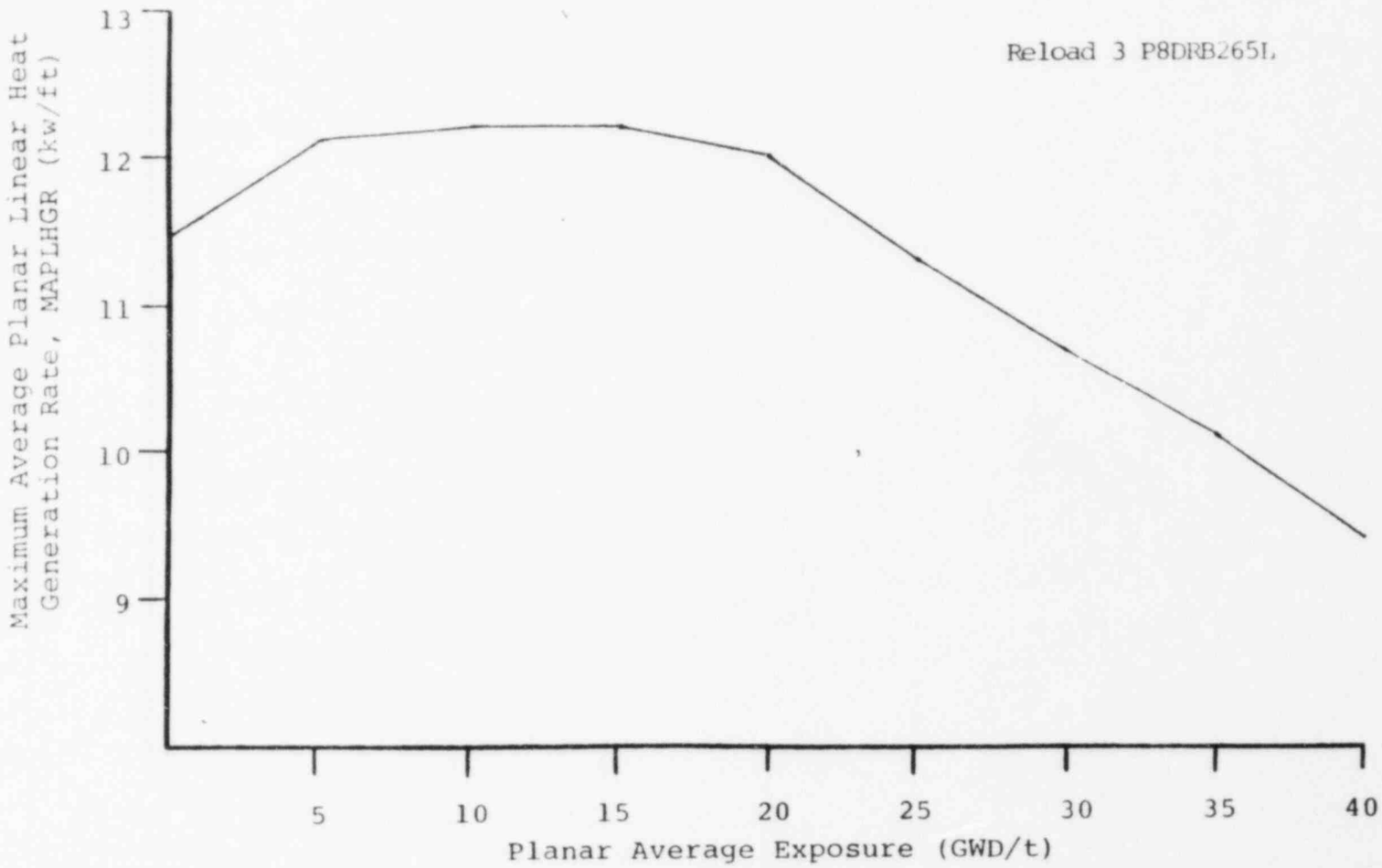
Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) Versus Planar Average Exposure

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

For single recirculation loop operation, these MAPLHGR values should be multiplied 0.84.

Amendment No. 45, 64

Fig 3.5-7



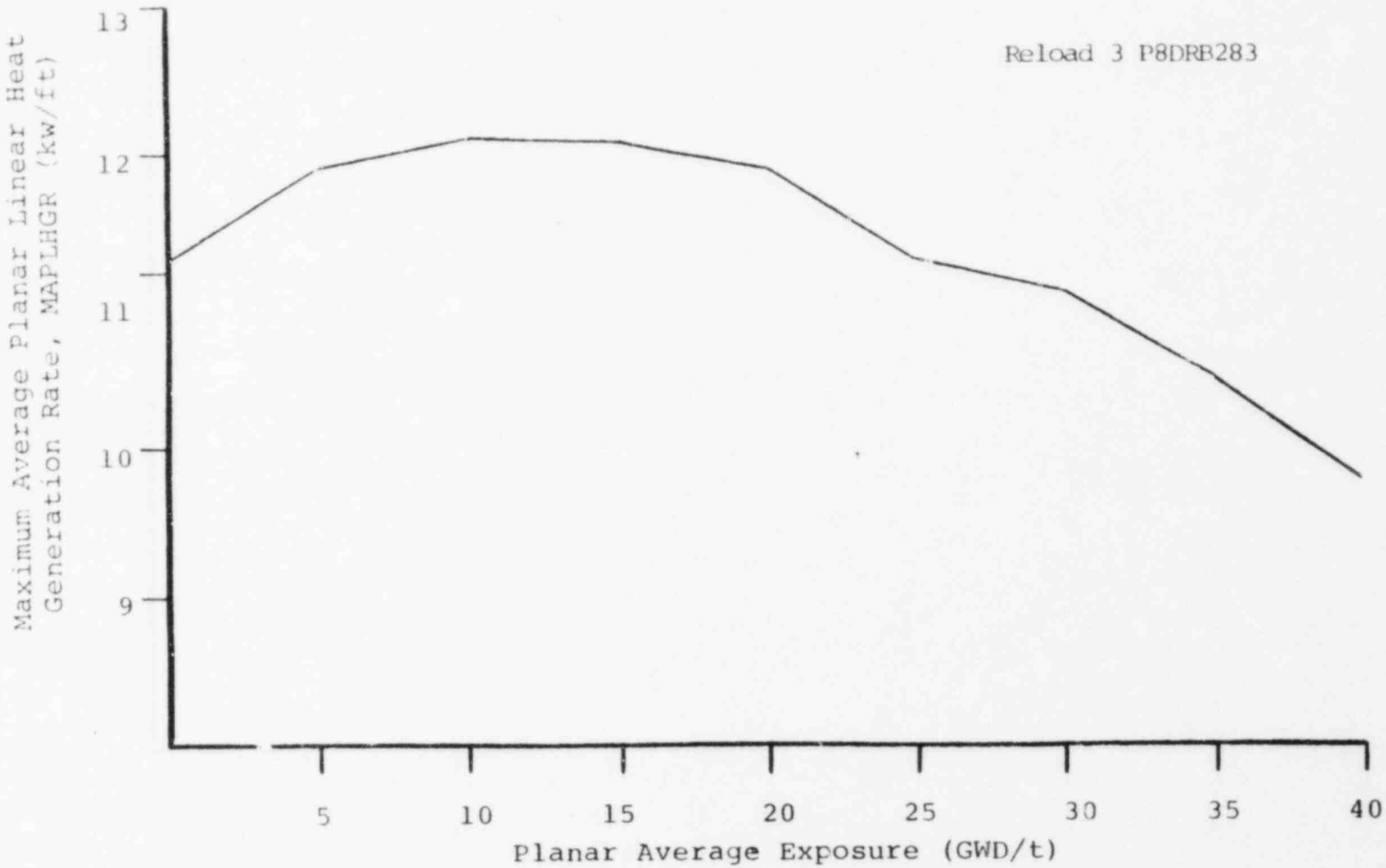
Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) Versus Planar Average Exposure

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

For single recirculation loop operation, these MAPLHGR values should be multiplied 0.84.

Amendment No. ~~45~~, ~~54~~

Fig 3.5-8



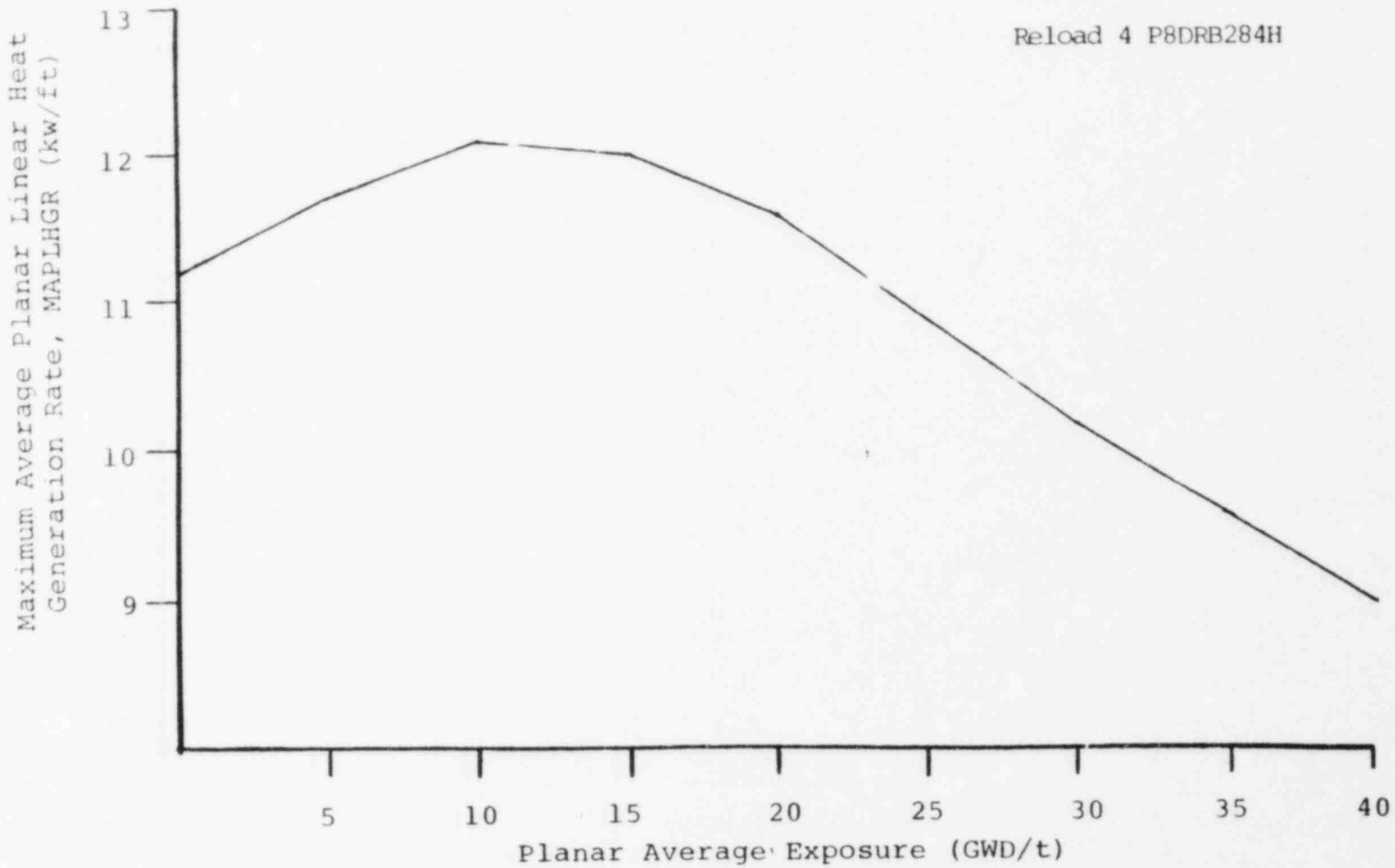
Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) Versus Planar Average Exposure

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

For single recirculation loop operation, these MAPLHGR values should be multiplied 0.84.

Amendment No. ~~46~~, ~~64~~

Figure 3.5-9



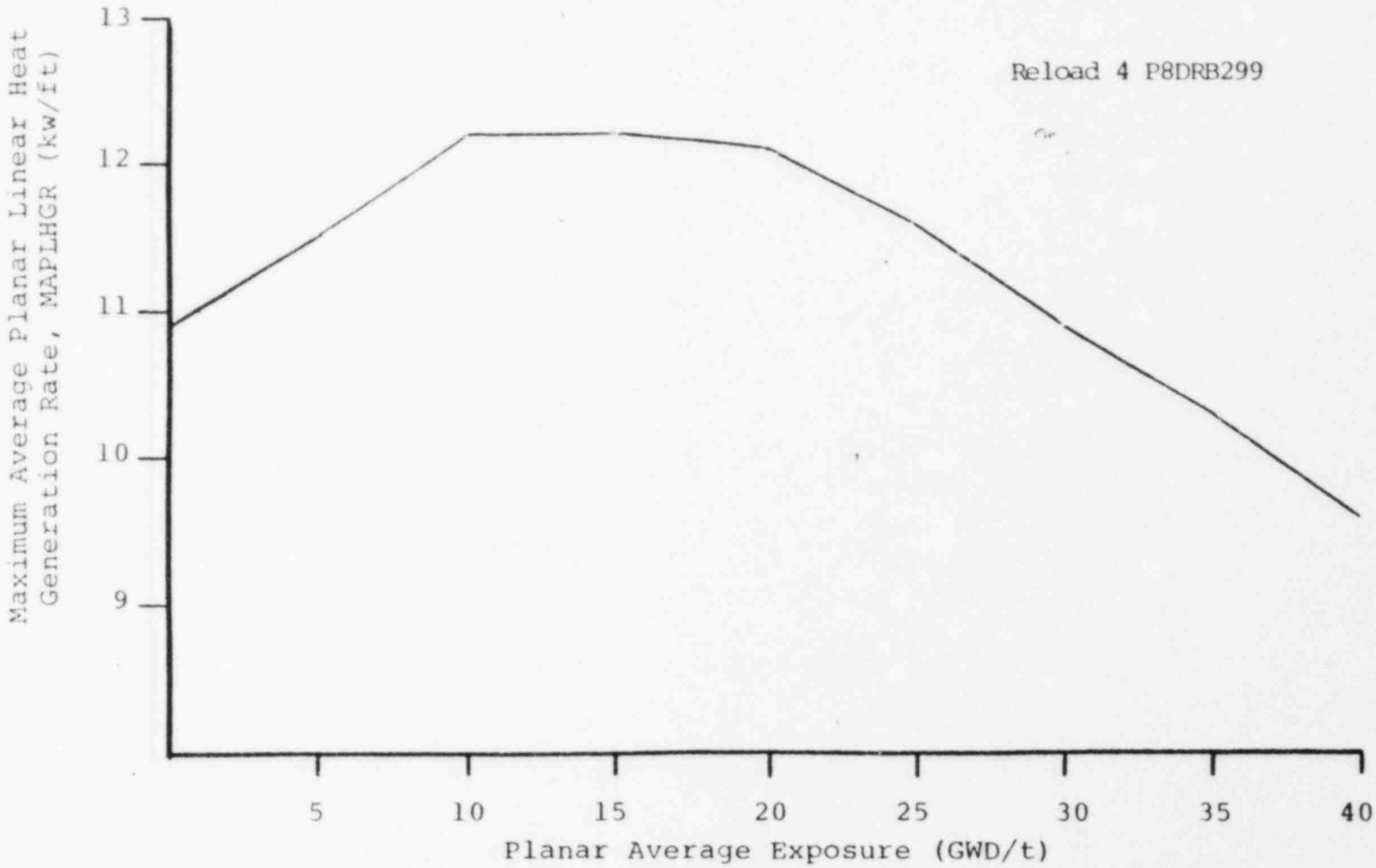
Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) Versus Planar Average Exposure

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

For single recirculation loop operation, these MAPLHGR values should be multiplied 0.84.

Amendment No. 68

Figure 3.5-10



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

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

For single recirculation loop operation, these MAPLHGR values should be multiplied 0.84.

Amendment No. 54

3.6 (cont'd)

F. Structural Integrity

The structural integrity of the reactor coolant system shall be maintained at the level required by the original acceptance standards throughout the life of the Plant.

G. Jet Pumps

Whenever the Reactor is in the startup/hot standby or run modes, all jet pumps shall be operable. If it is determined that a jet pump is not operable, the Reactor shall be placed in a cold condition within 24 hours.

4.6 (cont'd)

F. Structural Integrity

1. The nondestructive inspections listed in Table 4.6.1 shall be performed as specified. The results obtained from compliance with this specification will be evaluated after 5 years and the conclusions of this evaluation will be reviewed with the AEC.

2. An augmented in-service inspection program is required for those high stressed circumferential piping joints in the main steam and feed-water lines larger than 4 inches in diameter, where no restraint against pipe whip is provided.

   The augmented in-service inspection program shall consist of 100 percent inspection of the welds in place of the 25 percent inspection per inspection interval required by Section XI of the 1970 Edition of the ASME Boiler and Pressure Vessel Code.

G. Jet Pumps

1. Whenever there is two-loop recirculation flow with the reactor in the startup/hot standby or run modes, jet pump operability shall be checked daily by verifying that the following conditions do not occur simultaneously:

3.6 (cont'd)

4.6 (cont'd)

a. The two recirculation loops have a flow imbalance of 15 percent or more when the pump are operated at the same speed.

b. The indicated value of core flow rate varies from the value derived from loop flow measurements by more than 10 percent.

c. The diffuser to lower plenum differential pressure reading on an individual jet pump varies from the average of all jet pump differential pressures by more than 10 percent.

2. Whenever the reactor is in the startup/hot standby or run modes, and there is one loop recirculation flow, jet pump operability shall be verified as follows:

a. Baseline readings will be taken and operating characteristics for the following parameters established:

1. Jet Pump Loop Flow and Recirculation Pump Speed for the operating loop.

2. Individual Jet Pump percent differential pressures for all jet pumps.

b. Initially, and daily thereafter, jet pump operability will be verified by assuring that the following do not occur simultaneously:

3.6 (cont'd)

H. Jet Pump Flow Mismatch

1. When both recirculation pumps are in steady state operation, the speed of the faster pump may not exceed 122 percent the speed of the slower pump when core power is 80 percent or more of rated power, or 135 percent the speed of the slower pump when core power is below 80 percent of rated power.

2. Following one-pump operation, the discharge valve of the low speed pump may not be opened unless the speed of the faster pump is less than 50 percent of its rated speed.

4.6 (cont'd)

1. The ratio of jet pump loop flow to recirculation pump speed for the operating loop does not vary from the initially established value by more than 10 percent.

2. The ratio of individual jet pump percent differential pressure to the loop's average jet pump percent differential pressure does not vary from the initially established value by more than 20 percent.

H. Jet Pump Flow Mismatach

1. Recirculation pump speeds shall be checked and logged at least once/day.

3. Operation with a single recirculation loop is permitted with the designated adjustments for: APRM rod block and scram setpoints (Technical Specifications 2.1.A.1.c, 2.1.A.1.d, and Tables 3.1-1 and 3.2-3); RBM setpoint, Table 3.2-3; MCPR fuel cladding integrity safety limit and operating limits (Tech. Specs. 1.1.A and 3.1.B, respectively); and MAPLHGR (Tech. Spec. 3.5.H).

3.6 (cont'd)

1.6.1 Shock Suppressors (Snubbers)

Applicability

Applies to the operational status of the shock suppressors (snubbers).

Objective

To assure the capability of the snubbers to:

Prevent unrestrained pipe motion under dynamic loads as might occur during an earthquake or severe transient, and

Allow normal thermal motion during startup and shutdown.

Specification

1. During all modes operation except Cold Shutdown and Refuel, all snubbers which are required to protect the primary coolant system or any other safety related system or component shall be operable except as noted in 3.6.1.2 through 3.6.1.5 below. These safety related snubbers are listed in Table 3.6-1.

4.6 (cont'd)

4.6.1 Shock Suppressors (Snubber)

Applicability

Applies to the periodic testing requirement for the hydraulic shock suppressors (snubbers).

Objective

To assure the operability of the snubbers to perform their intended functions.

Specification

The following surveillance requirements apply to all hydraulic snubbers listed in Table 3.6-1.

1. All hydraulic snubbers whose sent material has been demonstrated by operating experience, lab testing or analysis to be compatible with the operating environment shall be visually inspected. This inspection shall include, but not necessarily be limited to, inspection of the hydraulic fluid reservoir, fluid connections, and linkage connections to the piping and anchor to verify snubber operability in accordance with the following schedule:

Number of Snubbers Found Inoperable During Inspection or During Inspection Interval	Next Required Inspection Interval
0	18 months ± 25%
1	12 months ± 25%
2	6 months ± 25%

2. From and after the time that a snubber is determined to be inoperable, continued reactor operation is permissible only during the succeeding 72 hours unless the snubber is sooner made operable or replaced.

3. If the requirements of 3.6.1.1 and 3.6.1.2 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 36 hours.

4. If a snubber is determined to be inoperable while the reactor is in the shutdown or refuel mode, the snubber shall be made operable or replaced prior to reactor startup.

5. Snubbers may be added to safety related systems without prior License Amendment to Table 3.6.1 provided that a revision to Table 3.6.1 is included with the next License Amendment request.

1. Shock Suppressors (Snubbers) (cont'd)

3,4	124 days ± 25%
5,6,7	62 days ± 25%
≥ 8	31 days ± 25%

The required inspection interval shall not be lengthened more than one step at a time.

Snubbers may be categorized in two groups, "accessible" or "inaccessible" based on their accessibility for inspection during reactor operation. These two groups may be inspected independently according to the above schedule.

2. All hydraulic snubbers whose seal materials are other than ethylene propylene or other material that has not been demonstrated to be compatible with the operating environment shall be visually inspected for operability every 31 days.

3. Once each refueling cycle, a representative sample of 10 hydraulic snubbers or piston approximately 10% of the hydraulic snubbers whichever is less, shall be functionally tested for operability including verification of proper movement, lock up and bleed. For each unit and subsequent unit found inoperable, an additional 10% or ten hydraulic snubbers shall be so tested until no more failures are found or all units have been tested. Snubbers of rated capcity greater than 50,000 lbs. need not be functionally tested.

ATTACHMENT II

SAFETY EVALUATION

RELATED TO

SINGLE RECIRCULATION LOOP OPERATION

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

Section I - Description of the Changes

The proposed amendment allows operation of the reactor with one recirculation loop out of service at reduced power. Adjustments are made to the flow biased APRM Scram and Rod Block lines, as well as the flow biased Rod Block Monitor line to account for the change in core flow versus recirculation flow behavior during single-loop operation. The MCPR fuel cladding integrity safety limit is raised by 0.01 during single-loop operation for increased uncertainties in total core flow and TIP reading. A reduction factor is applied to each MAPLHGR curve to account for the difference in LOCA analysis of one loop versus two-loop operation. The largest contribution to a decrease in MAPLHGR for one-loop operation is the very short time to onset of boiling transition used, 0.1 sec versus approximately 10 sec in the two-loop analysis. This conservative assumption is a result of the decrease in forced circulation provided by recirculation pump coast down during the early stages of LOCA.

A description of analyses referred to in this section can be found in the General Electric report NEDO-24281 (as amended) entitled, "FitzPatrick Nuclear Power Plant Single-Loop Operation". Copies of this report are included in the submittal to the NRC.

Surveillance testing of the operating recirculation loop will include verification of jet pump operability by monitoring jet pump instruments, and comparing the readings with baseline readings. These baseline readings are taken each time single loop recirculation is initiated.

Section II - Purpose of the Changes

The current Technical Specifications for FitzPatrick do not allow plant operation for more than 24 hours with one recirculation loop out of service. However, the capability of operating at reduced power with a single recirculation loop is highly desirable from a plant availability standpoint in the event maintenance on a recirculation pump or other component renders one loop inoperative.

Section III - Impact of the Changes

The changes to the Technical Specifications outlined in Section I of this evaluation ensure that the consequences of plant transients and accidents analyzed in the FSAR are unaffected during the single-loop operation, and do not alter conclusions reached in the FSAR and SER accident analysis.

The reduced power in single-loop operations ensures that pressurization transients are less severe as seen in Figure 3-1 of NEDO-24281. MAPLHGR reduction factors account for the decrease in blowdown heat transfer during a LOCA and maintain the fuel within the 2200°F peak clad temperature limit required by Appendix K of 10 CFR 50.

Section IV - Implementation of the Changes

These changes, as proposed, will not impact the ALARA or Fire Protection Programs at JAF. These changes will not impact the environment.

Section V - Conclusion

The incorporation of these changes: 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 NEDO-24281, "FitzPatrick Nuclear Power Plant Single-Loop Operation'.

(d) General Electric Company Service Information Letter (SIL) No. 330, "Jet Pump Beam Cracks", June 9, 1980