[redacted]

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

October 24, 1983

Mr. John F. Doherty
318 Summit Avenue
Apt. #3
Brighton, Massachusetts 02135

Dear Mr. Doherty:

Re: Pilgrim Station, Application for Amendment dated May 12, 1981

By letter dated October 11, 1983 you requested a copy of the Application for Amendment to DPR-35, Pilgrim Station dated May 12, 1981 which was mentioned in the Federal Register of August 31, 1983 at page 39,538. Enclosed is the application you requested.

Sincerely,

D B Vassallo

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

Enclosure:
As stated

50-293

8311080317 XA

BOSTON EDISON COMPANY
800 BOYLSTON STREET
BOSTON, MASSACHUSETTS 02199

J HOWARD
ENT

May 12, 1981

BECo. Ltr. #81-94

Proposed Change #81-02

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U.S. NUCLEAR REGULATORY COMMISSION

Mr. Thomas A. Ippolito, Chief
Operating Reactors Branch #2
Division of Licensing
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555

License No. DPR-35
Docket No. 50-293

Revised Request for Technical Specification Changes
Concerning Single Loop Operation

Reference (a) BECo letter #80-295 (J.E. Howard) to NRC (T.A. Ippolito) "Proposed Technical Specification Change Concerning Single Loop Operation", dated November 21, 1980

Dear Sir:

Facility Operating License No. DPR-35 for Pilgrim Nuclear Power Station Unit #1 (PNPS-1) requires that the plant be shutdown if an idle recirculation loop cannot be returned to service within 24 hours.

Boston Edison requests that this license provision be revised to allow operation with one recirculation loop out of service. It is also requested that this proposed change supercede in it's entirety, BECo. letter #80-295 dated November 21, 1980,(ref. (a))which included provisions for 50% reactor power with one recirculation loop out of service.

The loss of a single recirculation pump has occurred at several operating BWR's and is not, therefore, an improbable event. While the time required to procure necessary parts and to repair the loop depends on the nature of the failure, any loss of operating capacity would have a significant economic effect. Modifying the Technical Specifications as requested will reduce this potential economic impact without reducing the safety of plant operation.

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The evaluation of this proposed mode of operation provided to Boston Edison by GE and described in Attachment (C), supports the conclusion that this mode of operation will not reduce safety margins.

Schedule of Change

This change will be put into effect upon receipt of approval from the Commission.

Fee Consideration

No fee is proposed, as this change supercedes the Reference (a) letter which included payment pursuant to 10 CFR 170.12 and is still under review by your office.

Should you have any questions on this subject, please do not hesitate to contact us.

Very truly yours,

J Edward Howard

Attachments

(A) - Modified Rod Block Equation

(B) - License & Technical Specification Changes

(C) - Pilgrim Nuclear Power Station Single Loop Operation, NEDO-24268, June 1980 with Errata and Addenda Sheet No. 1, Sept. 1980

3 signed originals and 37 copies

Commonwealth of Massachusetts)
County of Suffolk)

Then personally appeared before me J. Edward Howard, who, being duly sworn, did state that he is Vice President - Nuclear of Boston Edison Company, the applicant herein, and that he is duly authorized to execute and file the submittal contained herein in the name and on behalf of Boston Edison Company and that the statements in said submittal are true to the best of his knowledge and belief.

My Commission expires: July 6, 1984

Dorothy M. Lopes
Notary Public

Attachment A

Modified Rod Block Equation

This attachment describes the procedure for modifying the rod block equation for one-pump operation.

a. The two pump rod block equation in the existing Technical Specification is of the form:

$$RB = (mW + K)\% \tag{1}$$

where

RB = power at rod block in %

m = flow reference slope for the rod block monitor (RBM)

W = drive flow in % of rated

K = power at rod block in % when $W = 0$.

For the case of top level rod block at 100% flow, denoted RB_{100}:

$$RB_{100} = m(100) + K$$

or

$$K = RB_{100} - m(100)$$

Substituting for K in Equation (1), the two pump equation becomes:

$$RB = mW + \left[RB_{100} - m(100)\right] \tag{2}$$

b. Next, the core flow (F_c) versus drive flow (W) curves are determined for the two-pump and one-pump cases. For the two-pump case the core flow and drive flow are derived by measuring the differential pressures in the jet pumps and recirculation loop, respectively. Core flow for one pump operation must be corrected for the backflow through the inactive jet pumps thus:

Actual core flow (one pump) = Active jet pump flow - inactive jet pump flow.

Both the active and inactive flows are derived from the jet pump differential pressures. The drive flow is derived from the differential pressure measurement in the active recirculation loop. These two curves are plotted in Figure 1. The maximum difference between the one-pump and two-pump core flow is determined graphically. This occurs at about 35% drive flow which is denoted W.

c. Next, a horizontal line is drawn from the 35% drive flow point on the one pump curve to the two pump curve and the corresponding flow, W_2, is determined. Thus, $\Delta W = W_1 - W_2$.

The rod block equation corrected for one pump flow is:

$$RB = mW + \left[RB_{100} - m(100)\right] - \Delta RB$$

where

$$\Delta RB = RB_1 - RB_2 = m\Delta W$$

$$RB = mW + RB_{100} - m(100 + \Delta W) \qquad (3)$$

d. The constants from the Technical Specification are:

$m = 0.65$

$RB_{100} = 107$

From Figure 2:

$$\Delta W = W_1 - W_2 = 35 - 30 = 5$$

Evaluating in Equation (3), the one-pump rod block equation becomes:

$$RB = 0.65W + 107 - 0.65(100+5) = 0.65W + 38.7 \qquad (4)$$

This line is depicted in Figure 1 as the corrected rod block line for one-pump operation.

APRM TRIP SETTING

The APRM trip settings are flow biased in the same manner as the rod block monitor trip setting. Therefore, the APRM rod block and scram trip settings are subject to the same procedural changes as the rod block monitor trip setting discussed above.

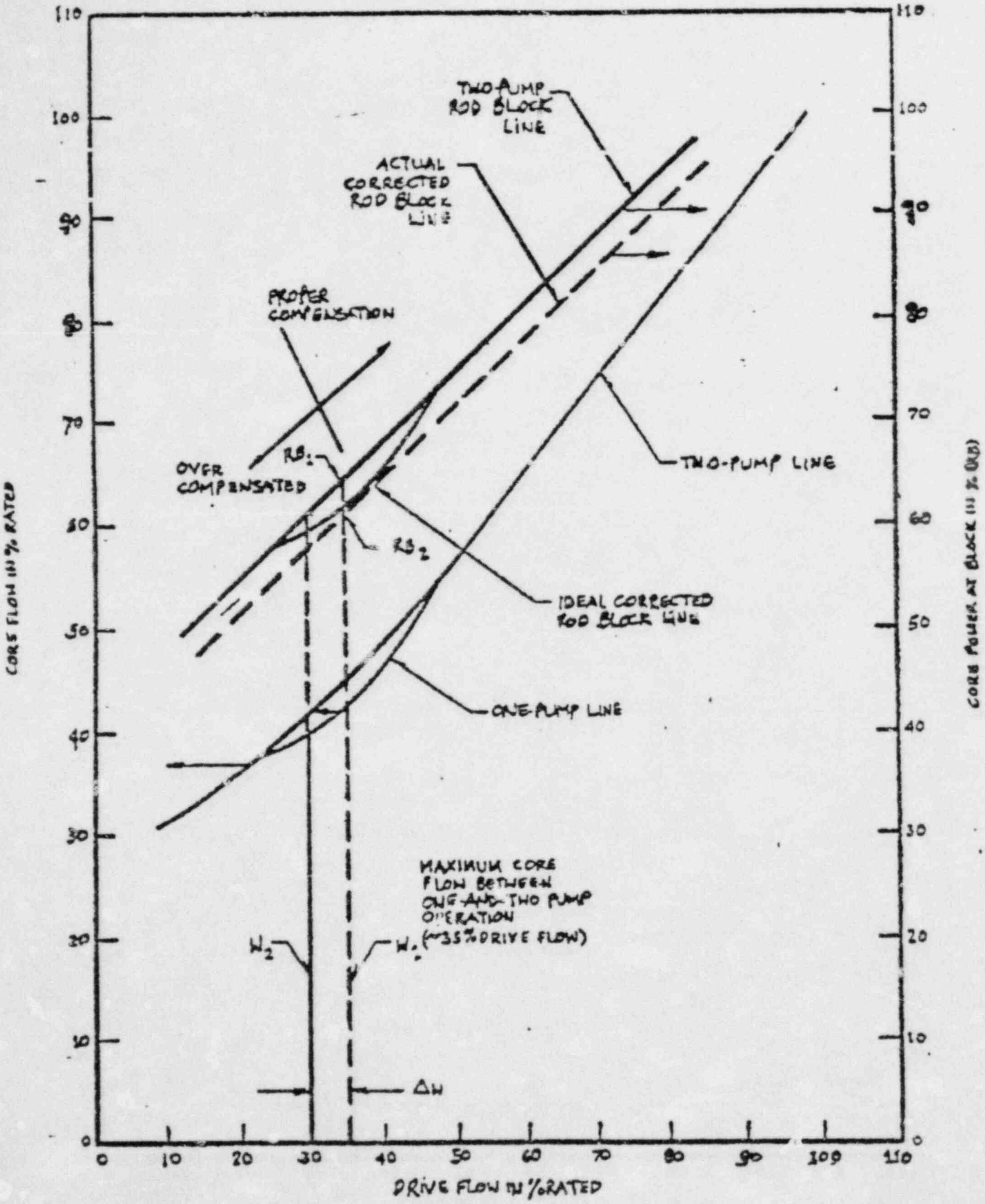

FIGURE 1 CORE FLOW VERSUS DRIVE FLOW FOR ONE-AND-TWO-PUMP OPERATION

ATTACHMENT B

License
and
Technical Specification Changes

Lic. Page 3

T.S. Pages 6
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205C-6
205D
205E-1
205E-2
205E-3
205E-4
205E-5

33pp.

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B. The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 48 are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

C. Records

Boston Edison shall keep facility operating records in accordance with the requirements of the Technical Specifications.

D. Equalizer Valve Restriction

The valves in the equalizer piping between the recirculation loops shall be closed at all times during reactor operation.

E. (Deleted)

F. Fire Protection

The licensee may proceed with and is required to complete the modifications identified in Paragraphs 3.1.1 through 3.1.19 of the NRC's Fire Protection Safety Evaluation (SE), dated December 21, 1978 for the facility. These modifications will be completed in accordance with the schedule in Table 3.1.

In addition, the licensee shall submit the additional information identified in Table 3.2 of this SE in accordance with the schedule contained therein. In the event these dates for submittal cannot be met, the licensee shall submit a report, explaining the circumstances, together with a revised schedule.

The licensee is required to implement the administrative controls identified in Section 6 of the SE. The administrative controls shall be in effect by December 31, 1978.

G. Physical Protection

The licensee shall fully implement and maintain in effect all provisions of the following Commission approved documents, including amendments and changes made pursuant to the authority of 10 CFR 50.54(p):

(1) "Security Plan for the Pilgrim Nuclear Power Station", dated Novmeber 7, 1977 with Revision 2 dated May 26, 1978 and Revision 3 dated January 8, 1979.

(2) "Pilgrim Nuclear Power Station Safeguards Contingency Plan", dated April 5, 1979 and revised by letters dated December 20, 1979 and April 22, 1980, submitted pursuant to 10 CFR 73.40. The Contingency Plan shall be fully implemented, in accordance with 10 CFR 73.40(b), within 30 days of the approval by the Commission.

(3) "Pilgrim Nuclear Power Station Guard Training and Qualification Plan", Revision 3, dated October 1980 includes pages dated August 18, 1979, May 28, 1980, and October 1, 1980. This Plan shall be followed in accordance with 10 CFR 73.55(b)(4), 60 days after approval by the Commission. All security personnel, as required in the above plans, shall be qualified within two years of this approval. The licensee may make changes to this plan without prior Commission approval if the changes do not decrease the safeguards effectiveness of the plan. The licensee shall maintain records of and submit reports concerning such changes in the

1.1 FUEL CLADDING INTEGRITY

Applicability:

Applies to the interrelated variables associated with fuel thermal behavior.

Objective:

To establish limits below which the integrity of the fuel cladding is preserved.

Specification:

A. Reactor Pressure > 800 psia and Core Flow >10% of Rated

The existance of a minimum critical power ratio (MCPR) less than 1.07 for two recirculation loop operation (1.08 for single-loop operation) shall constitute violation of the fuel cladding integrity safety limit. This value of the MCPR is hereinafter referred to as the Safety Limit MCPR.

B. Core Thermal Power Limit (Reactor Pressure ≤ 800 psia and/or Core Flow ≤ 10%)

When the reactor pressure is ≤ 800 psia or core flow is less than or equal to 10% of rated, the steady state core thermal power shall not exceed 25% of design thermal power.

C. Power Transient

The safety limit shall be assumed to be exceeded when scram is known to have been accomplished by a means other than the expected scram signal unless analyses demonstrate that the fuel cladding integrity safety limits defined in Specifications 1.1A and 1.1B were not exceeded during the actual transient.

2.1 FUEL CLADDING INTEGRITY

Applicability:

Applies to trip settings of the instruments and devices which are provided to prevent the reactor system safety limits from being exceeded.

Objective:

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.

Specification:

A. Neutron Flux Scram

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

1. Nuetron Flux Trip Settings

a. APRM Flux Scram Trip Setting (Run Mode)

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

$S \leq .65W + 55\%$ 2 loop
$S \leq .65W + 51.7\%$ 1 loop

Where:

S = Setting in percent of rated thermal power (1998 MWt)

W = Percent of drive flow to produce a rated core flow of 69 M lb/hr.

1.1 SAFETY LIMIT

D. Whenever the reactor is in the cold shutdown condition with irradiated fuel in the reactor vessel, the water level shall not be less than 12 in. above the top of the normal active fuel zone.

2.1 LIMITING SAFETY SYSTEM SETTING

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:

$$S \leq (0.65W + 55\%) \left[\frac{FRP}{MFLPD}\right] \quad \underline{\text{2 Loop}}$$

$$S \leq (0.65W + 51.7\%) \left[\frac{FRP}{MFLPD}\right] \quad \underline{\text{1 Loop}}$$

Where,

FRP = fraction of rated thermal power (1998 MWt)

MFLPD = maximum fraction of limiting power density where the limiting power density is 13.4 KW/ft for 8x8 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.

For no combination of loop recirculation flow rate and core thermal power shall the APRM flux scram trip setting be allowed to exceed 120% of rated thermal power.

b. <u>APRM Flux Scram Trip Setting (Refuel or Start and Hot Standby Mode)</u>

When the reactor mode switch is in the REFUEL or STARTUP position, the APRM scram shall be set at less than or equal to 15% of rated power.

c. <u>IRM</u>

The IRM flux scram setting shall be ≤120/125 of scale.

B. <u>APRM Rod Block Trip Setting</u>

The APRM rod block trip setting shall be:

$S_{RB} \leq 0.65W + 42\%$ <u>2 Loop</u>

$S_{RB} \leq 0.65W + 38.7\%$ <u>1 Loop</u>

Where,

S_{RB} = Rod block setting in percent of rated thermal power (1998 MWt)

W = Percent of drive flow required to produce a rated core flow of 69M lb/hr.

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

$$S_{RB} \leq (0.65\ W + 42\%) \left[\frac{FRP}{MFLPD}\right] \quad \underline{2\ \text{Loop}}$$

$$S_{RB} \leq (0.65\ W + 38.7\%) \left[\frac{FRP}{MFLPD}\right] \quad \underline{1\ \text{Loop}}$$

Where,

FRP = fraction of rated thermal power

MFLPD = maximum fraction of limiting power density where the limiting power density is 13.4 KW/ft for 8x8 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.

C. Reactor low water level scram setting shall be $\geq$ 9 in. on level instruments.

D. Turbine stop valve closure scram setting shall be $\leq$ 10 percent valve closure.

E. Turbine control valve fast closure setting shall be $\geq$ 150 psig control oil pressure at acceleration relay.

F. Condenser low vacuum scram setting shall be $\geq$ 23 in. Hg. vacuum.

G. Main steam isolation scram setting shall be $\leq$ 10 percent valve closure.

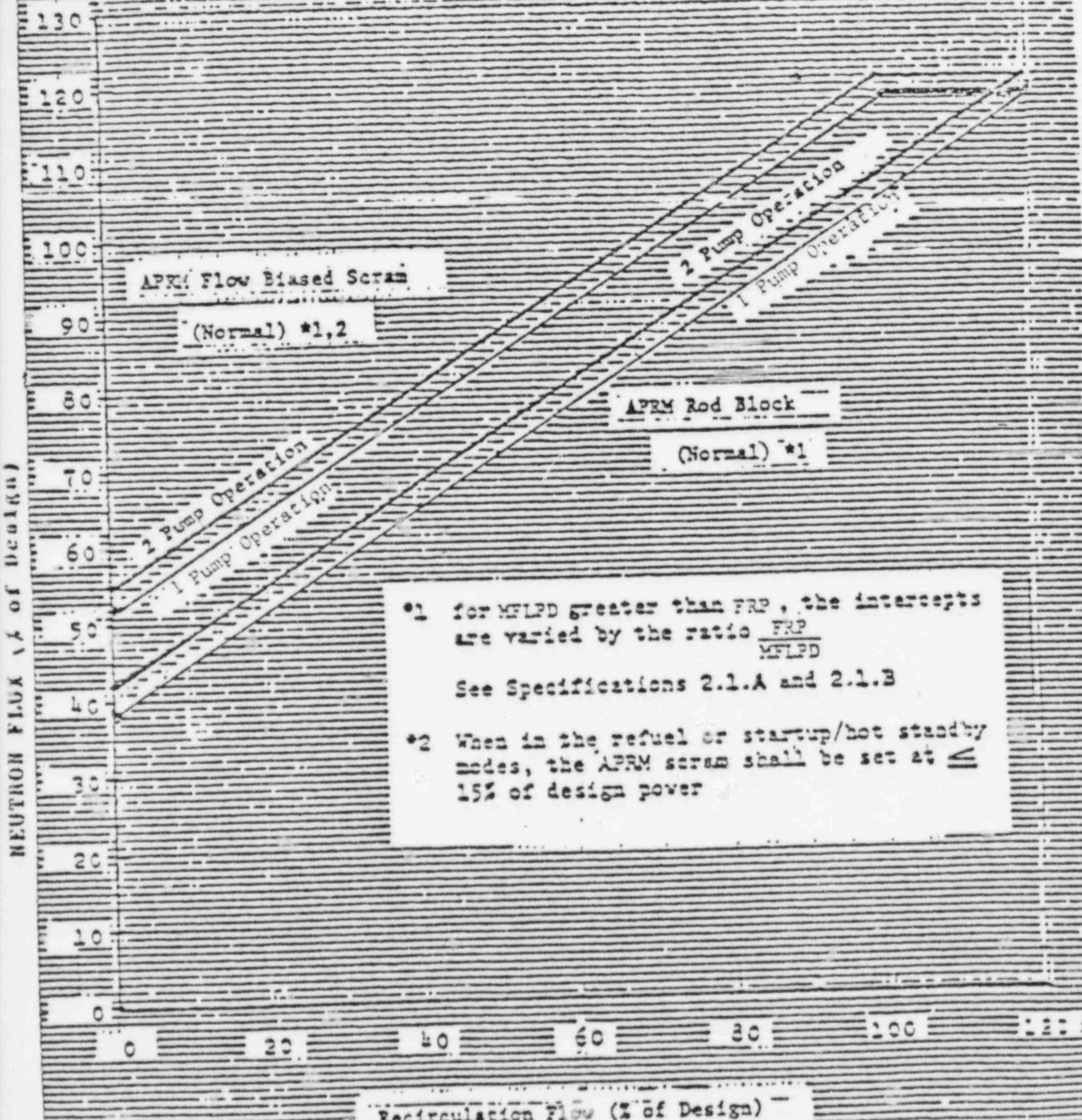

Figure 2.1.1

APRM Scram and Rod Block Trip Limiting Safety System Settings

BASES:

1.1 FUEL CLADDING INTEGRITY

A. Fuel Cladding Integrity Limit at Reactor Pressure ≥ 800 psia and Core Flow ≥ 10% of Rated

The fuel cladding integrity safety limit is set such that no fuel damage is calculated to occur if the limit is not violated. Since the parameters which result in fuel damage are not directly observable during reactor operation the thermal and hydraulic conditions resulting in a departure from nucleate boiling have been used to mark the beginning of the region where fuel damage could occur. Although it is recognized that a departure from nucleate boiling would not necessarily result in damage to BWR fuel rods, the critical power at which boiling-transition is calculated to occur has been adopted as a convenient limit. However, the uncertainties in monitoring the core operating state and in the procedure used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore the fuel cladding integrity safety limit is defined as the critical power ratio in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition considering the power distribution within the core and all uncertainties.

The Safety Limit MCPR is generically determined in Reference 1 (page 13) for two recirculation loop operation. This safety limit MCPR is increased by 0.01 for single loop operation as discussed in Reference 2 (page 13).

B. Core Thermal Power Limit (Reactor Pressure <800 psig or Core Flow <10% of Rated)

Since the pressure drop in the bypass region is essentially all elevation head which is 4.56 psi the core pressure drop at low power and all flows will always be greater than 4.56 psi. Analyses show that with a flow of $28x10^3$ 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 $28x10^3$ lbs/hr irrespective of total core flow and independent of bundle power for the range of bundle powers of concern. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors the 3.35 MWt bundle power corresponds to a core thermal power of more than 50%. Therefore a core thermal power limit of 25% for reactor pressures below 800 psia, or core flow less than 10% is conservative.

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C. Power Transient

Plant safety analyses have shown that the scrams caused by exceeding any safety setting will assure that the Safety Limit of Specification 1.1A or 1.1B 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 closures 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.

The computer provided with Pilgrim Unit 1 has a sequence annunciation program which will indicate the sequence in which events such as scram, APRM trip initiation, pressure scram initiation, etc. occur. This program also indicates when the scram setpoint is cleared. This will provide information on how long a scram condition exists and thus provide some measure of the energy added during a transient.

D. Reactor Water Level (Shutdown Condition)

During periods when the reactor is shutdown, 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 can be cooled sufficiently should the water level be reduced to two-thirds the core height. Establishment of the safety limit at 12 inches above the top of the fuel provides adequate margin. This level will be continuously monitored.

References

1. "General Electric Boiling Water Reactor Generic Reload Fuel Application", NEDE-24011-P - A-1, July 1979.

2. "Pilgrim Nuclear Power Station Single-Loop Operation", NEDO-24268, June 1980.

2.1 FUEL CLADDING INTEGRITY

The abnormal operational transients applicable to operation of the PNPS 1 Unit have been analyzed throughout the spectrum of planned operating conditions up to the thermal power condition of 1998 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, 1998 MWt is the licensed maximum power level of PNPS 1, and this represents the maximum steady-state power which shall not knowingly be exceeded.

Transient analyses performed each reload are given in Reference 1 (page 20). Models and model conservatisms are also described in this reference. As discussed in Reference 2 (page 20), the core wide transient analyses for one recirulation pump operation is conservatively bounded by two-loop operation analyses and the flow-dependent rod block and scram setpoint equations are adjusted for one-pump operation.

Steady-state operation without forced recirculation will not be permitted, except during startup testing.

2.1 BASES:

The bases for individual set points are discussed below:

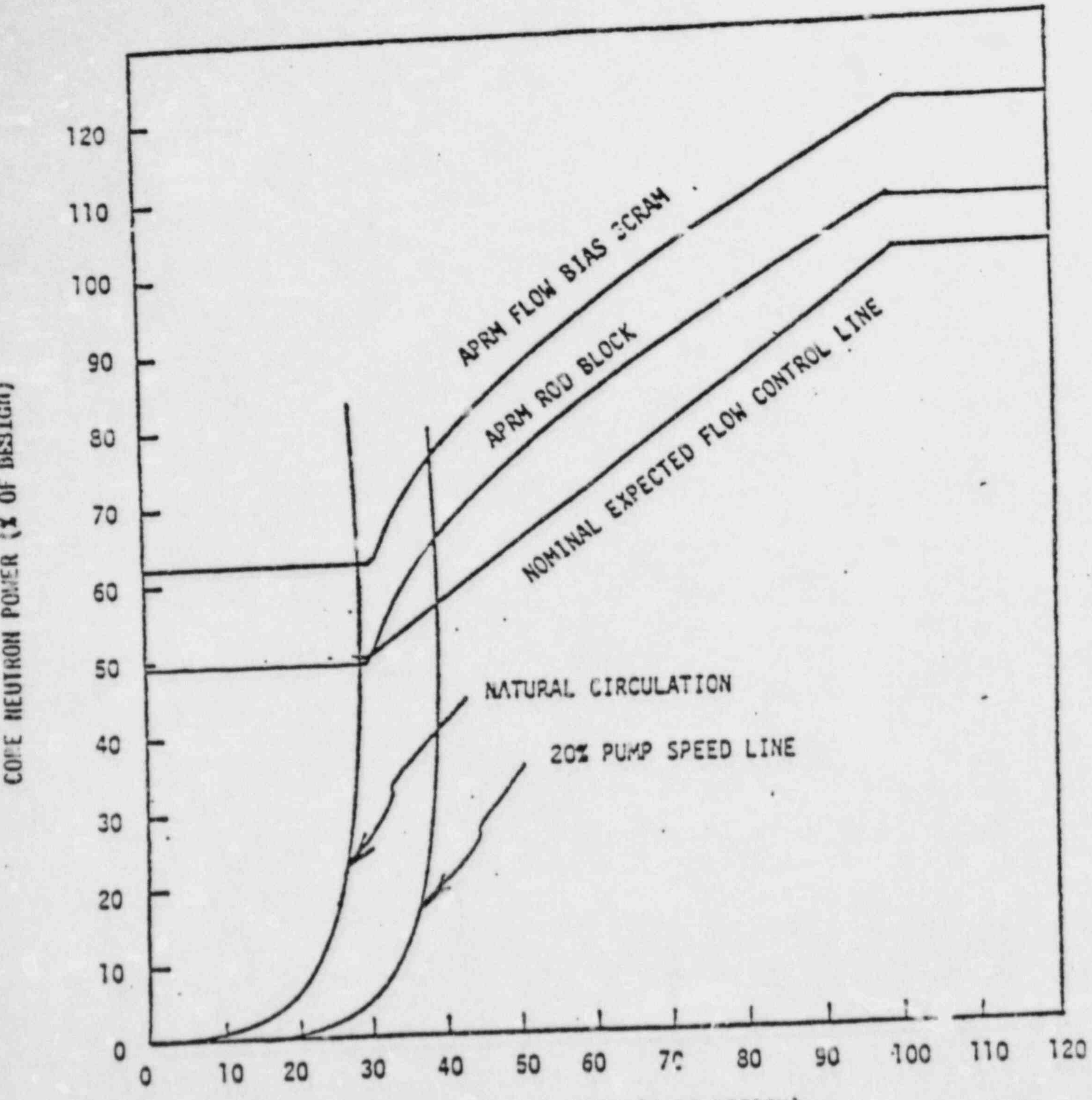
A. Neutron Flux Scram Trip Settings

APRM

The average power range monitoring (APRM) system, which is calibrated using heat balance data taken during steady-state conditions, reads in percent of design power (1998 MWt). Because fission chambers provide the basic input signals, the APRM system responds directly to average neutron flux. During transients, the instantaneous rate of heat transfer from the fuel (reactor thermal power) is less than the instantaneous neutron flux due to the time constant of the fuel. Therefore, during abnormal operational transients, the thermal power of the fuel will be less than that indicated by the neutron flux at the scram setting. Analyses demonstrated that with a 120 percent scram trip setting, none of the abnormal operational transients analyzed violate the fuel safety limit and there is a substantial margin from fuel damage. Therefore, the use of flow referenced scram trip provides even additional margin.

The flow biased scram plotted on Figure 2.1.1 is based on recirculation loop flow. Figure 2.1.3, which shows the flow biased scram as a function of core flow, has also been included.

An increase in the APRM scram setting would decrease the margin present before the fuel cladding integrity safety limit is reached. The APRM scram setting was determined by an analysis of margins required to provide a reasonable range for maneuvering during operation. Reducing this operating margin would increase the frequency of spurious scrams, which have an adverse effect on reactor safety because of the resulting thermal stresses. Thus, the APRM setting was selected because it provides adequate margin for the fuel cladding integrity safety limit yet allows operating margin that reduces the possibility of unnecessary scrams.



APRM FLOW BIAS SCRAM VERSUS REACTOR CORE FLOW
FIG. 2.1.3

Figure 2.1.3 above represents the APRM two loop flow bias scram with neutron flux plotted against core coolant flow rate instead of recirculation loop flow as shown in Figure 2.1.1.

setting. The actual power distribution in the core is established by specified control rod sequences and is monitored continuously by the in-core LPRM system. As with the APRM scram trip setting, the APRM rod block trip setting is adjusted downward if the maximum fraction of limiting power density exceeds the fraction of rated power, thus preserving the APRM rod block safety margin. Definition of single loop setpoints is given in Reference 3 (page 20).

C. Reactor Water Low Level Scram Trip Setting (LL1)

The set point for low level scram is above the bottom of the separator skirt. This level has been used in transient analyses dealing with coolant inventory decrease. The results show that scram at this level adequately protects the fuel and the pressure barrier, because MCPR remains well above the safety limit MCPR in all cases, and system pressure does not reach the safety valve settings. The scram setting is approximately 25 in. below the normal operating range and is thus adequate to avoid spurious scrams.

D. Turbine Stop Valve Closure Scram Trip Setting

The turbine stop valve closure scram anticipates the pressure, neutron flux and heat flux increase that could result from rapid closure of the turbine stop valves. With a scram trip setting of ≤ 10 percent of valve closure from full open, the resultant increase in surface heat flux is limited such that MCPR remains above the safety limit MCPR even during the worst case transient that assumes the turbine bypass is closed.

E. Turbine Control Valve Fast Closure Scram Trip Setting

The turbine control valve fast closure scram anticipates the pressure, neutron flux, and heat flux increase that could result from fast closure of the turbine control valves due to load rejection exceeding the capability of the bypass valves. The reactor protection system initiates a scram when fast closure of the control valves is initiated by the acceleration relay. This setting and the fact that control valve closure time is approximately twice as long as that for the stop valves means that resulting transients, while similar, are less severe than for stop valve closure. MCPR remains above the safety limit MCPR.

F. Main Condenser Low Vacuum Scram Trip Setting

To protect the main condenser against overpressure, a loss of condenser vacuum initiates automatic closure of the turbine stop valves and turbine bypass valves. To anticipate the transient and automatic scram resulting from the closure of the turbine stop valves, low condenser vacuum initiates a scram. The low vacuum scram set point is selected to initiate a scram before the closure of the turbine stop valves is initiated.

2.1 BASES:

Transient and accident analyses demonstrate that these conditions result in adequate safety margins for the fuel.

References

1. "Supplemental Reload Licensing Submittal for Pilgrim Nuclear Power Station, Reload 4", NEDO-24224

2. Final Safety Analysis Report for Pilgrim Nuclear Power Station Unit #1.

3. Attachment A, "Modified Rod Block Equation" to BECo letter (J. E. Howard) to NRC (T. A. Ippolito) dated May 12, 1981.

Min. Number Operable Inst. Channels per Trip (1) System	Trip Function	Trip Level Setting	Modes in Which Function Must Be Operable Refuel (7)	Startup/Hot Standby	Run	Action (1)
1	Mode Switch in Shutdown		X	X	X	A
1	Manual Scram		X	X	X	A
	IRM					
3	High Flux	$\leq$ 120/125 of full scale	X	X	(5)	A
3	Inoperative		X	X	(5)	A
	APRM					
2	High Flux	* (14) (15)	(17)	(17)	X	A or B
2	Inoperative		X	X(9)	X	A or
2	Downscale	$\geq$ 2.5 Indicated on Scale	(11)	(11)	X(12)	A or B
2	High Flux (15%)	$\leq$ 15% of Design Power	X	X	(16)	A or B
2	High Reactor Pressure	$\leq$ 1085 psig	X(10)	X	X	A
2	High Drywell Pressure	$\leq$ 2.5 psig	X(8)	X(8)	X	A
2	Reactor Low Water Level	$\geq$ 9 In. Indicated Level	X	X	X	A
2	High Water Level in Scram Discharge Tank	$\leq$ 39 Gallons	X(2)	X	X	A
2	Turbine Condenser Low Vacuum	$\geq$ 23 In. Hg Vacuum	X(3)	X(3)	X	A or C
2	Main Steam Line High Radiation	$\leq$ 7X Normal Full Power Background	X	X	X	A or C
4	Main Steam Line Isolation Valve Closure	$\leq$ 10% Valve Closure	X(3) (6)	X(3) (6)	X(6)	A or C
2	Turb. Cont. Valve Fast Closure	$\geq$ 150 psig Control Oil Pressure at Acceleration Relay	X(4)	X(4)	X(4)	A or D
4	Turbine Stop Valve Closure	$\leq$ 10% Valve Closure	X(4)	X(4)	X(4)	A or D

*APRM high flux scram setpoint $\leq (.65W + 55)\left[\frac{FRP}{MFLPD}\right]$ Two recirc. pump operation

or $\leq (.65W + 51.7)\left[\frac{FRP}{MFLPD}\right]$ One recirc. pump operation

PNPS

TABLE 3.2.C

INSTRUMENTATION THAT INITIATES ROD BLOCKS

Minimum # of Operable Instrument Channels Per Trip Systems (1)	Instrument	Trip Level Setting
2	APRM Upscale (Flow Biased)	Two Loop $(0.65W + 42)\left[\frac{FRP}{MFLPD}\right]$ (2) One Loop $(0.65W + 38.7)\left[\frac{FRP}{MFLPD}\right]$
2	APRM Downscale	2.[illegible] indicated on scale
1 (7)	Rod Block Monitor (Flow Biased)	Two Loop $(0.65W + 42)\left[\frac{FRP}{MFLPD}\right]$ (2) One Loop $(0.65W + 38.7)\left[\frac{FRP}{MFLPD}\right]$
1 (7)	Rod Block Monitor Downscale	5/125 of full scale
3	IRM Downscale (3)	5/125 of full scale
3	IRM Detector not in Startup Position	(8)
3	IRM Upscale	$\leq$108/125 of full scale
2 (5)	SRM Detector not in Startup Position	(4)
2 (5) (6)	SRM Upscale	$\leq 10^5$ counts/sec.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT
3.6.D Safety and Relief Valves (Cont'd) pressure shall be below 104 psig within 24 hours.	
E. Jet Pumps 1. Whenever the reactor is in the startup or run modes, all jet pumps shall be operable. If it is determined that a jet pump is inoperable, an orderly shutdown shall be initiated and the reactor shall be in a Cold Shutdown Condition within 24 hours.	E. Jet Pumps Whenever there is recirculation flow with the reactor in the startup or run modes, jet pump operability shall be checked daily by verifying that the following conditions do not occur simultaneously: 1. The two recirculation loops have a flow imbalance of 15% or more when the pumps are operated at the same speed. 2. The indicated value of core flow rate varies from the value derived from loop flow measurements by more than 10%. 3. The diffuser to lower plenum differential pressure reading on an individual jet pump varies from established jet pump P characteristics by more than 10%.
F. Jet Pump Flow Mismatch 1. Whenever both recirculation pumps are in operation, pump speeds shall be maintained within 10% of each other when power level is greater than 80% and within 15% of each other when power level is less than or equal to 80%. 2. Single loop reactor operation is not permitted for greater than 24 hours unless the following designated adjustments are made for APRM rod block and scram setpoints (Tech. Spec. 2.1.A and B) RBM setpoint (Table 3.2.C), MCPR fuel cladding integrity safety limit and operating limits (Tech Spec. 1.1.A and 3.11.C, respectively), and MAPLHGR (Tech Spec. 3.11.A).	F. Jet Pump Flow Mismatch Recirculation pump speeds shall be checked and logged at least once per day.

G. Structural Integrity

1. The structural integrity of the primary system boundary shall be maintained at the level required by the ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components." 1974

G. 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 AEC.

Edition (ASME Code, Section XI). In the interim until the nuclear system piping inspection evaluation level criteria of the ASME Boiler and Pressure Vessel Code, Section XI, 1974 Edition, are completed, the applicable evaluation level provisions of the ASME Boiler and Pressure Vessel Code, Section XI, 1971 Summer Addenda shall be used in the Inservice Inspection of nuclear piping. Components of the primary system boundary whose in-service examination reveals the absence of flaw indications not in excess of the allowable indication standards of this code are acceptable for continued service. Plant operation with components which have in-service examination flaw indication(s) in excess of the allowable indication standards of the Code shall be subject to NRC approval.

a. Components whose in-service examination reveals flaw indication(s) in excess of the allowable indication standards of the ASME Code, Section XI, are unacceptable for continued service unless the following requirements are met:

  (i) An analysis and evaluation of the detected flaw indication(s) shall be submitted to the NRC that demonstrate that the component structural integrity justifies continued service. The analysis and evaluation shall follow the procedures outlined in Appendix A, "Evaluation of Flaw Indications," of ASME Code, Section XI.

  (ii) Prior to the resumption of service, the NRC shall review the analysis and evaluation and either approve resumption of plant operation with the affected components or require that the component be repaired or replaced.

b. For components approved for continued service in accordance with paragraph a, above, reexamination of the area containing the flaw indication(s) shall be conducted during each scheduled successive in-service inspection. An analysis and evaluation shall be submitted to the NRC following each in-service inspection. The analysis and evaluation shall follow the procedures outlined in Appendix A, "Evaluation of Flaw Indications," of ASME Code, Section XI, and shall reference prior analyses submitted to the NRC to the extent applicable. Prior to resumption of service following each in-service inspection, the NRC shall review the analysis and evaluation and either approve resumption of plant operation with the affected components or require that the component be repaired or replaced.

c. Repair or replacement of components, including re-examinations, shall conform with the requirements of the ASME Code, Section XI. In the case of repairs, flaws shall be either removed or repaired to the extent necessary to meet the allowable indication standards specified in ASME Code, Section XI.

LIMITING CONDITIONS FOR OPERATION

3.11 REACTOR FUEL ASSEMBLY

Applicability

The Limiting Conditions for Operation associated with the fuel rods apply to those parameters which monitor the fuel rod operating conditions.

Objective

The Objective of the Limiting Conditions for Operation is to assure the performance of the fuel rods.

Specifications

A. Average Planar Linear Heat Generation Rate (APLHGR)

During power operation with both recirculation pumps operating, the APLHGR for each type of fuel as a function of average planar exposure shall not exceed the applicable limiting value shown in Figures 3.11-1 through 3.11-5. The top curves are applicable for core flow greater than or equal to 90% of rated core flow. When core flow is less than 90% of rated core flow, the lower curves shall be limiting. For greater than 24 hour operation with one recirculation pump, values from these curves are to be multiplied by 0.84 for 8x8 and 8x8R fuel. If at any time during operation it is determined by normal surveillance that the limiting value for APLHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the APLHGR is not returned within two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

SURVEILLANCE REQUIREMENTS

4.11 REACTOR FUEL ASSEMBLY

Applicability

The surveillance Requirements apply to the parameters which the fuel rod operating conditions.

Objective

The Objective of the Surveillance Requirements is to specify the type and frequency of surveillance to be applied to the fuel rods.

Specifications

A. 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.

C. Minimum Critical Power Ratio (MCPR)

During power operation MCPR for two recirculation loop operation shall be $\geq$ 1.35 for 8x8 and P8x8R fuel. If at any time during operation it is determined by normal surveillance that the limiting value for MCPR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the steady state MCPR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

For core flows other than rated the MCPR shall be $\geq$ 1.35 for 8x8 and P8x8R fuel times K_f, where K_f is as shown in Figure 3.11-8.

As an alternative method providing equivalent thermal-hydraulic protection at core flows other than rated, the calculated MCPR may be divided by K_f, where K_f is as shown in Figure 3.11-8.

For one recirculation loop operation, of greater than 24 hours, the MCPR limits at rated flow are 0.01 higher than the comparable two-loop values.

C. Minimum Critical Power Ratio (MCPR)

MCPR shall be determined daily during reactor power operation at >25% 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.

BASES

3.11A Average Planar Linear Heat Generation Rate (APLHGR)

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

The peak cladding temperature (PCT) 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. The peak clad temperature is calculated assuming a LHGR for the highest powered rod which is equal to or less than the design LHGR. This LHGR times 1.02 is used in the heat-up code along with the exposure dependent steady state gap conductance and rod-to-rod local peaking factors. The limiting value for APLHGR is this LHGR of the highest powered rod divided by its local peaking factor.

The calculational procedure used to establish the APLHGR limit for each fuel type is based on a loss-of-coolant accident analysis. The emergency core cooling system (ECCS) evaluation models which are employed to determine the effects of the loss of coolant accident (LOCA) in accordance with 10CFR50 and Appendix K are discussed in Reference 1. The models are identified as LAMB, SCAT, SAFE, REFLOOD, and CHASTE. The LAMB Code calculates the short term blowdown response and core flow, which are input into the SCAT code to calculate blowdown heat transfer coefficients. The SAFE code is used to determine longer term system response and flows from the various ECC systems. Where appropriate, the output of SAFE is used in the REFLOOD code to calculate liquid levels. The results of these codes are used in the CHASTE code to calculate fuel clad temperatures and maximum average planar linear heat generation rates (MAPLHGR) for each fuel type.

The significant plant input parameters are given in Reference 2. MAPLHGR's for the present fuel types were calculated by the above procedure and are included in Reference 3. The curves in Figures 3.11-1 through 3.11-5 were generated by multiplying the values in Reference 3 by factors given in Reference 4. These multipliers were developed assuming no core spray heat transfer credit in the LOCA analysis.

Reduction factors for one recirculation loop operation were derived in Reference 5.

REFERENCES

1. General Electric BWR Generic Reload Fuel Application, NEDE-24011-P.

2. Loss of Coolant Accident Analysis Report for Pilgrim Nuclear Power Station, NEDO-21696, August 1977.

3. "Supplemental Reload Licensing Submittal for Pilgrim Nuclear Power Station Unit 1 Reload 4", NEDO-24224, November 1979.

4. "Supplement 1 to Supplemental Reload Licensing Submittal for Pilgrim Nuclear Power Station Unit 1 Reload 4" NEDO-24224-1 March 1980.

5. "Pilgrim Nuclear Power Station Single Loop Operation", NEDO 24268, June 1980.

BASES:

3.11C MINIMUM CRITICAL POWER RATIO (MCPR)

Operating Limit MCPR

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 assuming instrument trip setting given in Specification 2.1. The required operating limit MCPR at steady state conditions in Specification 3.11.C was chosen conservatively at a value higher than MCPR's of past analysis with the objective of establishing an operating limit MCPR which is fuel type and cycle independant.

The difference between the specified Operating Limit MCPR in Specification 3.11C and the Safety Limit MCPR in Specification 1.1A defines the Largest reduction in critical power ratio (CPR) permitted during any anticipated abnormal operating transient. To ensure that this reduction is not exceeded, the most limiting transients are analized for each reload and fuel type (8x8 and P8x8R) to determine that transient which yields the largest value of Δ CPR. This value, when added to the Safety Limit MCPR must be less than the minimum operating limit MCPR's of Specification 3.11.C. The result of this evaluation is documented in the "Supplemental Reload Licensing Submittal" for the current reload.

Models used in the transient analyses are discussed in Reference 1 (page 205D).

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MCPR LIMITS FOR CORE FLOWS OTHER THAN RATED

The purpose of the K_f factor is to define operating limits at other than rated flow conditions. At less than 100% flow the required MCPR is the product of the operating limit MCPR and the K_f factor. Specifically, the K_f factor provides the required thermal margin to protect against a flow increase transient. The most limiting transient initiated from less than rated flow conditions is the recirculation pump speed up caused by a motor-generator speed control failure.

For operation in the automatic flow control mode, the K_f factors assure that the operating limit MCPR given in Specification 3.11C will not be violated should the most limiting transient occur at less than rated flow. In the manual flow control mode, the K_f factors assure that the Safety Limit MCPR will not be violated for the same postulated transient event.

The K_f factor curves shown in Figure 3.11-8[4] were developed generically which are applicable to all BWR/2, BWR/3, and BWR/4 reactors. The K_f factors were derived using the flow control line corresponding to rated thermal power at rated core flow as described in Reference 1 (page 206D).

The K_f factors shown in Figure 3.11-8[2] are conservative for the Pilgrim Unit 1 operation because the operating limit MCPR given in Specification 3.11C is greater than the original 1.20 operating limit MCPR used for the generic derivation of K_f.

1.C MINIMUM CRITICAL POWER RATIO (MCPR) - SURVEILLANCE REQUIREMENT

At core thermal power levels less than or equal to 25%, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience indicated that the resulting MCPR value is in excess of requirements by a considerable margin. With this low void content, any inadvertent core flow increase would only place operation in a more conservative mode relative to MCPR. During initial start-up testing of the plant, a MCPR evaluation will be made at 25% thermal power level with minimum recirculation pump speed. The MCPR margin will thus be demonstrated such that future MCPR evaluation below this power level will be shown to be unnecessary. The daily requirement for calculating MCPR above 25% rated thermal power is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR when a limiting control rod pattern is approached ensures that MCPR will be known following a change in power or power shape (regardless of magnitude) that could place operation at a thermal limit.

REFERENCES

1. General Electric BWR Generic Reload Fuel Application, NEDE-24011-P.

2. Letter from J. E. Howard, Boston Edison Company to D. L. Ziemann USNRC, dated October 31, 1975.

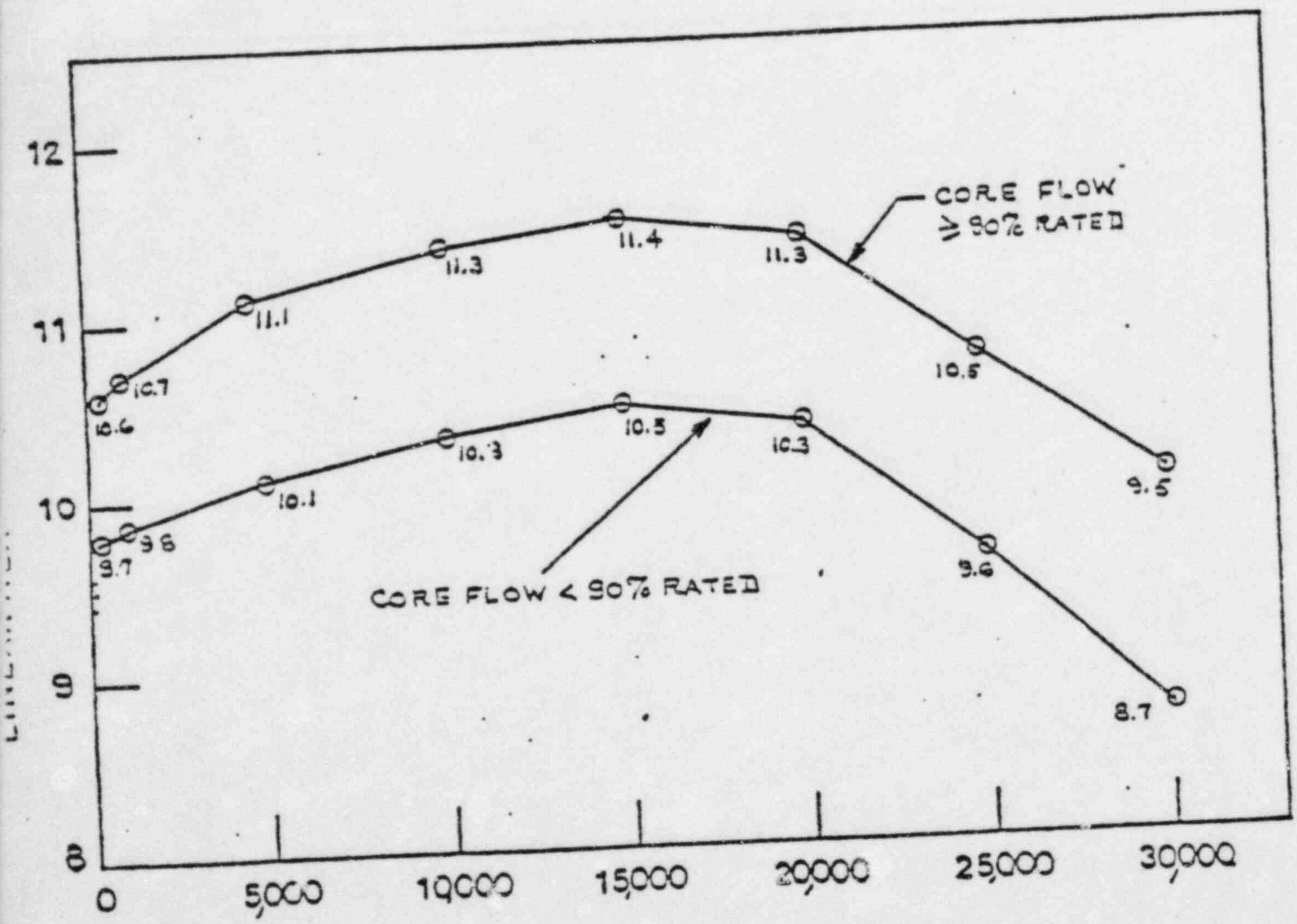
FIGURE 3.11-1

MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE

VERSUS

PLANAR AVERAGE EXPOSURE

FUEL TYPE 8DB219L



*For two recirculation loop
Reduction factors for one recirculation loop were derived in "Pilgrim Single-Loop Operation", NEDO 24268, June 1980.

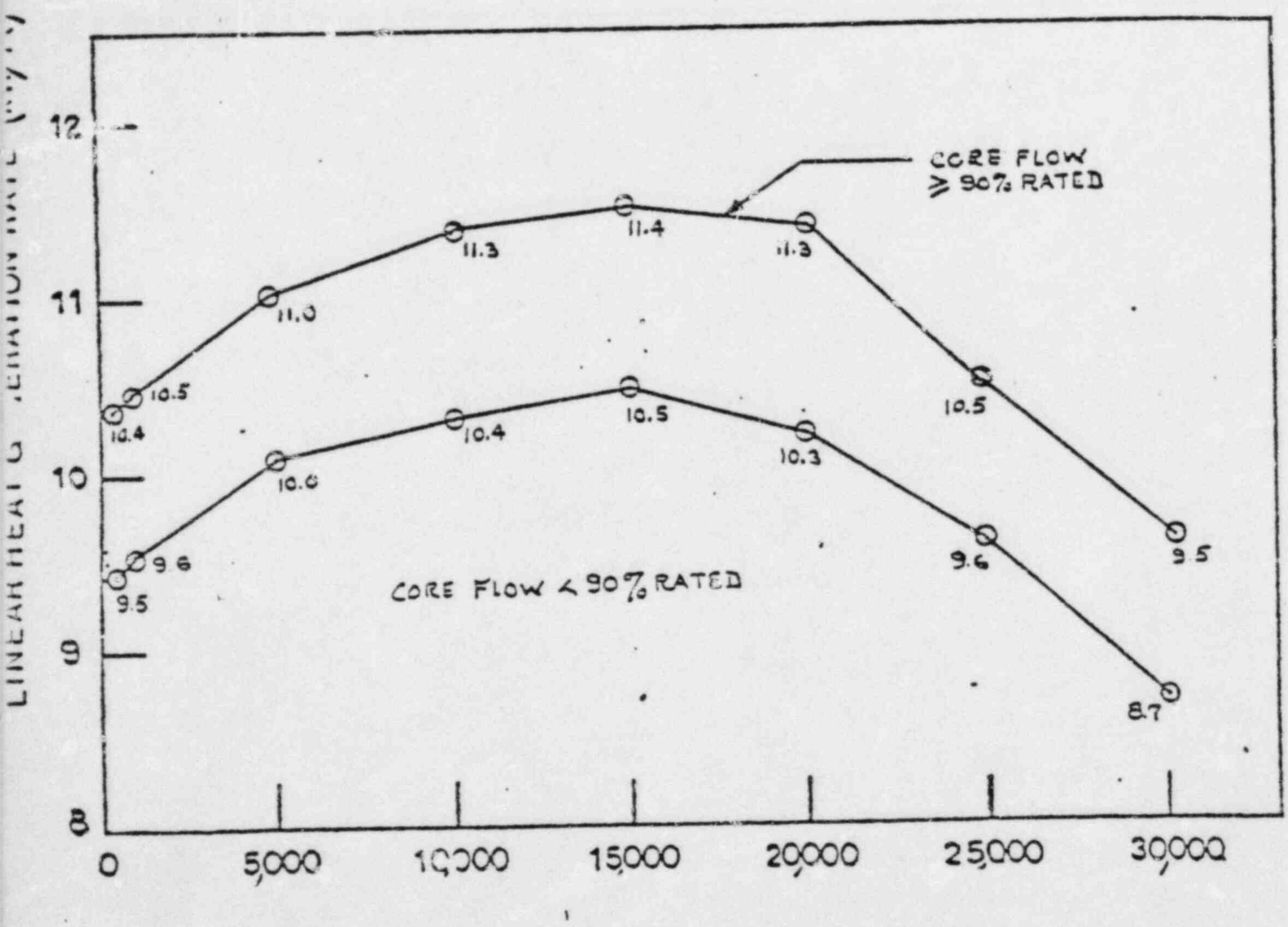
FIGURE 3.11-2

MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE

VERSUS

PLANAR AVERAGE EXPOSURE

FUEL TYPE 8DB219H



*For two recirculation loop
Reduction factors for one recirculation loop werederived in "Pilgrim Single-Loop Operation", NEDO 24268, June 1980.

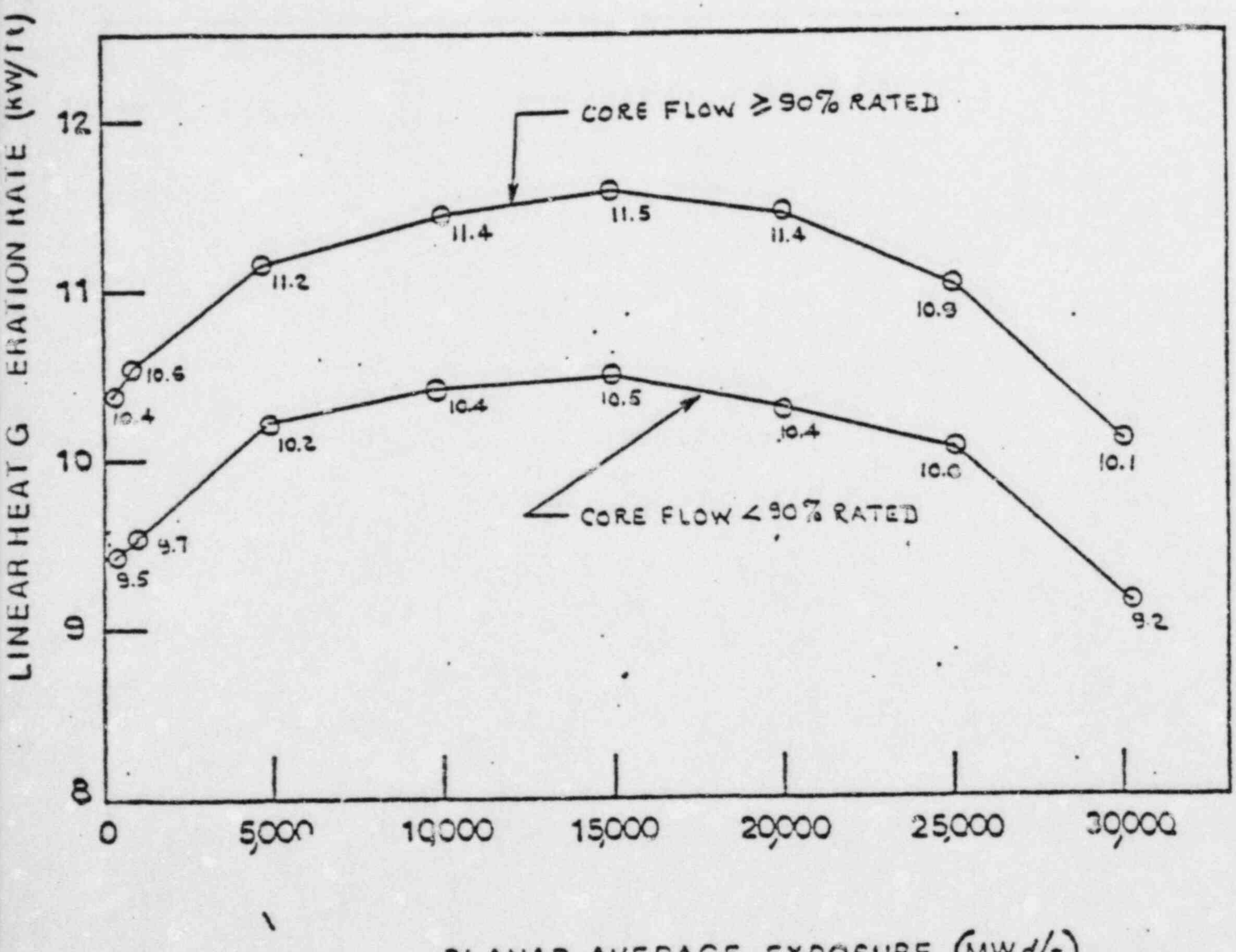
FIGURE 3.11-3

MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE

VERSUS

PLANAR AVERAGE EXPOSURE

FUEL TYPE 8DB262



*For two recirculation loop
Reduction factors for one recirculation loop were derived in "Pilgrim Single-Loop Operation", NEDO 24268, June 1980.

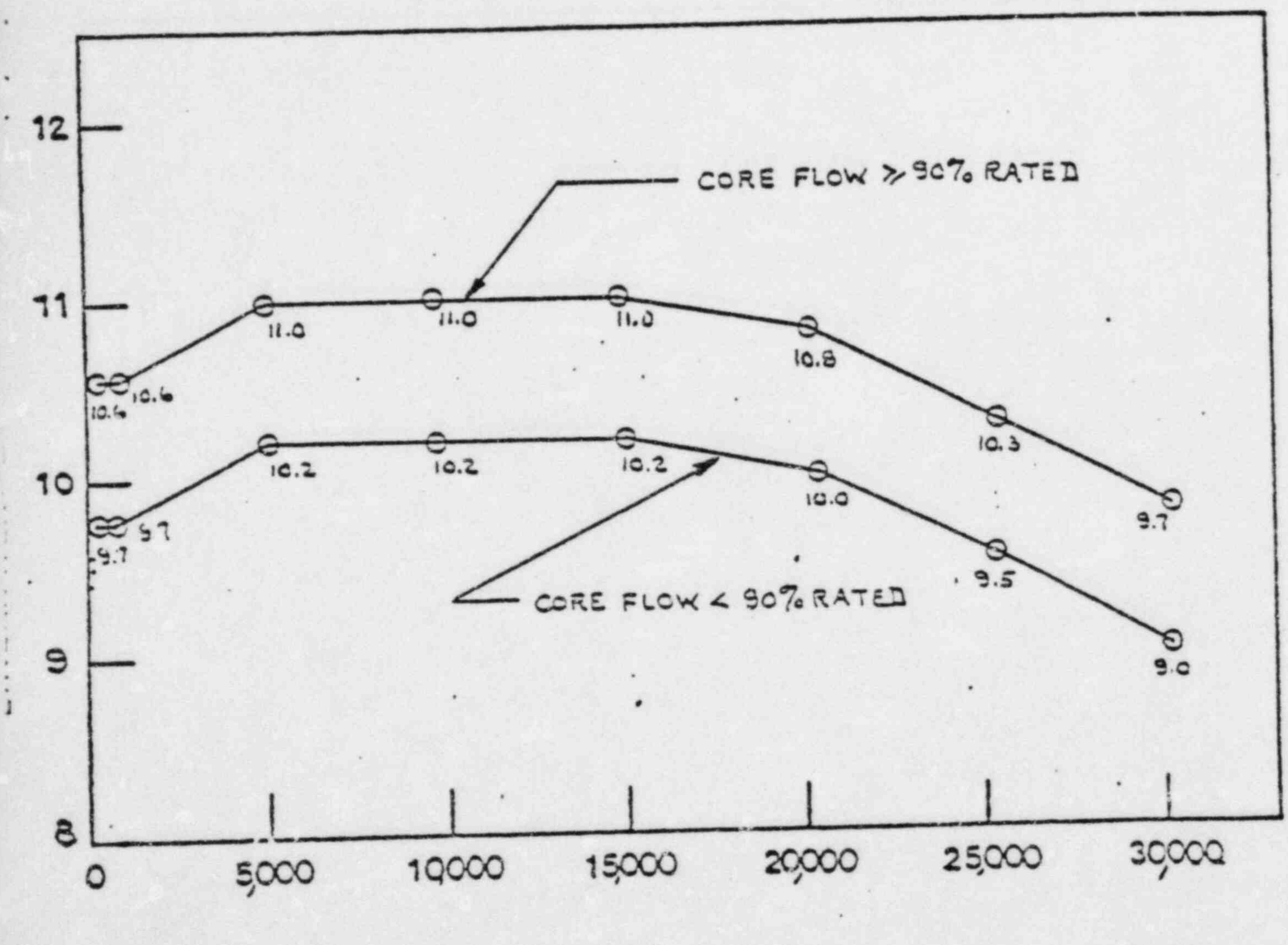
FIGURE 3.11-4

MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE

VERSUS

PLANAR AVERAGE EXPOSURE

FUEL TYPE P8DRB265L



*For two recirculation loop
Reduction factors for one recirculation loop were derived in "Pilgrim Single-Loop Operation", NEDO 24268, June 1980.

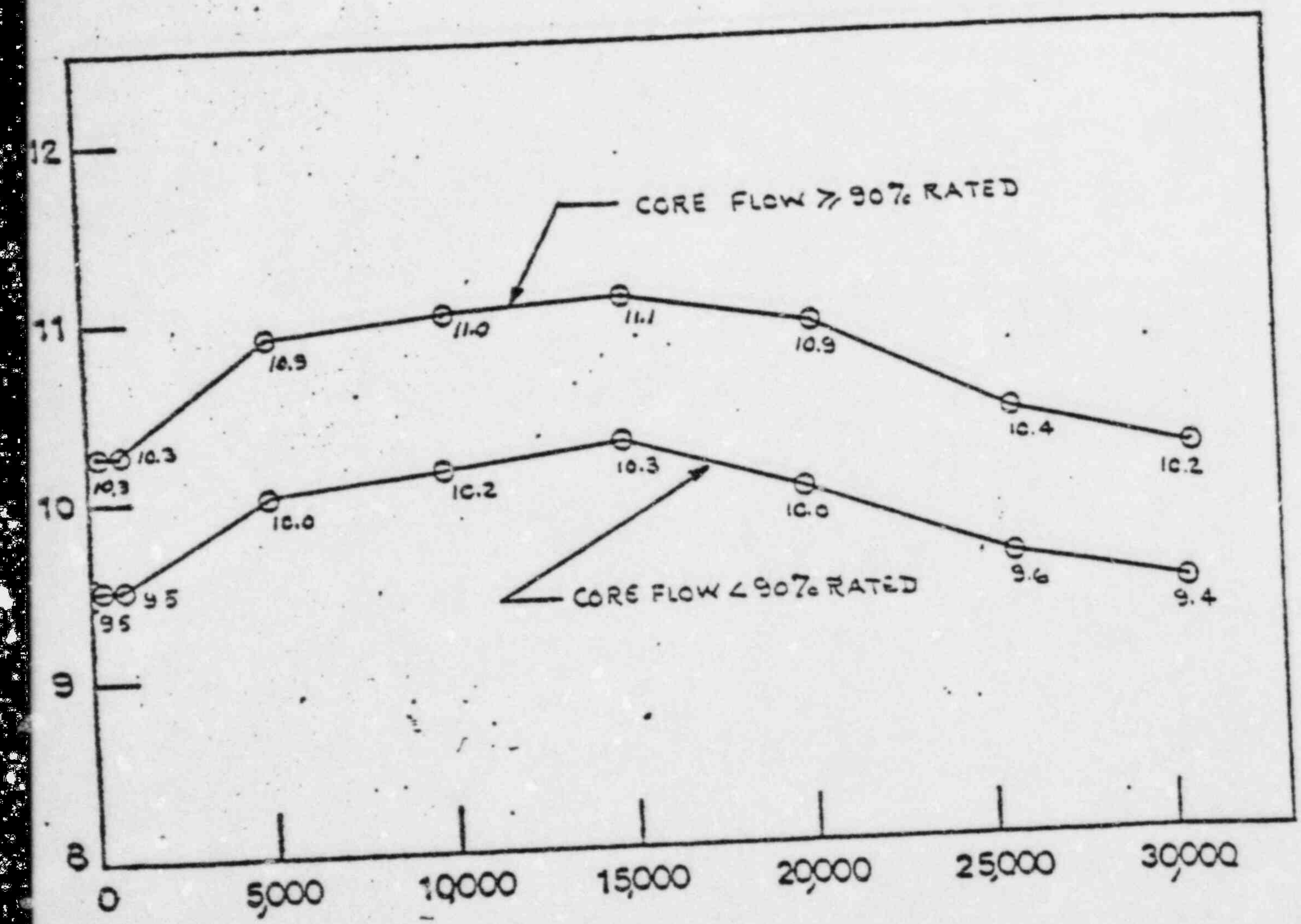
FIGURE 3.11-5

MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE

VERSUS

PLANAR AVERAGE EXPOSURE

FUEL TYPE P8DRB282



*For two recirculation loop
Reduction factors for one recirculation loop were derived in "Pilgrim Single-Loop Operation, NEDO 24268, June 1980