MAY 21 1976

Docket No. 50-293

Boston Edison Company
M/C NUCLEAR
ATTN: Mr. J. E. Larson
Nuclear Licensing
Administrator - Operations
800 Boylston Street
Boston, Massachusetts 02199

Gentlemen:

In response to your requests dated July 9 and July 29 and supplements thereto dated October 3, October 31, November 10, November 17, December 8, 1975, and March 1, March 19, and April 12, 1976, the Commission has issued the enclosed Amendment No. 15 to Facility Operating License No. DPR-35 for the Pilgrim Nuclear Power Station Unit No. 1.

This amendment (1) authorizes operation with additional 8 x 8 fuel assemblies, (2) establishes operating limits based upon the General Electric Thermal Analysis Basis (GETAB), and (3) incorporates operating limits in the Technical Specifications for the facility based on an acceptable evaluation model that conforms with the requirements of Section 50.46 of 10 CFR Part 50.

You have indicated that it will be several months before the additional information, requested by the staff on February 13, 1976 in support of your proposed changes to the APRM scram and rod block settings, will be made available by General Electric. Accordingly, our review of those changes cannot be completed at this time and we will defer action on those changes until the requested additional information is received and evaluated.

Additionally, we have determined that we do not have sufficient information to complete our review of your request relating to single loop operation. Accordingly, we have added a restriction to your license which prohibits operation of the reactor for more than 24 hours with one recirculation loop out of service. A request for the additional information that we require will be provided under separate cover.

OFFICE ➤						
SURNAME ➤						
DATE ➤						

MAY 21 1976

Copies of the related Safety Evaluation, the Negative Declaration, Environmental Impact Appraisal, and the Federal Register Notice are also enclosed.

Sincerely,

Original signed by
Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors

Enclosures:
1. Amendment No. 15 to License No. DPR-35
2. Safety Evaluation
3. Negative Declaration with Supporting Environmental Impact Appraisal
4. Federal Register Notice

cc w/enclosures:
See next page

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SURNAME	PWO'Connor:ah	RMDiggs		DLZiemann	KRGoller	
DATE	5/21/76	5/ /76	5/ /76	5/21/76	5/21/76	

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Docket No. 50-293

Boston Edison Company
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ATTN: Mr. J. E. Larson,
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Administrator - Operations
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This amendment (1) authorizes operation with additional 8 x 8 fuel assemblies, (2) establishes operating limits based upon the General Electric Thermal Analysis Basis (GETAB) and, (3) incorporates operating limits in the Technical Specifications for the facility based on an acceptable evaluation model that conforms with the requirements of Section 50.46 of 10 CFR Part 50.

You have indicated that it will be several months before the additional information, requested by the staff on February 13, 1976 in support of your proposed changes to the APRM scram and rod block settings, will be made available by General Electric. Accordingly, our review of those changes cannot be completed at this time and we will defer action on those changes until the requested additional information is received and evaluated.

Copies of the related Safety Evaluation, the Negative Declaration, Environmental Impact Appraisal, and the Federal Register Notice are also enclosed.

Sincerely,

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors

subject to memo [illegible] 5/20/76.

OFFICE➤	OR:ORB #2	OR:ORB #2	OELD	OR:ORB #2	OR:AD/OR	
SURNAME➤	RMDiggs	PO'Connor:	TOURTELLOTT	DLZiemann	KRGoller	
DATE➤	5/20/76	5/17/76	5/20/76	5/ /76	5/ /76	

cc w/enclosures:
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Pilgrim Station Manager
Boston Edison Company
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Roisman, Kessler and Cashdan
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Henry Herrmann, Esquire
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151 Tremont Street
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Plymouth Public Library
North Street
Plymouth, Massachusetts 02360

Mr. David F. Tarantino
Chairman, Board of Selectmen
11 Lincoln Street
Plymouth, Massachusetts 02360

cc w/enclosures and cy of BECo
filings of 7/29, 10/3 & 31,
11/10 & 17, 12/8/75; 3/1 & 19/76
and 4/12/76:
Henry Kolbe, M. D.
Acting Commissioner of Public Health
Massachusetts Department of
Public Health
600 Washington Street
Boston, Massachusetts 02202

BOSTON EDISON COMPANY

DOCKET NO. 50-293

PILGRIM NUCLEAR POWER STATION UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 15
License No. DPR-35

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The applications for amendment by the Boston Edison Company (the licensee) dated July 9 and 29, 1975, and supplements thereto dated October 3, October 31, November 10, November 17, December 8, 1975 and March 1, March 19 and April 12, 1976, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. After weighing the environmental aspects involved, the issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraphs 3.D, 3.E, and 3.F of Facility License No. DPR-35 are hereby added to read as follows:

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"D. Restrictions

Beyond a cycle exposure of 2600 MWd/t, the reactor power level shall be limited by the maximum power profile shown on figure 7-11 of NEDO 20855-01 "General Electric BWR Reload No. 2 Licensing Submittal for Pilgrim Nuclear Power Station Unit 1 with Bypass Flow Holes Plugged". This limit will be administered by small incremental power reductions. For each reduction, a flow control line shall be interpolated between the nominal 100% and 92% flow control line shown on figure 5.1 of Boston Edison's letter dated March 19, 1976, from C. G. Andognini to D. L. Ziemann, USNRC. Operation of the reactor shall not exceed the power vs. flow conditions defined by the appropriate flow control line.

E. Equalizer Valve Restriction

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

F. Recirculation Loop Inoperable

The reactor shall not be operated with one recirculation loop out of service for more than 24 hours. With the reactor operating, if one recirculation loop is out of service, the plant shall be placed in a hot shutdown condition within 24 hours unless the loop is sooner returned to service.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by:
Karl R. Goller
Karl R. Goller, Assistant Director
for Operating Reactors
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: MAY 21 1976

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ATTACHMENT TO LICENSE AMENDMENT NO. 15

FACILITY OPERATING LICENSE NO. DPR-35

DOCKET NO. 50-293

The following changes relate to the Appendix B portion of the Pilgrim Technical Specifications. The changed areas on the revised pages are shown by marginal lines.

Remove Pages	Insert Pages
(i)	(i)
(ii)	(ii)
4	4
6 through 13	6 through 13
	13a and 13b
14 through 21	14 through 21
24	24
27	27
29	29
54 and 55	54 and 55
71	71
83	83
91	91
	100
110	110
113	113
118	118
122A, 122B	
145	145
	205A through 205E

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1. At least one door in each access opening is closed.

2. The standby gas treatment system is operable.

3. All automatic ventilation system isolation valves are operable or secured in the isolated position.

O. Operating Cycle - Interval between the end of one refueling outage and the end of the next subsequent refueling outage.

P. Refueling Outage - Refueling outage is the period of time between the shutdown of the unit prior to a refueling and the startup of the plant after that refueling. For the purpose of designating frequency of testing and surveillance, a refueling outage shall mean a regularly scheduled outage; however, where such outages occur within 8 months of the completion of the previous refueling outage, the required surveillance testing need not be performed until the next regularly scheduled outage.

Q. Alteration of the Reactor Core - The act of moving any component in the region above the core support plate, below the upper grid and within the shroud. Normal control rod movement with the control rod drive hydraulic system is not defined as a core alteration. Normal movement of in-core instrumentation is not defined as a core alteration.

R. Reactor Vessel Pressure - Unless otherwise indicated, reactor vessel pressures listed in the Technical Specifications are those measured by the reactor vessel steam space detectors.

S. Thermal Parameters

1. Minimum Critical Power Ratio (MCPR) - the value of critical power ratio associated with the most limiting assembly in the reactor core. Critical Power Ratio (CPR) is the ratio of that power in a fuel assembly, which is calculated to cause some point in the assembly to experience boiling transition, to the actual assembly operating power.

2. Transition Boiling - Transition boiling means the boiling regime between nucleate and film boiling. Transition boiling is the regime in which both nucleate and film boiling occur intermittently with neither type being completely stable.

3. Total Peaking Factor - The ratio of the fuel rod surface heat flux to the heat flux of an average rod in an identical geometry fuel assembly operating at the core average bundle power.

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 existence of a minimum critical power ratio (MCPR) less than 1.06 shall constitute violation of the fuel cladding integrity safety limit.

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

When the reactor pressure is $\leq$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:

The limiting safety system settings shall be as specified below:

A. Neutron Flux Scram

1. APRM - The APRM scram trip set point shall be as shown on Fig. 2.1.1 and shall be:

$$S \leq (0.65W + 55\%) \left[\frac{A}{MTPF}\right] \quad \text{2 loop}$$

with a maximum set point of 120% for core flow equal to 69 million lb/hr and greater.

Where:

S = Setting in percent of design power (1998 MWt)

W = Percent of drive flow required to produce a rated core flow of 69 Mlb/hr

A = 3.07 for 7x7 fuel
3.02 for 8x8 fuel

MTPF = The value of the existing maximum total peaking factor

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

MTPF shall be set equal to A unless the actual operating MTPF is greater than the design value of A, in which case, the value for the actual operating MTPF will be used.

2. APRM (15%) - When the reactor mode switch is in the refuel or startup/hot standby position, the APRM scram shall be set at $\leq$ 15% of design power.

3. IRM - The IRM scram shall be set at $\leq$ 120/125 of full scale.

B. APRM Rod Block

The APRM Control Rod Block trip set point(s) shall be biased with flow as shown on Fig. 2.1.1 and shall be less than or equal to:

$S \leq (0.65W + 42\%) \left[\frac{A}{MTPF}\right]$ 2 loops

The definitions used above for the APRM scram trip apply.

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.

H. Main steam isolation on main steam line low pressure at inlet to turbine valves. Pressure setting shall be $\geq$ 880 psig.

I. Reactor low-low water level initiation of CSCS systems setting shall be at or above -49 in. indicated level.

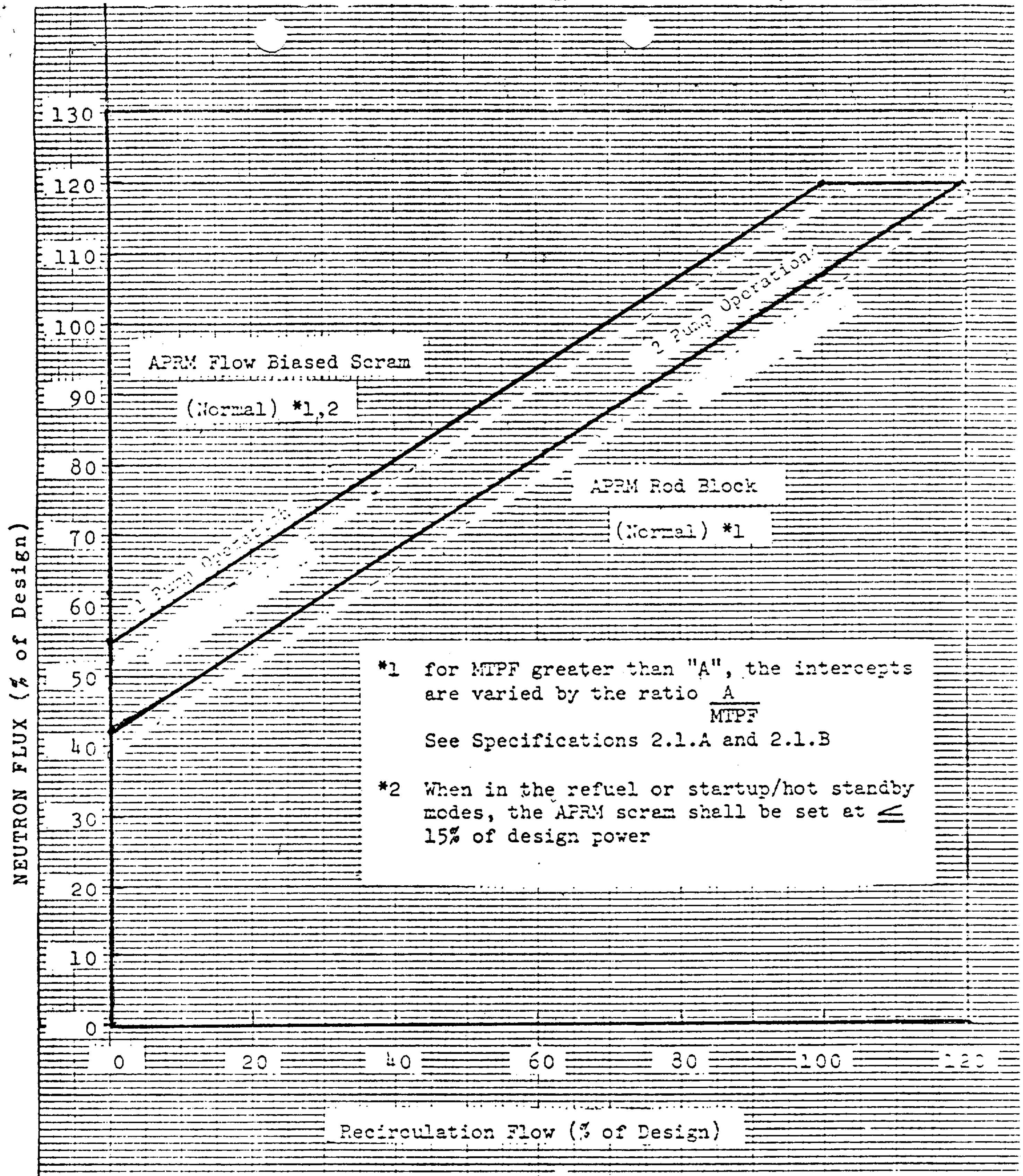

Figure 2.1.1

APRM Scram and Rod Block Trip Limiting Safety System Settings

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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 determined using the General Electric Thermal Analysis Basis, GETAB [1], which is a statistical model that combines all of the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the General Electric Critical Quality (X) - Boiling Length (L), GEXL, correlation.

The GEXL correlation is valid over the range of conditions used in the tests of the data used to develop the correlation. These conditions are:

Pressure:	800 to 1400 psig	
Max flux:	0.1 to 1.25×10^6 lb/hr-ft^2	
Inlet Subcooling:	0 to 100 Btu/lb	
Local Peaking:	1.61 at a corner rod to 1.47 at an interior rod	
Axial Peaking:	Shape	Max/Avg.
	Uniform	1.0
	Outlet Peaked	1.60
	Inlet Peaked	1.60
	Double Peak	1.46 and 1.38
	Cosine	1.39
Rod Array	16, 64 Rods in an 8x8 array 49 Rods in a 7x7 array	

The required input to the statistical model are the uncertainties listed on Table 1.1-1, the nominal values of the core parameters listed in Table 1.1-2 and the relative assembly power distribution shown in Table 1.1-3. Table 1.1-4 shows the R-factor distributions that are input to the statistical model which is used to establish the safety limit MCPR. The R-factor distributions shown are taken near the beginning of the fuel cycle.

The basis for the uncertainties in the core parameters are given in NEDO 20340 [2] and the basis for the uncertainty in the GEXL correlation is given in NEDO-10958[1]. The power distribution is based on a typical 764 assembly core in which the rod pattern was arbitrarily chosen to produce a skewed power distribution having the greatest number of assemblies at the highest power levels. The worst distribution in Pilgrim Nuclear Power Station Unit 1 during any fuel cycle would not be as severe as the distribution used in the analysis.

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

The use of the GEXL correlation is not valid for the critical power calculations at pressures below 800 psig or core flows less than 10% of rated. Therefore, the fuel cladding integrity safety limit is established by other means. This is done by establishing a limiting condition of core thermal power operation with the following basis.

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 28×10^3 lbs/hr bundle flow, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.56 psi driving head will be greater than 28×10^3 lbs/hr 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 approxiamtely 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.

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. Establishement 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 Thermal Analysis Basis (GETAB): Data, Correlation and Design Application, General Electric Co. BWR Systems Department, November 1973 (NEDO-10958).

2. Process Computer Performance Evaluation Accuracy, General Electric Company BWR Systems Department, June, 1974 (NEDO-20340).

Table 1.1-1

UNCERTAINTIES USED IN THE DETERMINATION

OF THE FUEL CLADDING SAFETY LIMIT

Quantity	Standard Deviation (% of Point)
Feedwater Flow	1.76
Feedwater Temperature	0.76
Reactor Pressure	0.5
Core Inlet Temperature	0.2
Core Total Flow	2.5
Channel Flow Area	3.0
Friction Factor Multiplier	10.0
Channel Friction Factor Multiplier	5.0
TIP Readings	8.7
Bypass void effect on TIP	4.46% (core midplane) 5.40% (core exit)
R Factor	1.6
Critical Power	3.6

Table 1.1-2

NOMINAL VALUES OF PARAMETERS USED IN

THE STATISTICAL ANALYSIS OF FUEL CLADDING INTEGRITY SAFETY LIMIT

Core Thermal Power	3293 MW
Core Flow	102.5 Mlb/hr
Dome Pressure	1010.4 psig
Channel Flow Area	0.1078 ft^2
R-factors	1.098 (7x7) 1.100 (8x8)

Table 1.1-3

RELATIVE BUNDLE POWER DISTRIBUTION

USED IN THE GETAB STATISTICAL ANALYSIS

Range of Relative Bundle Power	Percent of Fuel Bundles Within Power Interval
1.375 to 1.425	6.6
1.325 to 1.375	3.2
1.275 to 1.325	15.6
1.225 to 1.275	10.8
1.175 to 1.225	6.6
1.125 to 1.175	4.9
1.075 to 1.125	9.0
1.025 to 1.075	4.0
0.175 to 1.025*	39.3
	Sum = 100

*Relative bundle powers in this interval sum up to 39.3% of the total distribution.

Table 1.1-4

R-FACTOR DISTRIBUTION USED IN GETAB STATISTICAL ANALYSIS

7x7 Rod Array		8x8 Rod Array	
R-Factor	Rod Sequence No.	R-Factor	Rod Sequence No.
1.098	1	1.100	1
1.083	2	1.100	2
1.075	3	1.095	3
1.062	4	1.095	4
1.052	5	1.093	5
1.042	6	1.093	6
1.042	7	1.092	7
$\leq$1.027	8 thru 49	$\leq$ 1.077	8 thru 63

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.

Conservatism is incorporated in the transient analyses in estimating the controlling factors such as void reactivity coefficient, control rod scram worth, scram delay time, peaking factors, and axial power shapes. These factors are selected conservatively with respect to their effect on the applicable transient results as determined by the current analysis model. This transient model, evolved over many years, has been substantiated in operation as a conservative tool for evaluating reactor dynamic performance. Results obtained from a General Electric boiling water reactor have been compared with predictions made by the model. The comparisons and results are summarized in Reference 1.

The absolute value of the void reactivity coefficient used in the analysis is conservatively estimated to be about 25% greater than the nominal maximum value expected to occur during the core lifetime. The scram worth used has been derated to be equivalent to approximately 80% of the total scram worth of the control rods. The scram delay time and rate of rod insertion allowed by the analyses are conservatively set equal to the longest delay and slowest insertion rate acceptable by Technical Specifications. The effect of scram worth, scram delay time and rod insertion rate, all conservatively applied, are of greatest significance in the early portion of the negative reactivity insertion. The rapid insertion of negative reactivity is assured by the time requirements for 10% and 30% insertion. By the time the rods are 60% inserted, approximately four dollars of negative reactivity have been inserted which strongly turns the transient, and accomplishes the desired effect. The times for 50% and 90% insertion are given to assure proper completion of the expected performance in the earlier portion of the transient, and to establish the ultimate fully shutdown steady-state condition.

This choice of using conservative values of controlling parameters and initiating transients at the design power level, produces more pessimistic answers than would result by using expected values of control parameters and analyzing at higher power levels.

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

2.1 BASES:

In summary:

i. The abnormal operational transients were analyzed to a power level of 1998 MWt.

ii. The licensed maximum power level is 1998 MWt.

iii. Analyses of transients employ adequately conservative values of the controlling reactor parameters.

iv. The analytical procedures now used result in a more logical answer than the alternative method of assuming a higher starting power in conjunction with the expected values for the parameters.

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.

The scram trip setting must be adjusted to ensure that the LHGR transient peak is not increased for any combination of MTPF and reactor core thermal power. The scram setting is adjusted in accordance with the formula in Specification 2.1.A.1 when the maximum total peaking factor is greater than 3.07 for 7x7 fuel and 3.02 for 8x8 fuel.

Analyses of the limiting transients show that no scram adjustment is required to assure MCPR > 1.06 when the transient is initiated from MCPR above the normal operating limit.

For operation in the startup mode while the reactor is at low pressure, the APRM scram setting of 15 percent of rated power provides adequate thermal margin between the setpoint and the safety limit, 25 percent of rated. The margin is adequate to accommodate anticipated maneuvers associated with power plant startup. Effects of increasing pressure at zero or low void content are minor, cold water from sources available during startup is not much colder than that already in the system, temperature coefficients are small, and control rod patterns are constrained to be uniform by operating procedures backed up by the rod worth minimizer.

Worth of individual rods is very low in a uniform rod pattern. Thus, of all possible sources of reactivity input, uniform control rod withdrawal is the most probable case of significant power rise. Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks, and because several rods must be moved to change power by a significant percentage of rated power, the rate of power rise is very slow. Generally the heat flux is in the near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the scram level, the rate of power rise is no more than five percent of rated power per minute, and the APRM system would be more than adequate to assure a scram before power could exceed the safety limit. The 15% APRM scram remains active until the mode switch is placed in the RUN position. This switch occurs when reactor pressure is greater than 880 psig.

The analysis to support operation at various power and flow relationships has considered operation with either one or two recirculation pumps.

IRM

The IRM system consists of 8 chambers, 4 in each of the reactor protection system logic channels. The IRM is a 5-decade instrument which covers the range of power level

2.1 BASES:

between that covered by the SRM and the APRM. The 5 decades are covered by the IRM by means of a range switch and the 5 decades are broken down into 10 ranges, each being one-half of a decade in size. The IRM scram setting of 120/125 of full scale is active in each range of the IRM. For example, if the instrument were on range 1, the scram setting would be a 120/125 of full scale for that range; likewise, if the instrument were on range 5, the scram would be 120/125 of full scale on that range. Thus, as the IRM is ranged up to accommodate the increase in power level, the scram setting is also ranged up. The most significant sources of reactivity change during the power increase are due to control rod withdrawal. For in-sequence control rod withdrawal, the rate of change of power is slow enough due to the physical limitation of withdrawing control rods that heat flux is in equilibrium with the neutron flux, and an IRM scram would result in a reactor shutdown well before any safety limit is exceeded.

In order to ensure that the IRM provided adequate protection against the single rod withdrawal error, a range of rod withdrawal accidents was analyzed. This analysis included starting the accident at various power levels. The most severe case involves an initial condition in which the reactor is just subcritical and the IRM system is not yet on scale. This condition exists at quarter rod density. Additional conservatism was taken in this analysis by assuming that the IRM channel closest to the withdrawn rod is bypassed. The results of this analysis show that the reactor is scrammed and peak core power limited to one percent of rated power, thus maintaining MCPR above 1.06. Based on the above analysis, the IRM provides protection against local control rod withdrawal errors and continuous withdrawal of control rods in sequence and provides backup protection for the APRM.

B. APRM Control Rod Block

Reactor power level may be varied by moving control rods or by varying the recirculation flow rate. The APRM system provides a control rod block to prevent rod withdrawal beyond a given point at constant recirculation flow rate, and thus to protect against the condition of a MCPR less than 1.06. This rod block set point, which is automatically varied with recirculation loop flow rate, prevents an increase in the reactor power level to excessive values due to control rod withdrawal. The flow variable trip setting provides substantial margin from fuel damage, assuming a steady-state operation at the trip setting, over the entire recirculation flow range. The margin to the safety limit increases as the flow decreases for the specified trip setting versus flow relationship; therefore, the worst case MCPR which could occur during steady-state operation is at 107% of rated thermal power because of the APRM rod block trip 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 total peaking factor exceeds 3.07 for 7x7 fuel and 3.02 for 8x8 fuel, thus preserving the APRM rod block safety margin.

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 1.06 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 $\leq$ 10 percent of valve closure from full open, the resultant increase in surface heat flux is limited such that MCPR remains above 1.06 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 1.06.

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:

G & H. Main Steam Line Isolation on Low Pressure and Main Steam Line Isolation Scram

The low pressure isolation of the main steam lines at 880 psig was provided to protect against rapid reactor depressurization and the resulting rapid cooldown of the vessel. Advantage is taken of the scram feature that occurs when the main steam line isolation valves are closed, to provide for reactor shutdown so that high power operation at low reactor pressure does not occur; thus providing protection for the fuel cladding integrity safety limit. Operation of the reactor at pressures lower than 800 psig requires that the reactor mode switch be in the STARTUP position, where protection of the fuel cladding integrity safety limit is provided by the IRM high neutron flux scram and APRM 15% scram. Thus, the combination of main steam line low pressure isolation and isolation valve closure scram assures the availability of neutron flux scram protection over the entire range of applicability of the fuel cladding integrity safety limit. In addition, the isolation valve closure scram anticipates the pressure and flux transients that occur during normal or inadvertent isolation valve closure. With the scrams set at 10 percent of valve closure, neutron flux does not increase.

I. Reactor Low-Low Water Level Set Point for Actuation of Core Standby Cooling System

The core standby cooling subsystems are designed to provide sufficient cooling to the core to dissipate the energy associated with the loss-of-coolant accident and to limit fuel clad temperature, to assure that core geometry remains intact and to limit any clad metal-water reaction to less than 1%. To accomplish their intended function, the capacity of each Core Standby Cooling System component was established based on the reactor low water level scram set point. To lower the set point of the low water level scram would increase the capacity requirement for each of the CSCS components. Thus, the reactor vessel low water level scram was set low enough to permit margin for operation, yet will not be set lower because of CSCS capacity requirements.

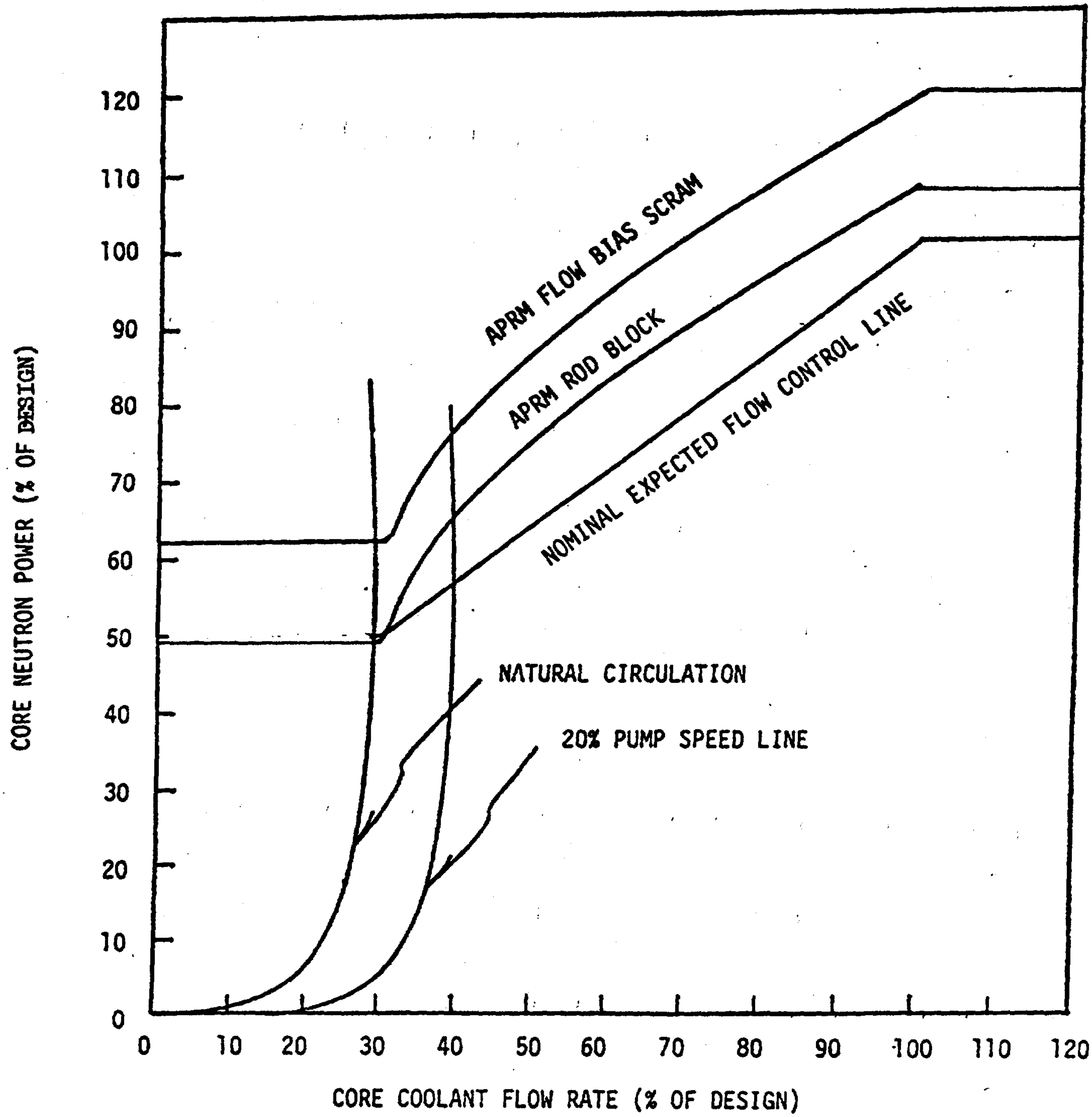
The design of the CSCS components to meet the above guidelines was dependent upon three previously set parameters: the maximum break size, low water level scram set point and the CSCS initiation set point. To lower the set point for initiation of the CSCS may lead to a decrease in effective core cooling. To raise the CSCS initiation set point would be in a safe direction, but it would reduce the margin established to prevent actuation of the CSCS during normal operation or during normally expected transients.

2.1 BASES:

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

References

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



APRM FLOW BIAS SCRAM VERSUS REACTOR CORE FLOW
FIG. 2.1.3

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

The valve sizing analysis considered four, 10% capacity relief/safety valves and two 8% capacity safety valves. These are sized and set pressures are established in accordance with the following three requirements of Section III of the ASME Code:

1. The lowest safety valve must be set to open at or below vessel design pressure and the highest safety valve be set at or below 105% of design pressure.

2. The valves must limit the reactor pressure to no more than 110% of design pressure.

3. Protection systems directly related to the valve sizing transient must not be credited with action (i.e., an indirect scram must be assumed).

A main steam line isolation with flux scram has been selected to be used as the safety valve sizing transient since this transient results in the highest peak vessel pressure of any transient when analyzed with an indirect scram. The original FSAR analysis concluded that the peak pressure transient with indirect scram would be caused by a loss of condenser vacuum (turbine trip with failure of the bypass valves to open). However, later observations have shown that the long lengths of steam lines to the turbine buffer the faster stop valve closure isolation and thereby reduce the peak pressure caused by this transient to a value below that produced by a main steam line isolation with flux scram.

Item 3 above indicates that no credit be taken for the primary scram signal generated by closure of the main steam isolation valves. Two other scram initiation signals would be generated, one due to high neutron flux and one due to high reactor pressure. Thus item 3 will be satisfied by assuming a scram due to high neutron flux.

Relieving capacity of 40% (4 relief/safety valves) results in a peak pressure during the transient conditions used in the safety valve sizing analysis which is well below the pressure safety limit.

Table 3..

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENT

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

*APRM high flux scram setpoint $\leq (.65W + 55) \frac{A}{MTPF}$ Two recirc. pump operation

<u>NOTES FOR TABLE 3.[illegible] (Cont'd)</u>

10. Not required to be operable when the reactor pressure vessel head is not bolted to the vessel.

11. The APRM downscale trip function is only active when the reactor mode switch is in run.

12. The APRM downscale trip is automatically bypassed when the IRM instrumentation is operable and not high.

13. An APRM will be considered inoperable if there are less than 2 LPRM inputs per level or there is less than 50% of the normal complement of LPRM's to an APRM.

14. W is percent of drive flow required to produce a rated core flow of 69 Mlb/hr. Trip level setting in percent of design power (1998 MWt).

15. See Section 2.1.A.1.

16. The APRM (15%) high flux scram is bypassed when in the run mode.

17. The APRM flow biased high flux scram is bypassed when in the refuel or startup/hot standby modes.

18. A = 3.07 for 7x7 fuel
  = 3.02 for 8x8 fuel

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)	$(0.65W + 42)\left[\frac{A}{MTPF}\right]$	(2)
2	APRM Downscale	2.5 indicated on scale	
1 (7)	Rod Block Monitor (Flow Biased)	(0.58W + 42) ** (0.58W + 39) *	(2)
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.	

NOTES FOR TABLE 3.2.C

1. For the startup and run positions of the Reactor Mode Selector Switch, there shall be two operable or tripped trip systems for each function. The SRM and IRM blocks need not be operable in "Run" mode, and the APRM and RBM rod blocks need not be operable in "Startup" mode. If the first column cannot be met for one of the two trip systems, this condition may exist for up to seven days provided that during that time the operable system is functionally tested immediately and daily thereafter; if this condition lasts longer than seven days, the system shall be tripped. If the first column cannot be met for both trip systems, the systems shall be tripped.

2. W is percent of drive flow required to produce a rated core flow of 69 Mlb/hr. Trip level setting is in percent of design power (1998 MWt).

   A = 3.07 for 7x7 fuel
   = 3.02 for 8x8 fuel

3. IRM downscale is bypassed when it is on its lowest range.

4. This function is bypassed when the count rate is $\geq$ 100 cps.

5. One of the four SRM inputs may be bypassed.

6. This SRM function is bypassed when the IRM range switches are on range 8 or above.

7. The trip is bypassed when the reactor power is $\leq$ 30%.

8. This function is bypassed when the mode switch is placed in Run.

The control rod block functions are provided to prevent excessive control rod withdrawal so that MCPR does not decrease to 1.06. The trip logic for this function is 1 out of n: e.g., any trip on one of six APRM's, eight IRM's, or four SRM's will result in a rod block.

The minimum instrument channel requirements assure sufficient instrumentation to assure the single failure criteria is met. The minimum instrument channel requirements for the RBM may be reduced by one for maintenance, testing, or calibration. This time period is only 3% of the operating time in a month and does not significantly increase the risk of preventing an inadvertent control rod withdrawal.

The APRM rod block function is flow biased and prevents a significant reduction in MCPR, especially during operation at reduced flow. The APRM provides gross core protection; i.e., limits the gross core power increase from withdrawal of control rods in the normal withdrawal sequence. The trips are set so that MCPR is maintained greater than 1.06.

The RBM rod block function provides local protection of the core, for a single rod withdrawal error from a limiting control rod pattern.

The IRM rod block function provides local as well as gross core protection. The scaling arrangement is such that trip setting is less than a factor of 10 above the indicated level.

A downscale indication on an APRM or IRM is an indication the instrument has failed or the instrument is not sensitive enough. In either case the instrument will not respond to changes in control rod motion and thus, control rod motion is prevented. The downscale trips are set at 2.5 indicated on scale.

The flow comparator and scram discharge volume high level components have only one logic channel and are not required for safety.

The refueling interlocks also operate one logic channel, and are required for safety only when the mode switch is in the refueling position.

For effective emergency core cooling for small pipe breaks, the HPCI system must function since reactor pressure does not decrease rapidly enough to allow either core spray or LPCI to operate in time. The automatic pressure relief function is provided as a backup to the

3.3.2 Control Rods

4. Control rods shall not be withdrawn for startup or refueling unless at least two source range channels have an observed count rate equal to or greater than three counts per second.

5. During operation with limiting control rod patterns, as determined by the Reactor Engineer, either:

   a. Both RBM channels shall be operable: or

   b. Control rod withdrawal shall be blocked: or

   c. The operating power level shall be limited so that the MCPR will remain above 1.06 assuming a single error that results in complete withdrawal of any single operable control rod.

C. Scram Insertion Times

1. The average scram insertion time, based on the de-energization of the scram pilot valve solenoids as time zero, of all operable control rods in the reactor power operation condition shall be no greater than:

% Inserted From Fully Withdrawn	Avg. Scram Insertion Times (set)
10	.55
30	1.275
50	2.00
90	3.50

4.3.B Control Rods

4. Prior to control rod withdrawal for startup or during refueling, verify that at least two source range channels have an observed count rate of at least three counts per second.

5. When a limiting control rod pattern exists, an instrument functional test of the RBM shall be performed prior to withdrawal of the designated rod(s) and daily thereafter.

C. Scram Insertion Times

1. Following each refueling outage, each operable control rod shall be subjected to scram time tests from the fully withdrawn position. If testing is not accomplished with the nuclear system pressure above 950 psig, the measured scram insertion time shall be extrapolated to reactor pressures above 950 psig using previously determined correlations. Testing of all operable control rods shall be completed prior to exceeding 40% rated thermal power.

BASES:

During reactor operation with certain limiting control rod patterns, the withdrawal of a designated single control rod could result in one or more fuel rods with MCPR's less than 1.06. 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 withdrawal does not occur. It is the responsibility of the Reactor Engineer 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.

C. Scram Insertion Times

The control rod system is designed to bring the reactor subcritical at a rate fast enough to prevent fuel damage; i.e., to prevent the MCPR from becoming less than 1.06. The limiting power transient is that resulting from a turbine stop valve closure with failure of the turbine bypass system. Analysis of this transient shows that the negative reactivity rates resulting from the scram with the average response of all the drives as given in the above Specification, provide the required protection, and MCPR remains greater than 1.06.

The scram times for all control rods will be determined at the time of each refueling outage. A representative sample of control rods will be scram tested during each cycle as a periodic check against deterioration of the control rod performance.

3.4 STANDBY LIQUID CONTROL SYSTEM

A. The conditions under which the Standby Liquid Control System must provide shutdown capability are identified via the Station Nuclear Safety Operational Analysis (Appendix G). The requirements of this specification are taken from the Operational Nuclear Safety Requirements of subsection 3.8.6 of the Final Safety Analysis Report. If no more than one operable control rod is withdrawn, the basic shutdown reactivity requirement for the core is satisfied and the Standby Liquid Control system is not required. Thus, the basic reactivity requirement for the core is the primary determinant of when the liquid control system is required.

The purpose of the liquid control system is to provide the capability of bringing the reactor from full power to a cold, xenon-free shutdown condition assuming that none of the withdrawn control rods can be inserted. To meet this objective, the liquid control system is designed to inject a quantity of boron that produces a concentration of 700 ppm of boron in the reactor core in less than 125 minutes. The 700 ppm concentration in the reactor core is required to bring the reactor from full power to a three percent Δk subcritical condition, considering the hot to cold reactivity difference, xenon poisoning, etc. The time requirement for inserting the boron solution was selected to override the rate of reactivity insertion caused by cooldown of the reactor following the xenon poison peak.

The minimum limitation on the relief valve setting is intended to prevent the loss of liquid control solution via the lifting of a relief valve at too low a pressure. The upper limit on the relief valve settings provides system protection from overpressure.

B. Only one of the two standby liquid control pumping loops is needed for operating the system. One inoperable pumping circuit does not immediately threaten shutdown capability, and reactor operation can continue while the circuit is being repaired. Assurance that the

3.5.E Automatic Depressurization System (ADS) (Cont'd)

2. From and after the date that one valve in the automatic depressurization subsystem is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding seven days unless such valve is sooner made operable, provided that during such seven days the HPCI subsystem is operable.

3. If the requirements of 3.5.E cannot be met, an orderly shutdown shall be initiated and the reactor pressure shall be reduced to at least 104 psig within 24 hours.

4.5.E Automatic Depressurization System (ADS) (Cont'd)

2. When it is determined that one valve of the ADS is inoperable, the ADS subsystem actuation logic for the other ADS valves and the HPCI subsystem shall be demonstrated to be operable immediately and at least weekly thereafter until the valve is repaired.

3.5.F Minimum Low Pressure Cooling and Diesel Generator Availability

1. During any period when one diesel generator is inoperable, continued reactor operation is permissible only during the succeeding seven days unless such diesel generator is sooner made operable, provided that all of the low pressure core and containment cooling subsystems and the remaining diesel generator shall be operable. If this requirement cannot be met, an orderly shutdown shall be initiated and the reactor shall be placed in the Cold Shutdown Condition within 24 hours.

2. Any combination of inoperable components in the core and containment cooling systems shall not defeat the capability of the remaining operable components to fulfill the cooling functions.

4.5.F Minimum Low Pressure Cooling and Diesel Generator Availability

1. When it is determined that one diesel generator is inoperable, all low pressure core cooling and containment cooling subsystems shall be demonstrated to be operable immediately and daily thereafter. In addition, the operable diesel generator shall be demonstrated to be operable immediately and daily thereafter until the inoperable diesel is repaired.

BASES:

3.5.A Core Spray and LPCI Subsystem

This specification assures that adequate emergency cooling capability is available whenever irradiated fuel is in the reactor vessel.

Based on the loss of coolant analysis performed by General Electric in accordance with Section 50.46 and Appendix K of 10CFR50, the Pilgrim I Emergency Core Cooling Systems are adequate to provide sufficient cooling to the core to dissipate the energy associated with the loss of coolant accident, to limit calculated fuel clad temperature to less than 2200°F, to limit calculated local metal water reaction to less than or equal to 17%, and to limit calculated core wide metal water reaction to less than or equal to 1%.

The limiting conditions of operation in Specifications 3.5.A.1 through 3.5.A.6 specify the combinations of operable subsystems to assure the availability of the minimum cooling systems noted above. No single failure of CSCS equipment occurring during a loss-of-coolant accident under these limiting conditions of operation will result in inadequate cooling of the reactor core.

Core spray distribution has been shown, in full-scale tests of systems similar in design to that of Pilgrim, to exceed the minimum requirements by at least 25%. In addition, cooling effectiveness has been demonstrated at less than half the rated flow in simulated fuel assemblies with heater rods to duplicate the decay heat characteristics of irradiated fuel. The accident analysis is additionally conservative in that no credit is taken for spray coolant entering the reactor before the internal pressure has fallen to 104 psig.

The LPCI subsystem is designed to provide emergency cooling to the core by flooding in the event of a loss-of-coolant accident. This system functions in combination with the core spray system to prevent excessive fuel clad temperature. The LPCI subsystem and the core spray subsystem provide adequate cooling for break areas of approximately 0.2 square feet up to and including the double-ended recirculation line break without assistance from the high pressure emergency core cooling subsystems.

The allowable repair times are established so that the average risk rate for repair would be no greater than the basic risk rate. The method and concept are described in reference (1). Using the results developed in

BASES:

3.5.E Automatic Depressurization System (ADS)

The limiting conditions for operating the ADS are derived from the Station Nuclear Operational Analysis (Appendix G) and a detailed functional analysis of the ADS (Section 6).

This specification ensures the operability of the ADS under all conditions for which the automatic or manual depressurization of the nuclear system is an essential response to station abnormalities.

The nuclear system pressure relief system provides automatic nuclear system depressurization for small breaks in the nuclear system so that the low pressure coolant injection (LPCI) and the core spray subsystems can operate to protect the fuel barrier.

Because the Automatic Depressurization System does not provide makeup to the reactor primary vessel, no credit is taken for the steam cooling of the core caused by the system actuation to provide further conservatism to the CSCS. Performance analysis of the Automatic Depressurization System is considered only with respect to its depressurizing effect in conjunction with LPCI or Core Spray. There are four valves provided and each has a capacity of 800,000 lb/hr at a reactor pressure of 1125 psig.

The allowable out of service time for one ADS valve is determined as seven days because of the redundancy and because the HPCIS is demonstrated to be operable during this period. Therefore, redundant protection for the core with a small break in the nuclear system is still available.

The ADS test circuit permits continued surveillance on the operable relief valves to assure that they will be available if required.

BASES:

3.6.D and 4.6.D

Safety and Relief Valves

As discussed in Subsection 4.4.6 of the Final Safety Analysis Report, design of the nuclear system pressure relief system is intended to protect the nuclear system from overpressurization in the event of the safety valve sizing transient. An indirect scram is assumed because ASME Boiler and Pressure Vessel Code, Section III, requires that protection systems directly related to the valve sizing transient must not be credited with action in determining valve relieving capacity. A total of 4 relief/safety valves and 2 safety valves is provided by the design.

Experience in safety valve operation shows that a testing of at least 50% of the safety valves per refueling outage is adequate to detect failures or deterioration. The tolerance value of $\pm 1\%$ is in accordance with Section III of the ASME Boiler and Pressure Vessel Code. An analysis has been performed which shows that with all safety valves set 1% higher, the reactor coolant pressure safety limit of 1375 psig is not exceeded.

The relief/safety valves have two functions; i.e., power relief or self-actuated by high pressure. Power relief is a solenoid actuated function (Automatic Pressure Relief) in which external instrumentation signals of coincident high drywell pressure and low-low water level initiate the valves to open. This function is discussed in Specification 3.5.D. In addition, the valves can be operated manually.

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 3.11-7. 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 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.

4.11 REACTOR FUEL ASSEMBLY

Applicability

The surveillance Requirements apply to the parameters which monitor 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 ≥25% rated thermal power.

B. Linear Heat Generation Rate (LHGR)

During reactor power operation the linear heat generation rate (LHGR) of any rod in any fuel assembly at any axial location shall not exceed the maximum allowable LHGR as calculated by the following equation:

$$LHGR_{max} \leq LHGR_d \left[1 - (\Delta P/P)_{max} (L/LT)\right]$$

$LHGR_d$ = Design LHGR = G kW/ft

$(\Delta P/P)_{max}$ = Maximum power spiking penalty = N

LT = Total core length = 12 feet

L = Axial position above bottom of core

G = 17.5 kW/ft for 7x7 fuel bundles
= 13.4 kW/ft for 8x8 fuel bundles

N = 0.026 for 7x7 fuel bundles
= 0.022 for 8x8 fuel bundles

If at any time during operation it is determined by normal surveillance that the limiting value for LHGR is being exceeded, action shall 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, 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.

B. Linear Heat Generation Rate (LHGR)

The LHGR as a function of core height shall be checked daily during reactor operation at $\geq$25% rated thermal power.

C. Minimum Critical Power Ratio (MCPR)

During power operation MCPR shall be $\geq$ 1.31 for 7 x 7 fuel and $\geq$ 1.39 for 8 x 8 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 $\geq$ 1.31 for 7 x 7 fuel and $\geq$ 1.39 for 8 x 8 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.

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.

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 corrected for densification. 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 Technical Specification APLHGR is this LHGR of the highest powered rod divided by its local peaking factor. The limiting value for APLHGR are shown in Figure 3.11.

The calculational procedure used to establish the APLHGR shown on Figure 3.11 is based on a loss-of-coolant accident analysis. The analysis was performed using General Electric (GE) calculational models which are consistent with the requirements of Appendix K to 10 CFR 50. A complete discussion of each code employed in the analysis is presented in Reference 1. Differences in this analysis are compared to previous analyses performed with Reference 1 are: (1) The analyses assumes a fuel assembly planar power consistent with 102% of the MAPLHGR shown in Figure 3.11; (2) Fission product decay is computed assuming an energy release rate of 200 MEV/Fission; (3) Pool film boiling is assumed after nucleate boiling is lost during the flow stagnation period; (4) The effects of core spray entrainment and counter-current flow limiting as described in Reference 2, are included in the reflooding calculations.

A list of the significant plant input parameters to the loss-of-coolant accident analysis is presented in Table 3.11-1.

Table 3.11-1

SIGNIFICANT INPUT PARAMETERS TO THE
LOSS-OF-COOLANT ACCIDENT ANALYSIS

PLANT PARAMETERS:

Core Thermal Power . . . . 2038 MWt which corresponds to 102% % of licensed core power

Vessel Steam Output . . . . 8.15×10^6 LBm/h which corresponds to 102% % of licensed core power

Vessel Steam Dome Pressure . . . . 1050 psia

Recirculation Line Break Area for Large Breaks (Ft.^{2}) . . . . 4.35 ft^2(DBA), 1.0

Recirculation Line Break Area for Small Breaks (Ft.^{2}) 1.0, 0.10

FUEL PARAMETERS:

FUEL TYPE	FUEL BUNDLE GEOMETRY	PEAK TECHNICAL SPECIFICATION LINEAR HEAT GENERATION RATE (kw/ft)	DESIGN AXIAL PEAKING FACTOR	INITIAL MINIMUM CRITICAL POWER RATIO
Initial Core	7 x 7	17.5	1.5	1.18
Reload 8D262	8 x 8	13.4	1.4	1.18

A more detailed list of input to each model and its source is presented in Section II or Reference 1.

The DBA area includes: the area of the recirculation suction line vessel nozzle (3.56 ft^2), the throat area of 10 jet pumps (.7 ft^2), plus the area of the cleanup line (.08 ft^2).

REFERENCES

1. General Electric Company Analtyical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50, Appendix K, NEDE-20566 (Draft), submitted August 1974.

2. General Electric Refill Reflood Calculation (Supplement to SAFE Code Description) transmitted to USAEC by letter, G. L. Gyorey to V. Stello, Jr., dated December 20, 1974.

3. Pilgrim Nuclear Power Station Unit 1 License Amendment for Single-Loop Operation, NEDO-20999, October 1975.

BASES:

3.11C MINIMUM CRITICAL POWER RATIO (MCPR)

Operating Limit MCPR

The required operating limit MCPR's at steady state operating conditions as specified in Specification 3.11C are derived from the established fuel cladding integrity Safety Limit MCPR of 1.06, and an analysis of abnormal operational transients (1). 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.

To assure that the fuel cladding integrity Safety Limit is not exceeded during any anticipated abnormal operational transient, the most limiting transients have been analyzed to determine which result in the largest reduction in critical power ratio (CPR). The type of core wide transients evaluated were loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The local transient evaluated was the rod withdrawal error.

The limiting transient, which determines the required steady state MCPR limit for the initial portion of the fuel cycle when the scram reactivity insertion rate is greatest is different for the two types of fuel. For the 8x8 fuel bundles the loss of 100°F in feedwater heating is limiting, while the rod withdrawal error is limiting for the 7x7 fuel bundles. At later exposures in the fuel cycle, when the scram reactivity insertion rate is less, the turbine trips with failure of the turbine bypass is the limiting transient for both types of fuel. The turbine trip transient yields the largest values of ΔMCPR. When added to the Safety Limit MCPR of 1.06 the required minimum operating limit MCPR's of specification 3.11C are obtained although not required this single limit is used for the entire cycle.

Prior to the analysis of abnormal core wide operational transients an initial fuel bundle MCPR was determined. This parameter is based on the bundle flow calculated by a GE multi-channel steady state flow distribution model as described in

Section 4.4 of NEDO-20360[2] and on core parameters shown in Table 5-4 (pages 5-11) of NEDO-20855-01[1].

The evaluation of a given transient begins with the system initial parameters shown in Table 7-2 (page 7-13) of NEDO-20855-01[1] that are input to a GE core dynamic behavior transient computer program described in NEDO-10802 [3]. Also, the void reactivity coefficients that were input to the transient calculational procedure are based on a new method of calculation termed NEV which provides a better agreement between the calculated and plant instrument power distributions. The outputs of this program along with the initial MCPR form the input for further analyses of the thermally limiting bundle with the single channel transient thermal hydraulic SCAT code described in NEDE-20566.[4] The principal result of this evaluation is the reduction in MCPR caused by the transient.

Two codes are used to analyze the rod withdrawal error transient, The first code simulates the three dimensional BWR core nuclear and thermal-hydraulic characteristics. Using this code a limiting control rod pattern is determined; the following assumptions are included in this determination:

(1) The core is operating at full power in the xenon-free condition.

(2) The highest worth control rod is assumed to be fully inserted.

(3) The analysis is performed for the most reactive point in the cycle.

(4) The control rods are assumed to be the worst possible pattern without exceeding thermal limits.

(5) A bundle in the vicinity of the highest worth control rod is assumed to be operating at the maximum allowable linear heat generation rate.

(6) A bundle in the vicinity of the highest worth control rod is assumed to be operating the minimum allowable critical power ratio.

The three-dimensional BWR code then simulates the core response to the control rod withdrawal error. The second code calculates the Rod Block Monitor response to the rod withdrawal error. This code simulates the Rod Block Monitor under selected failure conditions (LPRM) for the core rod use (calculated by the 3-dimensional BWR simulation code) for the control rod withdrawal.

The analysis of the rod withdrawal error for Pilgrim Unit 1 considers the continuous withdrawal of the maximum worth control rod at its maximum drive speed from the reactor consistant with the control rod pattern shown in figure 7-12 of NEDO-20855-01[1]. A summary of the analytical methods used to determine the nuclear characteristics is given in Section 5.3 of NEDO-20360[2].

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 shown in Table 3.11.1 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 were developed[5] 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.

For the manual flow control mode, the K_f factors were calculated such that at the maximum flow state (as limited by the pump scoop tube set point) and the corresponding core power (along the rated flow control line), the limiting bundle's relative power was adjusted until the MCPR was slightly above the Safety Limit. Using this relative bundle power, the MCPR's were calculated at different points along the rated flow control line corresponding to different core flows. The ratio of the MCPR calculated at a given point of core flow, dived by the operating limit MCPR determines the K_f.

For operation in the automatic flow control mode, the same procedures was employed except the initial power distribution was established such that the MCPR was equal to the operating limit MCPR at rated power and flow.

The K_f factors shown in Figure 3.11-8[5] are conservative for the Pilgrim Unit 1 operation because the operating limit MCPR's shown in 3.11.1 are greater than the original 1.20 operating limit MCPR used for the generic derivation of K_f.

4.11C 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.

Table 3.11.1

OPERATING LIMIT MCPRs FOR CYCLE 3

Scram Reactivity Curve	Power	Operating Limit MCPR 7x7	Operating Limit MCPR 8x8
"B" curve + 2600 MWD/t	100	1.26	1.26
EOC3	92	1.31	1.39

REFERENCES

1. Pilgrim Nuclear Power Station, Unit 1 Reload No. 2 License Amendment Submittal, September 1975, (NEDO-20855-01).

2. General Electric BWR Generic Reload Application for 8x8 fuel, NEDO-20360, Revision 1, November, 1974.

3. R. B. Linford, Analytical Methods of Plant Transient Evaluations for the GE BWR, February 1973 (NEDO-10802).

4. General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50, Appendix K, NEDE-20566 (Draft), August 1974.

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

Figure 3.1[illegible] 1

Maximum Average Planar Linear Heat Generation Rate

Versus Planar Average Exposure

Initial Core With No Curtains

Planar Average Exposure (MWd/t)

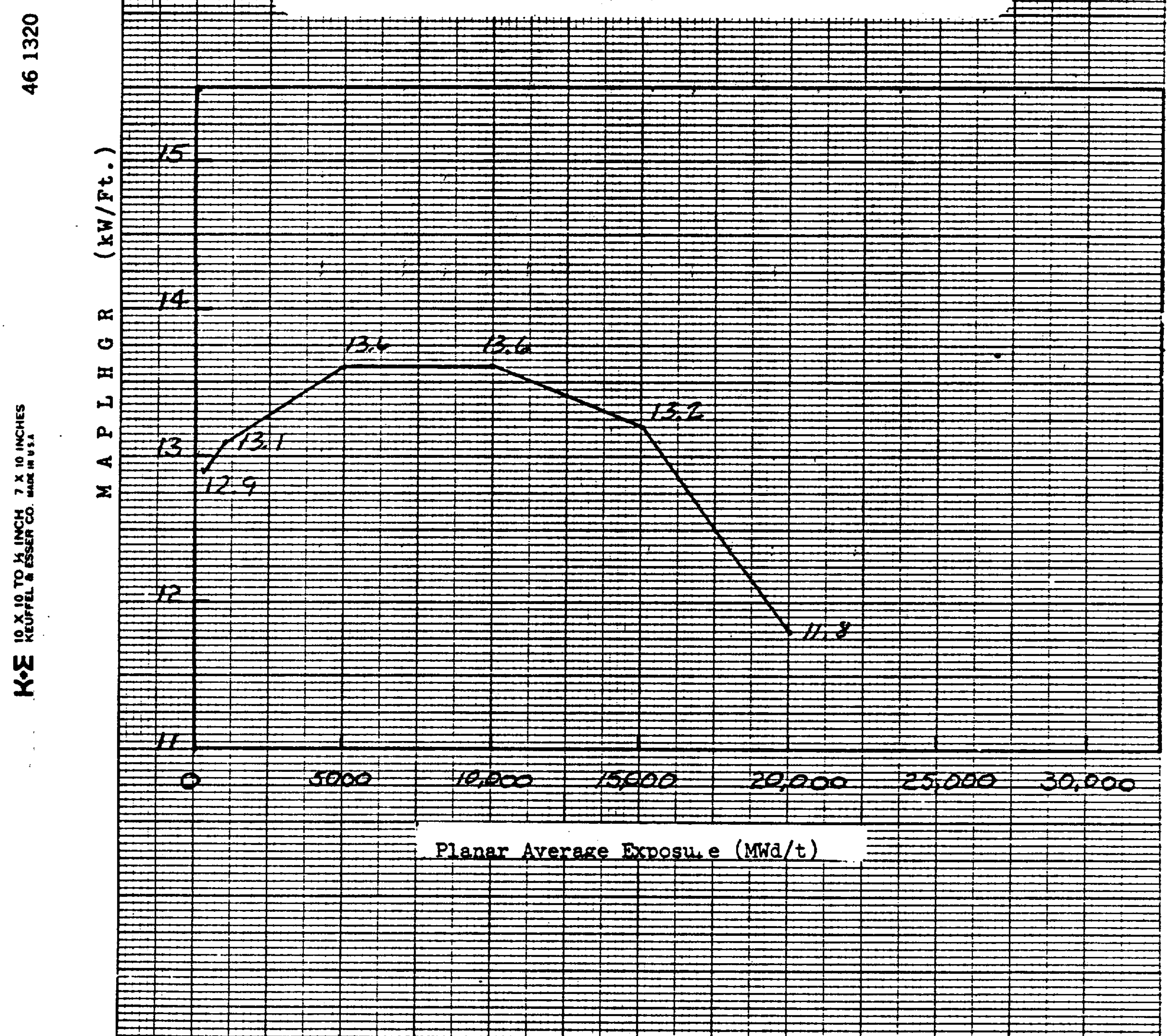
Amendment No. 15

Figure 3.11-2

MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE

VERSUS PLANAR AVERAGE EXPOSURE

Initial Core - 1 Strong Curtain



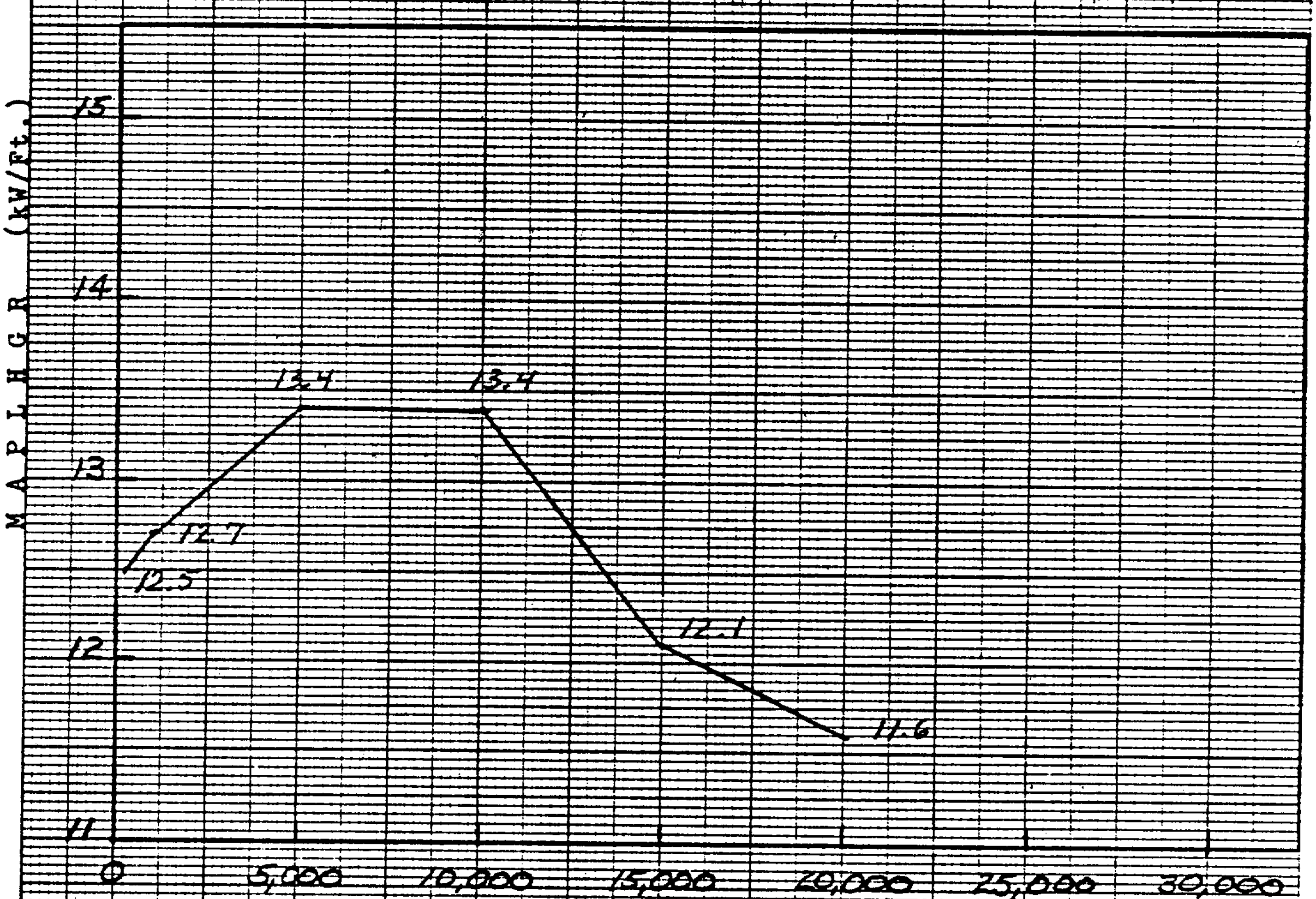
Amendment No. 15

46 1320

Figure 3.11-3

MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE VERSUS PLANAR AVERAGE EXPOSURE

Initial Core, 2 Strong Curtains



Planar Average Exposure (MWd/t)

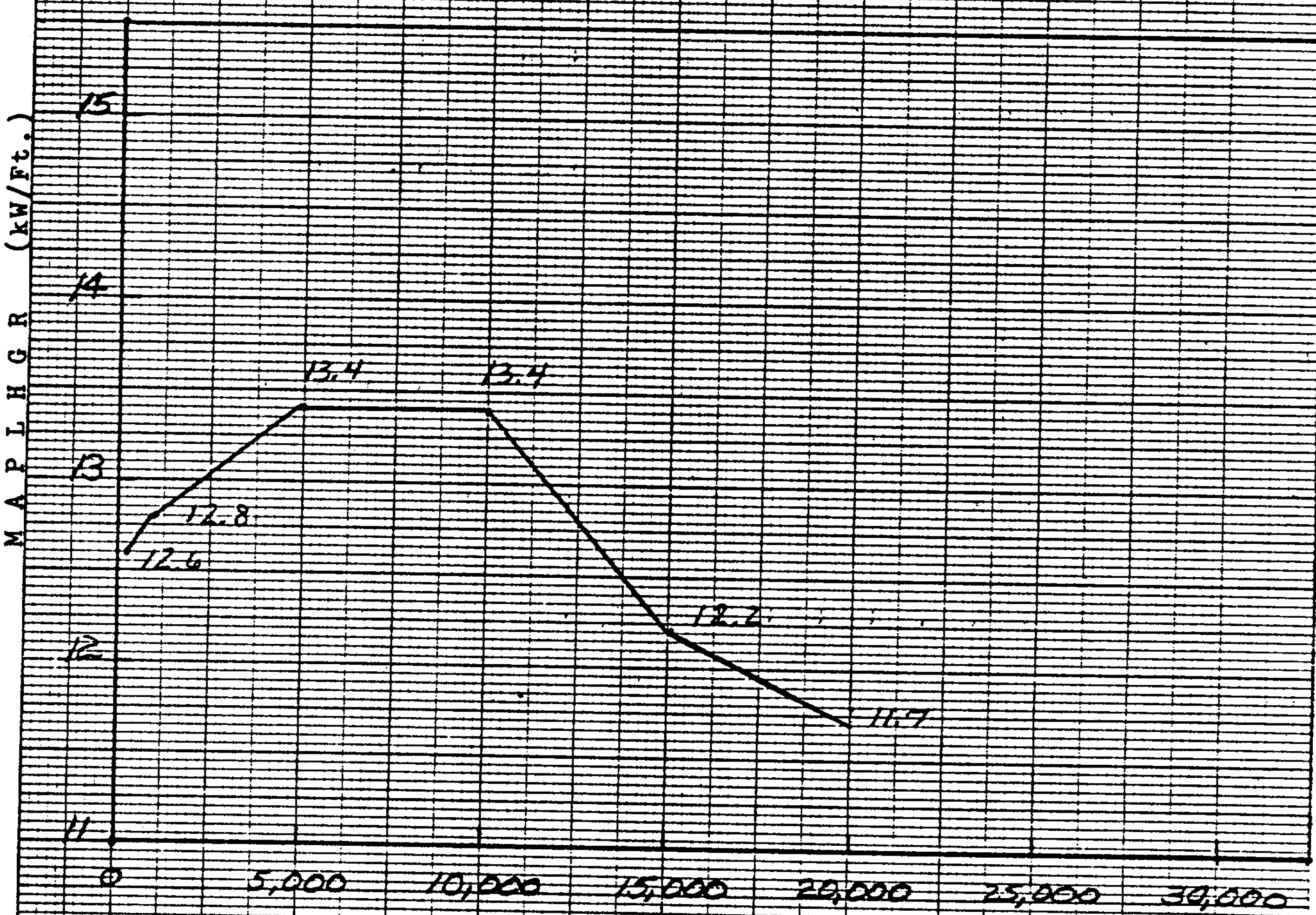
Amendment No. 15

Figure 3.11-

MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE

VERSUS PLANAR AVERAGE EXPOSURE

Initial Core, 1 Strong and 1 Weak Curtain



Planar Average Exposure (MWd/t)

Amendment No. 15

Figure 3.11-5

MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE

VERSUS PLANAR AVERAGE EXPOSURE

Initial Core, 2 Weak Curtains

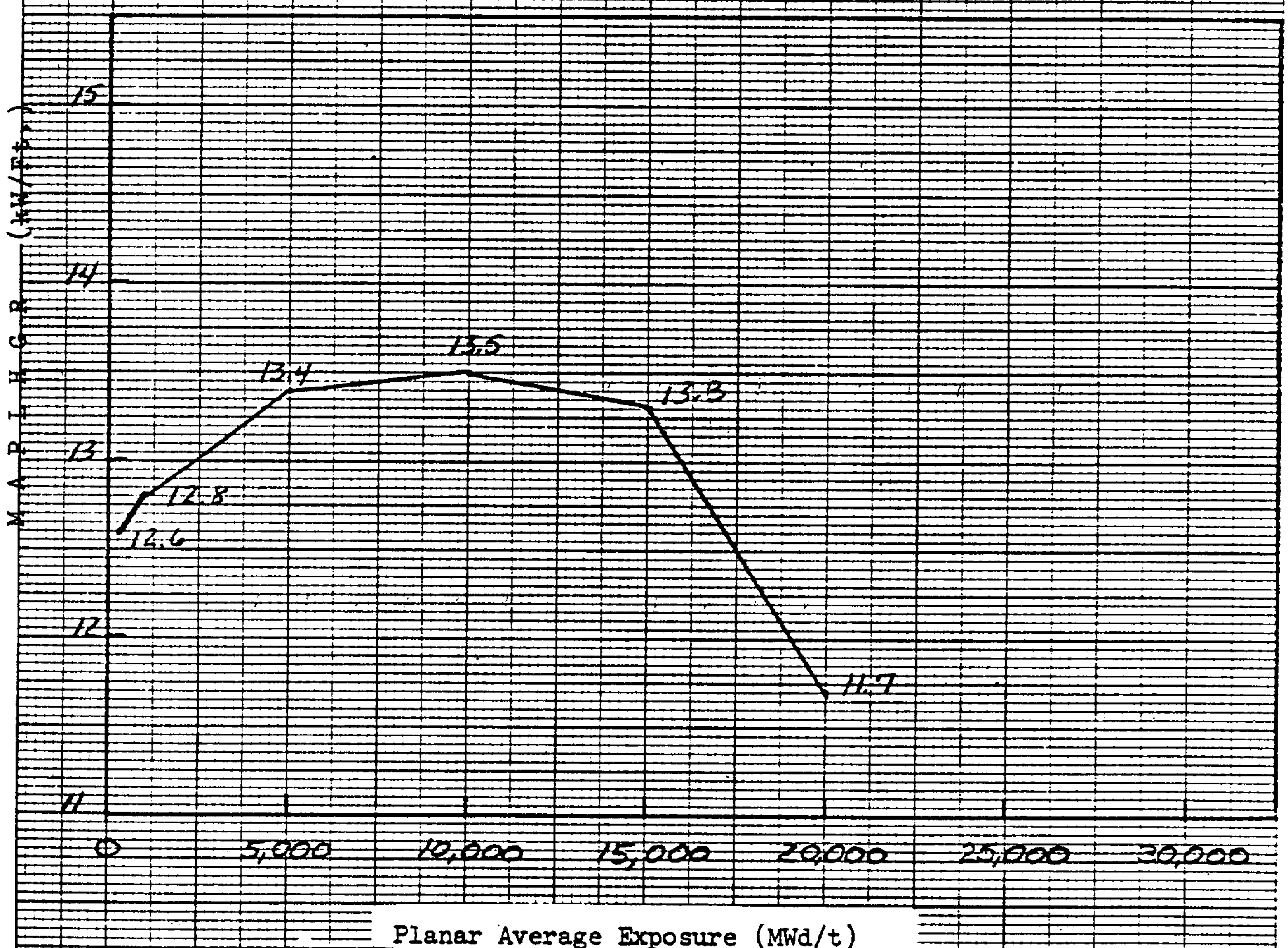


Figure 3.11-6

MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE

VERSUS PLANAR AVERAGE EXPOSURE

Initial Core, 1 Weak Curtain

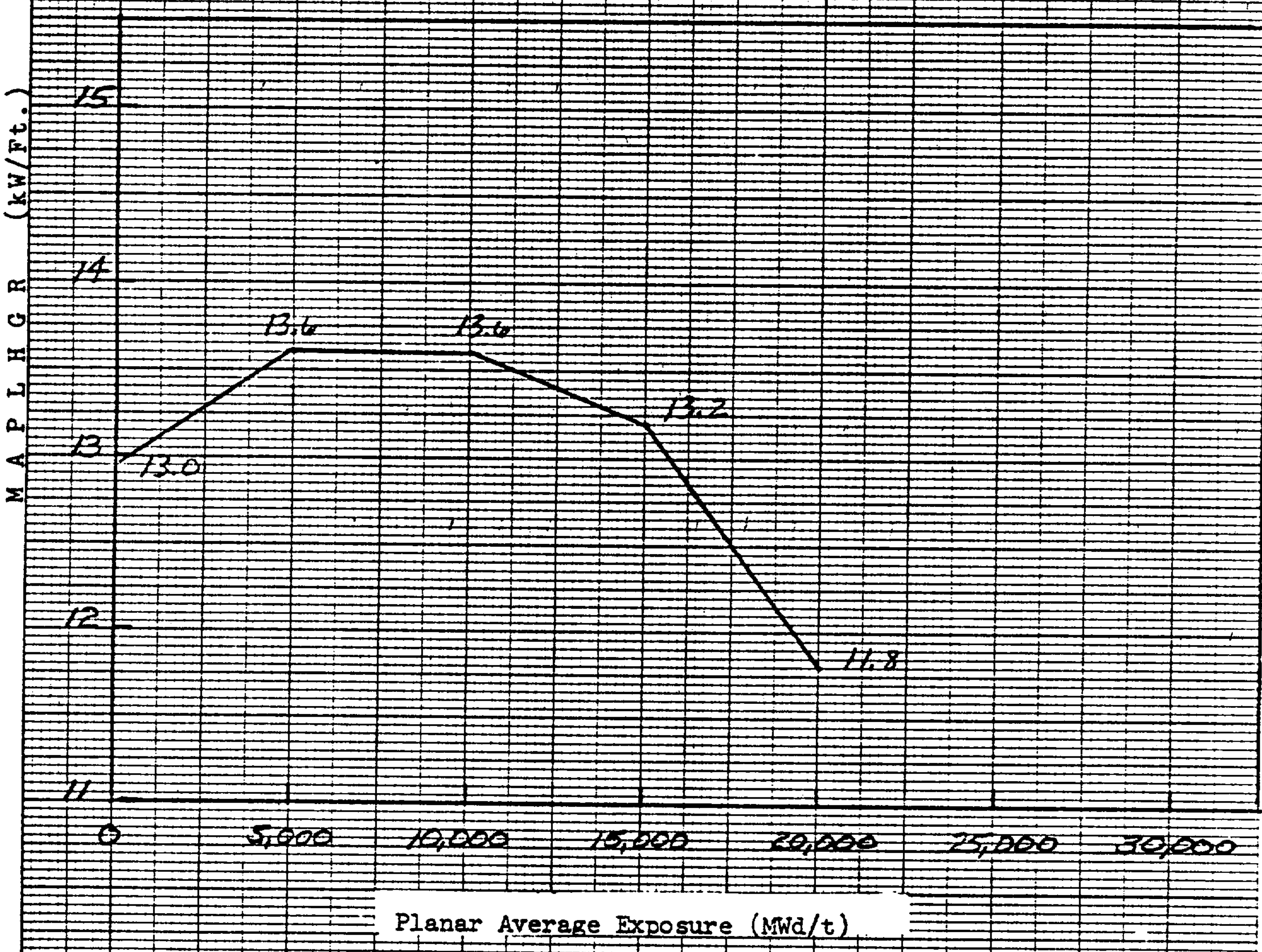


Figure 3.11-7

MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE

VERSUS PLANAR AVERAGE EXPOSURE

Reload (8D262)
Fuel

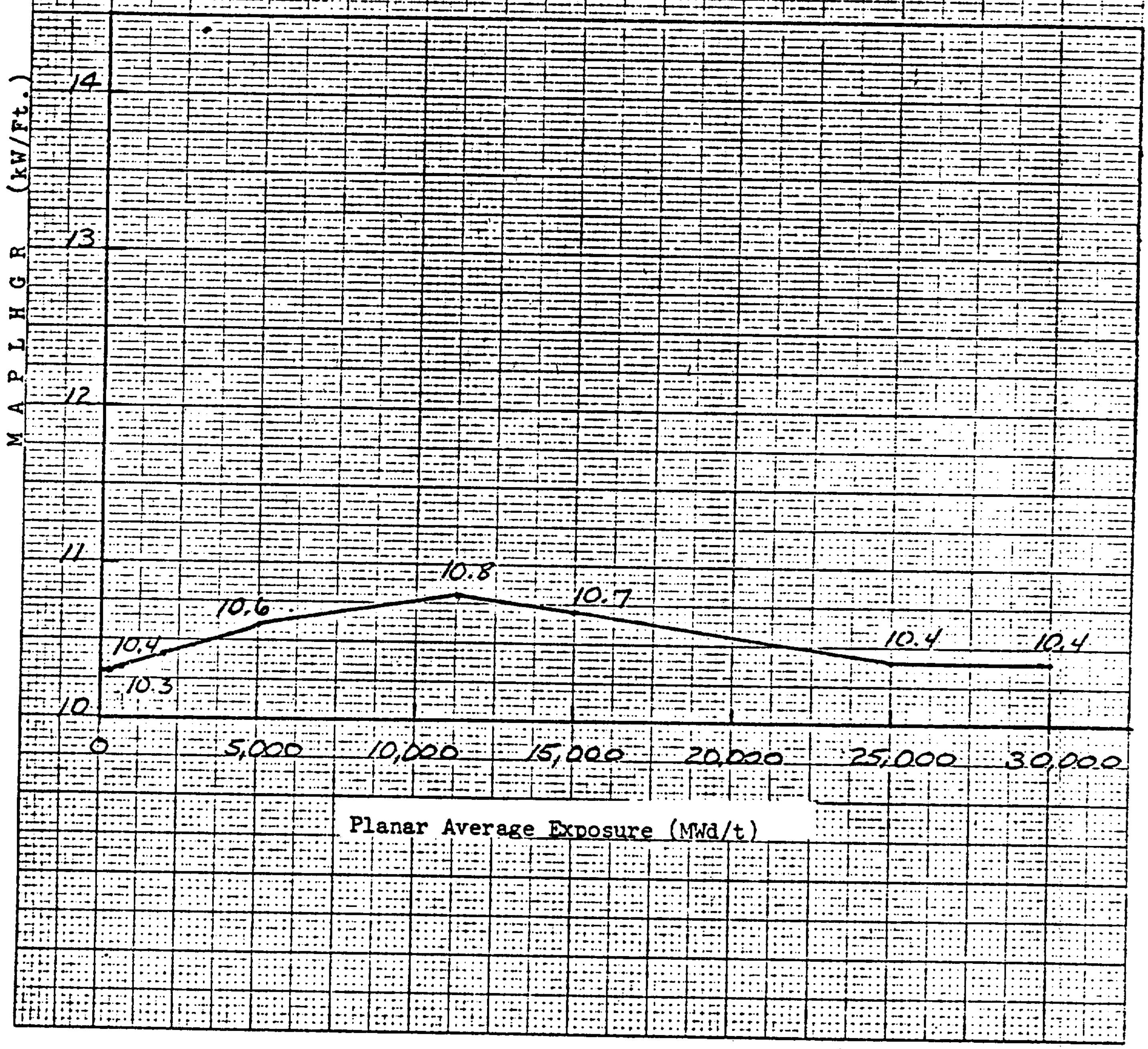
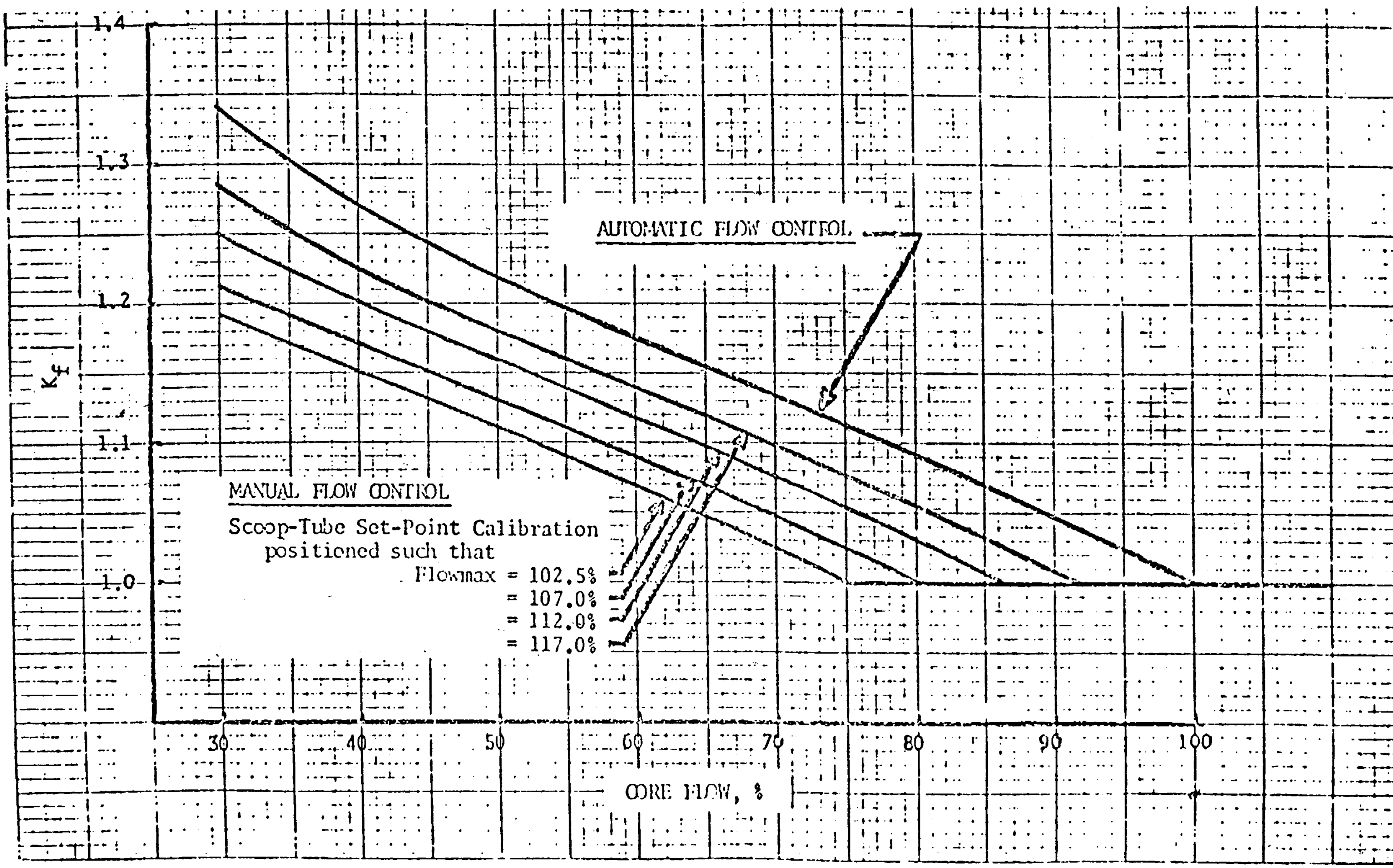


FIGURE 3.11-8

K_f FACTOR



SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO.15 TO FACILITY OPERATING LICENSE NO. DPR-35

BOSTON EDISON COMPANY

PILGRIM UNIT 1

DOCKET NO. 50-293

1.0 INTRODUCTION

Boston Edison Company has proposed to operate Pilgrim Unit 1:

(1) with additional 8 x 8 fuel assemblies, as requested in their application dated July 29, 1975, and supplements dated December 8, 1975 and March 19, 1975.

(2) using operating limits based on the General Electric Thermal Analysis Basis (GETAB), as requested in their application dated July 9, 1975, and supplements dated October 31, 1975 and November 10, 1975;

(3) using modified operating limits based on an acceptable emergency core cooling system evaluation model that conforms with Section 50.46 of 10 CFR Part 50, as requested in their application dated July 9, 1975, and supplements dated October 3, October 31, November 17, 1975, March 19, and April 12,1976.

2.0 RELOAD

2.1 DISCUSSION

The reference core loading for Pilgrim 1 Reload 2 consists of 428 initial 7 x 7 fuel assemblies, 20 Reload 1 8 x 8 assemblies, and 132 Reload 2 8 x 8 fuel assemblies. The reload assemblies are scatter loaded throughout the core. The acceptability of the neutronic, thermal-hydraulic, and mechanical design of 8 x 8 fuel assemblies during normal operation, operational transients and postulated accidents was evaluated by the NRC staff in a previous report[1/]. The use of 8 x 8 fuel assemblies for reloads was also reviewed by the Advisory Committee on Reactor Safeguards and discussed in its report dated February 12, 1974.[2/] The use of 8 x 8 reload fuel assemblies in Pilgrim 1 was evaluated and approved by Amendment No. 4 to Facility Operating License No. DPR-35 dated July 23, 1974.

1/ Technical Report on General Electric Company 8 x 8 Fuel Assembly, dated February 5, 1974, by the Directorate of Licensing.

Our safety evaluation of this reload (Reload No. 2) for the Pilgrim Unit 1 core is based on the licensee's application as amended, on information contained in a GE topical report, NEDO-20360[3] referred to in the application, and the Pilgrim Unit 1 Appendix K Reanalysis. The NEDO-20360 report is still being evaluated by the staff for use as a topical. Our use of that report in this analysis was limited to considerations applicable to Pilgrim 1 and does not imply acceptability of its use for other facilities.

2.2 EVALUATION

2.2.1 NUCLEAR CHARACTERISTICS

The information presented in the licensing submittal for the reconstituted core[5/6] closely follows the guidelines of Appendix A of Reference 3. Up to 132, 8 x 8 fuel bundles, each having an average enrichment of 2.62% by weight of the uranium 235 isotope, are loaded throughout the core. The remaining poison curtains will be removed. The plugs will not be removed from the bypass leakage augmentation holes. Thus, nearly 23 percent of the fuel bundles are being replaced for this reload. The loading scheme consists, with an exception, of replacing one fuel bundle in a four bundle array surrounding a control rod with a reload fuel bundle. The exception to this loading pattern is that around the periphery of the core two diagonally located fuel bundles of the four surrounding a control rod are replaced. The 8 x 8 reload fuel for the Reload 2 core are, therefore, basically scatter loaded. The data in Reference 5 indicates that the nuclear characteristics of the Reload 2 8 x 8 fuel bundles are similar to those previously loaded. Thus, the total control system worth, temperature, and void dependent behavior of the reconstituted core will not differ significantly for those values which were previously analyzed and approved for Pilgrim Unit 1.

2/ Report on General Electric 8 x 8 Fuel Design for Reload Use, Advisory Committee on Reactor Safeguards, February 12, 1974.

3/ General Electric Boiling Water Reactor Generic Reload Application for 8 x 8 Fuel, NEDO-20360 Supplement 2 (May 1975).

4/ Pilgrim Nuclear Power Station, Loss of Coolant Accident Analyses (Plugged Bypass Holes (July 9, 1975).

5/ General Electric BWR Reload-2 Licensing Submittal for Pilgrim Unit 1 Nuclear Power Station - NEDO-20855, June 1975.

6/ General Electric BWR Reload No. 2 Licensing Submittal for Pilgrim Nuclear Power Station Unit 1, with Bypass Holes Plugged (NEDO 20855-01), September 1975.

The shutdown margin of the reconstituted core meets the Technical Specification requirement that the core be at least 0.25%Δk subcritical in the most reactive operating state with the largest worth control rod fully withdrawn and with all other control rods fully inserted. A minimum shutdown margin of 0.0112Δk exists at the beginning of the Reload 2 cycle. Thus the value of R, the calculated difference between the core reactivity at the beginning of the cycle and the core reactivity at any time later in the cycle where it would be greater than at the beginning, is zero. However, a 0.04% Δk allowance for inverted tubes in the control blades is made and added to R, giving a value for R of 0.04% Δk. This value of R is needed for the Reload 2 cycle Technical Specifications to assure that a sufficient shutdown margin is maintained throughout the cycle.

The information presented in Reference 5 indicates that a boron concentration of 700 ppm in the moderator will make the reactor subcritical by at least 0.03Δ k at 20 °C, xenon free. Therefore, the alternate shutdown requirement of the General Design Criteria is met.

The Technical Specification requirement for the storage of fuel for Pilgrim Unit 1 is that the effective multiplication factor k_{eff} of the fuel as stored in the fuel storage rack is equal to or less than 0.90. This is achieved if the uncontrolled k_∞ of a single fuel bundle is less than 1.30[3/] at 65 °C. The 8 x 8 (8D262) fuel bundle, at both zero exposure and the peak reactivity point, has a k_∞ less than 1.25 and, therefore, meets the dry and spent fuel storage requirement for Pilgrim Unit No. 1.

The full power scram reactivity curves for the Reload 2 cycle are the GE generic "B" curve and the end of cycle 3 curve shown in Figure 7-6 of Reference 6. The "B" scram curve is applicable to the reload 2 cycle for the first 2600 MWD/t of exposure while the end of reload 2 cycle scram curve is applicable for the remainder of the cycle. The scram curves are multiplied by a design conservatism factor of 0.8 for use in the abnormal operating transient analyses.

The void and Doppler coefficients of reactivity for the Reload 2 cycle are given in Table 6-1 of Reference 6. The void coefficient of reactivity at the core average void fraction of 34% varies from -15.3 to -17.0 x 10^{-4} $\Delta k/k/\Delta\%V$. The Doppler coefficient of reactivity at a fuel temperature of 1202°F varies from -1.15 to -1.23 x 10^{-5} $\Delta k/k/\Delta T$.

2.2.1.1 CONCLUSION

Thus, based on our review of the information presented in the Pilgrim Unit 1 licensing submittal, and the generic 8 x 8 reload report (Reference 3), we conclude that the nuclear characteristics (e.g., scram reactivity, void coefficient of reactivity and Doppler coefficient of reactivity) and performance of the reconstituted core for the Reload-2 cycle will not differ significantly from previously analyzed and approved Pilgrim Unit 1 fuel cycles and are acceptable.

2.2.2 Mechanical Design

The Pilgrim 1 reload fuel consists of 132 new General Electric type 8D262, 8 x 8 fuel bundles, with average bundle enrichment of 2.62 w/o identical to the fuel described in "GE/BWR Generic Reload Application for 8 x 8 Fuel,"3/. This generic report has been reviewed and with some modifications was found acceptable for use for reactors containing 8 x 8 fuel reload fuel, when supplemented with information required by our status report7/ on the GE generic report evaluation.

Mechanical and operating parameters for the 8 x 8 assemblies are compared to the 7 x 7 assemblies in Table 1. The small diameter rods, with lower linear heat generation rate and increased cladding thickness/diameter ratio for the 8 x 8 fuel design as compared to the 7 x 7 fuel assemblies, result in increased safety margins with respect to maximum design linear power and maximum fuel temperature.

Fuel performance calculations that account for the effects of fuel densification have been performed with our approved version of the General Electric analytical model, GEGAP III8/ 9/. Fuel densification results in increases in stored energy, linear thermal output, and the probability of local power spikes from axial gaps. The primary effects of densification on the fuel rod mechanical design are manifested in calculations of fuel-clad gap conductance and cladding collapse time. The approved analytical model incorporates time-dependent fuel densification, time dependent gap closure and cladding creepdown for the calculation of gap conductance. Clad collapse has not been observed in BWR fuel rods and is calculated to occur at residence times in excess of 5 years.

7/ Status Report on the Licensing Topical Report "General Electric Boiling Water Reactor Generic Reload Application for 8 x 8 Fuel," NEDO-20360, Revision 1 and Supplement 1 by Division of Technical Review, Office of Nuclear Reactor Regulation, U. S. Nuclear Regulatory Commission, April, 1975.

8/ GEGAP-III, " A Model for the Prediction of Pellet Clad Thermal Conductance in BWR Fuel Rods," NEDO-20181, December 3, 1973, Supplement 1 (Proprietary).

9/ V. A. Moore, NRC letter to I. S. Mitchell, "Modified GE Model for Fuel Densification," Docket 50-321, March 22, 1974.

The reload 8 x 8 fuel design is currently in operation in Nine Mile Point-1, Pilgrim-1, Monticello, Dresden 2 and 3, Quad Cities 1 and 2 and Vermont Yankee. Post irradiation examination of reload 8 x 8 assemblies at Monticello after one complete cycle indicated satisfactory performance.

2.2.2.1 CONCLUSION

On the basis of our review of the generic 8 x 8 reload report, current operating experience with the 8 x 8 reload design in similar plants, and our review of Boston Edison's Reload-2 licensing submittal, we conclude that for Pilgrim Unit 1 Reload 2:

1. the fuel rod mechanical design provides acceptable safety margins for normal operation,

2. the effects of fuel densification have been acceptably accounted for in the fuel design, and

3. fuel rod integrity will remain satisfactory during transients and accidents.

Specific generic items related to accident and transient evaluation have been identified in Reference 7 and are being reviewed on a generic basis.

2.2.3 Thermal-Hydraulics (GETAB)

To apply GETAB to the Technical Specifications involves; 1) establishing the fuel damage safety limit, 2) establishing limiting conditions of operation such that the safety limit is not exceeded for normal operation and anticipated transients, and 3) establishing limiting conditions for operation such that the initial conditions assumed in accident analyses are satisfied. We have evaluated and report herein the Pilgrim Unit 1 developed thermal margins based on the NEDO-10958 report[10/] and plant specific input information provided by the licensee.[6/] [11/]

2.2.3.1 Fuel Cladding Integrity Safety Limit MCPR

The fuel cladding integrity safety limit MCPR is 1.06. It is based on the GETAB statistical analysis which assures that more than 99.9% of the fuel rods in the core are expected to avoid boiling transition during abnormal operational transients. The uncertainties in the

10/ "General Electric BWR Thermal Basis (GETAB): Data, Correlation and Design Application," NEDC-10958, 73NED9, Class I, November, 1973.

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

core and system operating parameters and the GEXL correlation, Table 5-1 of the licensee submittal,[6] combined with the relative bundle power distribution in the core form the basis for the GETAB statistical determination of the safety limit MCPR. The tabulated list of uncertainties for Pilgrim Unit 1 is the same as or more conservative than those reported in NEDO-10958[10] and NEDO-20340[12]. The Pilgrim Unit 1, Reload 2 standard deviation for the TIP readings uncertainty is 8.7% whereas the GETAB NEDO-10958 report shows 6.3%. The increase in uncertainty for the Pilgrim Unit 1 is a consequence of the increase in uncertainty in the measurement of power in a reload core. A TIP reading uncertainty of 6.3% would be applicable if this were the initial core. The 4.46-5.40% uncertainty of the bypass void effect on TIP which accounts for additional uncertainty due to bypass void content resulting from plugging the core support plate leakage augmentation holes has been taken into account.

The reactor core selected for the GETAB statistical analyses is a typical 251/764 core. This typical core is of the same reactor class as Pilgrim Unit 1 core (224/580) but is larger. The bundle power distribution used for the GETAB application has more high power bundles than the distribution expected during operation of the Pilgrim 1 reactor. This results in a conservative value of the MCPR which meets the 99.9% statistical criterion.

We conclude that the proposed fuel integrity safety limit, a MCPR of 1.06 is acceptable for Pilgrim Unit 1 fuel cycle number three (Reload 2).

2.2.3.2 Operating Limit MCPR

Various transient events will reduce the MCPR below the operating MCPR. To assure that the fuel cladding integrity safety limit (MCPR of 1.06) is not exceeded during anticipated abnormal operational transients, the most limiting transients have been analyzed to determine which one results in the largest reduction in critical power ratio (ΔMCPR). The licensee has submitted the results of analyses of those transients which produce a significant decrease in MCPR. The types of core wide transients evaluated were losses of flow, pressure and power increases, and coolant temperature decreases. The most limiting transients in these categories were two-pump trip, turbine trip without bypass, and loss of feedwater heating. Of these the most limiting transients were loss of feedwater heater, assuming beginning of cycle (BOC) scram reactivity insertion rates (100% of rated power and flow from BOC to 2600 MWd/t) and turbine trip without bypass assuming end of cycle (EOC)

12/ "Process Computer Performance Evaluation Accuracy," and Amendment 1, NEDO-20340 and NEDO-20340-1, dated June 1974 and December 1974.

scram reactivity insertion rates (92% of rated power, 100% of rated flow). The loss of feedwater transient results in a MCPR of 0.17 (7 x 7 fuel) and 0.20 (8 x 8 fuel) and the turbine trip without bypass results in a maximum MCPR of 0.25 (7 x 7 fuel) and 0.33 (8 x 8 fuel). Addition of these MCPR's to the safety limit MCPR gives the minimum operating limit MCPR for each fuel type required to avoid violation of the safety limit, should this limiting transient occur. Therefore, the operating limit MCPR's are 1.31 for 7 x 7 fuel and 1.39 for 8 x 8 fuel.

The transient analyses were evaluated with scram reactivity insertion rates that include a design conservatism factor of 0.80. The initial conditions[4/5/6/] and the design conservatism factors used for the worst operational transient are acceptable. The initial MCPR assumed in the transient analyses was equal to or greater than the established operating limit MCPR of 1.31 and 1.39 for 7 x 7 and 8 x 8 fuel assemblies respectively. This results in a conservative MCPR and is acceptable.

A GE study[10/] has shown that the required operating MCPR varies with the axial and local power peaking distribution. Axial peaking in the middle or upper portion of the core results in higher required MCPR's than peaking in the lower portion of the core. In the analyses the axial power peak was assumed to be representative of beginning-of-cycle conditions and to be located at the midplane (node 12, axial peak-to-average of 1.40).

The R-factors, which are a function of the local power peaking, assumed in the analyses are also representative of beginning-of-cycle conditions. The values assumed are 1.100 for 7 x 7 fuel and 1.102 for 8 x 8 fuel. During the cycle the local peaking and therefore the R-factor is reduced while the peak in the axial shape moves toward the bottom of the core. Although the operating limit MCPR would be increased by approximately 1% by the reduced end-of-cycle R-factor, this is offset by the reduction in MCPR resulting from the relocation of the axial peak to below the midplane.

Conservatism was applied in the determination of the required operating limit MCPR because the assumed axial and local peaking were representative of the beginning of the fuel cycle. This is the worst consistent set of axial and local peaking.

It is concluded from the analyses of the limiting pressure/power transient, a turbine trip with bypass failure, that Pilgrim Unit 1 Reload 2 can operate at 100% power until that point in the fuel cycle when the scram reactivity is less than that of the "B" scram reactivity curve which occurs at 2600 MWd/t into cycle 3. The power will then be limited such that it is equal to 92% of rated power at 100% of rated flow at the end of the fuel cycle.

Figure 7-11 of Reference 6 shows the resultant maximum power level profile as a function of cycle exposure. The derate schedule is shown as a linear function of fuel exposure connecting the specified calculational operating power limit points. The use of the linear relationship to connect the two calculational points in Figure 7-11 is conservative because the scram reactivity degrades gradually and would thus be a smooth function of core exposure. Thus, for the pressure transients connecting the actual calculational points with a straight line will conservatively maintain a minimum pressure margin of 25 psi since the actual allowable power level would be expected to lie somewhere above this operating limit line. Conservatism is incorporated into the operating MCPR's by imposing the limiting operating MCPRs of 1.31 for 7 x 7 fuel and 1.39 for the 8 x 8 fuel, calculated for the worst degraded condition (end-of-cycle), over the entire cycle. Thus, the analyses at the endpoints of 100% power with "B" scram curve, and 92% power with the EOC-3 scram curve, are bounding for all MCPR and pressure transients for all power and burnup combinations shown on Figure 7-11.

Operation at 100% power level is permissible over the range of "B" scram curve applicability. Beyond that point, 2600 MWd/t into the cycle, the power will be reduced from 100% power to 92% power at EOC-3, limited by the maximum power profile shown on Figure 7-11. This limit will be administered by imposing small step derates, each of which is valid for some incremental exposure. For each derate there will be administered a corresponding nominal power-flow line, interpolated between the nominal 100% flow control line and the nominal 92% flow control line shown in Figure 5-1 on Ref. 13. Operation, therefore is restricted to power/flow conditions along or below these derated flow control lines which are consistent with the rod patterns necessary to give the step derates mentioned above.

2.2.3.3 Local Event Rod Withdrawal Error

The rod withdrawal error (RWE) transient is discussed in Reference 6 in terms of worst case conditions. Assumptions and descriptions of the rod withdrawal event are given in Reference 3. The information in these two references indicate that the local power range monitor subsystem (LPRM's) will detect high local powers and alarm. However, if the operator ignores the LPRM alarm, the rod block monitor subsystem (RBM) will stop the rod withdrawal while the critical power ratio is still equal to or greater than the 1.06 MCPR safety limit and the cladding is under the one percent plastic strain limit. The analysis of the rod withdrawal error (RWE) results in operating

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13/ Letter from G.C. Andognini, Boston Edison Company to D. L. Ziemann USNRC dated March 19, 1976.

MCPR's of 1.26 for 7 x 7 fuel and 1.20 for 8 x 8 fuel with the RBM setting at 107% of its initial value. This transient is the most limiting event for full power operation in the early portion of the cycle ("B" scram curve) for the 7 x 7 fuel. For the 8 x 8 fuel the loss of feedwater heater transient produced a greater ΔCPR than the RWE. For the latter portion of the cycle (EOC-3 scram curve) at reduced power, a turbine trip without bypass produces the greatest ΔCPR. Since this ΔCPR is the greatest, for both fuel types, over the entire cycle 3; the licensee has used these to impose the operating MCPR limit for Pilgrim Unit 1 Reload 2. With the operating MCPR limits of 1.31 (7 x 7) and 1.39 (8 x 8) the RWE transient will have a greater margin to the 1.06 MCPR safety limit with the RBM at the 107% setpoint. We conclude that the consequences of this localized transient are acceptable.

2.2.3.4 Operating MCPR Limits for Less Than Rated Power and Flow

For the limiting transient of recirculation pump speed control failure at lower than rated power and flow condition, the licensee will conform to Technical Specification limiting conditions for operation; Paragraph 3.11.C. This requires the licensee to maintain the required operating MCPR greater than 1.31 (7 x 7) and 1.39 (8 x 8) times the K_f factor for core flows less than rated. The K_f factor curves were generically derived and assure that the most limiting transient occurring at less than rated flow will not exceed the safety limit MCPR of 1.06. We conclude that the calculated consequences of the anticipated abnormal transients do not violate the thermal and plastic strain limits of the fuel or the pressure limits of the reactor coolant boundary.

2.2.3.5 CONCLUSION.

Based upon the above, we conclude that the analyses and operating limits based upon the use of the General Electric Thermal Analysis Basis have been conservatively applied to Reload-2 (Cycle-3) and are acceptable.

3.0 ACCIDENT ANALYSIS

3.1 ECCS Appendix K Analysis

On December 27, 1974, the Atomic Energy Commission issued an Order for Modification of License implementing the requirements of 10 CFR 50.46 "Acceptance Criteria and Emergency Core Cooling Systems for Light Water Nuclear Power Reactors." One of the requirements of the Order was that prior to any license amendment authorizing any core reloading"...the licensee shall submit a re-evaluation of ECCS cooling performance calculated in accordance with an acceptable evaluation model which conforms to the provisions of 10 CFR Part 50, 50.46." The order also required that the evaluation shall be accompanied by such proposed changes in Technical Specifications or license amendments as may be necessary to implement the evaluation results.

On July 9, 1975, the licensee submitted an evaluation of the ECCS performance for the design basis pipe break for Pilgrim Unit No. 1, along with an amendment requesting changes to the Technical Specifications for Pilgrim Unit No. 1 to implement the results of the evaluation.[4/] [14/] The licensee incorporated further information relating to the details of the ECCS evaluation by referencing an appropriate lead plant analysis[15/] to show compliance to the 10 CFR 50.46 criteria and Appendix K to 10 CFR Part 50.

The Order for Modification of License issued December 27, 1974, stated that evaluation of ECCS cooling performance may be based on the vendor's evaluation model as modified in accordance with the changes described in the staff Safety Evaluation Report of the Millstone Nuclear Power Station dated December 27, 1974.

The background of the staff review of the General Electric (GE) ECCS model and its application to Pilgrim Unit 1 is described in the Staff Safety Evaluation Report (SER) for these facilities dated December 27, 1974 (The December 27, 1974 SER) issued in connection with the Order. The bases for acceptance of the principal portions of the evaluation model are set forth in the staff's Status Report of October 1974 which are referenced in the December 27, 1974 SER. The December 27, 1974 SER also describes the various changes required in the earlier GE evaluation model. Together the December 27, 1974 SER and the Status Report and its Supplement, describe an acceptable ECCS evaluation model and the basis for the staff's acceptance of the model. The Pilgrim Unit 1 evaluation which is covered by this SER properly conforms to the accepted model.

14/ Letter from J. E. Howard, Boston Edison Company, to D. L. Ziemann, USNRC, dated October 3, 1975.

15/ Quad Cities Unit 2, Special Report No. 15, Supplement C, Docket No. 50-265, April 8, 1975.

With respect to reflood and refill computations, the Pilgrim Unit 1 analysis was based on modified version of the SAFE computer code, with explicit consideration of the staff recommended limitations. These are described on pages 7 and 8 of the December 27, 1974 SER. The Pilgrim evaluation did not attempt to include any further credit for other potential changes which the December 27, 1974 SER indicated were under consideration by GE at that time.

During the course of our review, we concluded that additional individual break sizes should be analyzed to substantiate the break spectrum curves submitted in connection with the evaluation provided in August 1974.

We also requested that other break locations be studied to substantiate that the limiting break location was the recirculation line.

The additional analyses supported the earlier submittal which concluded that the worst break was the complete severence of the recirculation line. These additional calculations provided further details with regard to the limiting location and size of break as well as worst single failure for the Pilgrim Unit 1 design. The limiting break which is the design basis accident is the complete severence of the recirculation suction line assuming a failure of the LPCI injection valve.

3.1.1 Conclusion

We have reviewed the evaluation of ECCS performance submitted by Boston Edison Company for Pilgrim Unit 1 and conclude that the evaluation has been performed wholly in conformance with the requirements of 10 CFR 50.46(a). Therefore, operation of the reactor would meet the requirements of 10 CFR 50.46 provided that operation is limited to the maximum average planar linear heat generation rates (MAPLHGR) of figures C-12A through C-12g of the Boston Edison Company submittal dated July 9, 1975;[4/] and to a minimum critical power ratio (MCPR) greater than 1.18.

However, certain changes must be made to the proposed technical specifications to conform with the evaluation of ECCS performance. The largest recirculation break area assumed in the evaluation was 4.35 square feet. This break size is based on operation with a closed valve in the equalizer line between the two recirculation loops. Therefore, reactor operation is prohibited unless the valve in the equalizer line is closed.

The ECCS performance analysis assumed that reactor operation will be limited to a MCPR of 1.18. However, the operating MCPR limits will be more limiting.

The LOCA analysis assumed all ADS valves operated for small line breaks with HPCI failure. Since the licensee did not provide a LOCA analysis with one ADS valve out of service for small line breaks, the Technical Specifications are required to be modified so as not to allow continuous operation with any ADS valve out of service.

3.2 Steamline Break Accident

The steamline break accident analysis as presented by the licensee is acceptable based on our generic review of NEDO-20360. 3/

3.3 Fuel Loading Error

Fuel loading errors are discussed in Reference 6 and 17 for an 8 x 8 reload fuel bundle placed in an improper position or rotated 180 degrees in a location near the center of the core. The information in Reference 17 indicates that a fuel loading error results in a peak linear heat generation rate (LHGR) of 16.6 kW/ft and a minimum critical power ratio (MCPR) of 1.01 in the misplaced fuel bundle. The peak LHGR is less than that needed to cause a 1% plastic strain in the cladding. The MCPR of 1.01 in the misplaced fuel bundle indicates that about two fuel rods may be expected to experience boiling transition. Fuel assemblies adjacent to a misplaced fuel assembly will be negligibly affected. We conclude that the consequences of a fuel loading error are acceptable.

3.4 Control Rod Drop Accident

The control rod drop accident for the Pilgrim Unit 1 reloaded core is within the bounding analysis presented in NEDO-20360 3/. The Doppler coefficient of reactivity, the accident reactivity shape and magnitude function, and the rod drop scram reactivity functions are compared with the technical bases presented in NEDO-20360. This analysis is performed for Doppler coefficients of reactivity at the beginning of the Reload 2 cycle, zero void fraction, and at both cold (20°C) and hot (286°C) startup conditions. Incremental bank withdrawal is also assumed; intermediate positions for Group 3 Control rods are 1, 2, and 4 ft. withdrawn and for group 4 Control rods 2.5 ft. withdrawn. It is shown that the maximum values of the parameters for this reloaded core will not exceed the bounding values.

Therefore, we conclude that the consequences of a control rod drop accident from any insequence control rod during startup will be below the design limit of 280 cal/gm.

3.5 Fuel Handling Accident

With respect to fuel handling accidents, the licensee calculates the number of failed fuel rods in the same manner as for the previous core loading. The total activity from an 8 x 8 assembly (63 rods) is given as equal to the 7 x 7 assemblies (49 rods). Accordingly, under the worst postulated condition, i.e., equivalent fission gas release from the fuel, the total activity release from a fuel handling accident with an 8 x 8 would be equivalent to the 7 x 7. For the 7 x 7 assemblies, the fission gas release has been calculated to be 20.8% of the 10 CFR 100 limits. Based on the conservative assumptions for fission gas release and the lower final temperatures, we conclude that the consequences of the fuel handling accident with the 8 x 8 assemblies will not exceed that with the 7 x 7 assemblies.

3.6 Overpressure Analysis

The licensee submitted an overpressure analysis in order to demonstrate that an adequate margin exists below the ASME code allowable vessel pressure of 110% of vessel design pressure. The transient analyzed was the closure of all main steam isolation valves with high neutron flux scram. The analysis was performed at 92% power with the end of cycle (EOC) scram reactivity insertion rate curve (this case with EOC scram and void coefficient is more limiting than 100% power and the "B" scram curve for Cycle 3), scram initiated by high neutron flux, void reactivity applicable to this reload, no credit for relief function of safety/relief valves, and the failure of one safety/relief valve to operate was assumed. This analysis (Reference 11) utilized input parameters which were equal to or more severe than those which will be experienced during this fuel cycle. The results of the analysis indicate that the peak pressure at the vessel bottom was calculated to be 1311 psig yielding a 64 psi margin below the code allowable, which is acceptable to the staff.

3.7 Conclusion

We have concluded that the accident analyses for Reload 2 have been performed in accordance with methods acceptable to the NRC staff and demonstrate that the consequences of postulated accidents are acceptable.

4.0 TECHNICAL SPECIFICATION CHANGES

The proposed Technical Specification changes based on GETAB for Pilgrim Unit 1 identify the same Fuel Cladding Integrity Safety Limit MCPR of 1.06, but different operating limit MCPRs for the fuel types. We accept the incorporation of the Operating Limit MCPRs specified in Section 5.4 of Reference 6 into the Technical Specification for the Pilgrim Unit.

The proposed Technical Specification Limiting Conditions of Operation present two limitations on power distribution related to the LOCA analysis. These are the limiting assembly maximum average planar power density, MAPLHGR, and the minimum power ratio limit related to boiling crisis, MCPR. The MCPR value used in the LOCA analysis was 1.18 and this value is less than the value determined from the transient analysis which will be incorporated in the proposed Technical Specifications. The bases for establishing the limiting value of MAPLHGR are indicated in Section 3.0 of this evaluation.

Boston Edison did not include the equalizer line area in the LOCA analysis, therefore, the license has been modified to require that the equalizer line valves remain closed at all times during reactor operation.

The proposed changes to the APRM flux scram setting and APRM rod block trip setting have not been issued at this time as there are outstanding items to be resolved.

The LOCA analysis assumed all ADS valves operated for small line breaks with HPCI failure. Since the licensee did not provide a LOCA analysis with one ADS valve out of service for small line breaks, we have modified the Technical Specifications so as not to allow continuous operation with any ADS valve out of service; except one valve may be out of service for seven days, with HPCI tested daily. The modified specification reduced the period of time that one ADS valve may be out of service from 30 days to 7 days.

For each change in power level a new flow control line must be defined as the basis for maximum power/flow operation and established as a license restriction. Since the transient and safety analyses with a reduced scram reactivity insertion rate are based on these power/flow lines defined by the 92% power/100% flow points, operation above these lines could result in calculated transients that violate the MCPR and pressure safety limits.

Operation, therefore, is restricted to power/flow conditions along or below these derated flow control lines which are consistent with the rod patterns necessary to give the derated power levels at 100% flow. These flow control lines are shown on the power/flow map for Pilgrim Unit 1, cycle 3, in Figure 5-1 of Reference 13.

During our review of the proposed amendments, we have identified certain changes that were necessary to conform to the NRC staff's requirements. These changes have been discussed with and agreed to by representatives of Boston Edison, and they have been made.

4.1 Conclusion

We conclude that the Technical Specifications as modified are consistent with the evaluations and are acceptable.

5.0 RECIRCULATION LOOP OUT OF SERVICE

The LOCA analysis did not adequately address one loop operation, therefore, the License had been modified to prohibit continuous operation with one loop out of service. The reactor may operate for periods up to 24 hours with one recirculation loop out of service. This short period of time permits corrective action to be taken and reduces the number of unnecessary shutdowns which is consistent with other Technical Specifications. During this period the reactor will be operated within the restrictions of the thermal analysis and will be protected from fuel damage resulting from anticipated transients.

6.0 ENVIRONMENTAL CONSIDERATIONS

The proposed action would result in a reduction in power level from 100% to 92% because of non ECCS related considerations. This reduction will begin at a point in cycle 2 where the burnup is equal to 2600 MWd/t and will continue to the end of cycle 2. This 4% average reduction of power during the later portion of the cycle will not result in a significant reduction in the total energy produced during cycle 2. Therefore the action does not significantly affect the benefits of electrical power production considered in the Commission's Final Environmental Statement (Commission's FES) for the Pilgrim Nuclear Power Station Unit No. 1 Docket No. 50-293 dated May 1972.

The Commission's staff has evaluated the potential for environmental impact associated with operation of Pilgrim Unit 1 in the proposed manner. From this evaluation, the staff had determined that there will be no change in effluent types or total amounts, no significant change in authorized power level, and no significant environmental impact attributable to the proposed action. Having made this determination, the Commission has further concluded pursuant to 10 CFR Section 51.5(c)(1) that no environmental impact statement need be prepared for this action. A Negative Declaration and supporting Environmental Impact Appraisal are being issued with this amendment to the license. As required by Part 51, the Negative Declaration is being filed with the Office of the Federal Register for publication.

7.0 CONCLUSION

Based on our evaluation of reactor operation with Reload-2 fuel, we have concluded that because this change does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the change does not involve a significant hazards consideration and that there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner. Based on our evaluation of operating limits based upon GETAB and on an acceptable ECCS evaluation model, we have concluded that there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner. We have also concluded, based on the considerations discussed in this evaluation, that all of the activities discussed herein will be conducted in compliance with the Commission's regulations and that the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: MAY 21 1976

TABLE 1

COMPARISON OF PARAMETERS FOR 8 X 8 AND 7 X 7

ROD FUEL ASSEMBLY DESIGN

	7 x 7	8 X 8
Pellet Outside Diameter (in.)	0.477	0.416
Rod Outside Diameter (in.)	0.563	0.493
Rod-to-Rod Pitch (in.)	0.738	0.640
Water-Fuel Ratio (cold)	2.53	2.60
U Bundle Weight (pounds)	412.8	404.6
Cladding Thickness (mils)	37	34
Active Fuel Length (in.)	144	144

NEGATIVE DECLARATION

REGARDING PROPOSED CHANGES TO THE

TECHNICAL SPECIFICATIONS OF LICENSE NO. DPR-35

PILGRIM NUCLEAR POWER STATION UNIT NO. 1

DOCKET NO. 50-293

The Nuclear Regulatory Commission (the Commission) has considered the issuance of changes to the Technical Specifications of Facility Operating License No. DPR-35. These changes would authorize the Boston Edison Company (the licensee) to operate the Pilgrim Nuclear Power Station Unit No. 1 (located in Plymouth County, Massachusetts) with changes to the limiting conditions for operation associated with fuel assembly specific power (average planar linear heat generation rate) resulting from application of the Acceptance Criteria for Emergency Core Cooling System (ECCS). This change is being made in conjunction with refueling with additional 8 x 8 fuel.

The U. S. Nuclear Regulatory Commission, Division of Operating Reactors, has prepared an environmental impact appraisal for the proposed changes to the Technical Specifications of License No. DPR-35, Pilgrim Unit No. 1, described above. On the basis of this appraisal, the Commission has concluded that an environmental impact statement for this particular action is not warranted because there will be no environmental impact attributable to the proposed action other than that which has already been predicted and described in the Commission's Final Environmental Statement for the Pilgrim Nuclear Power Station Unit No. 1 published in May 1972.

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The environmental impact appraisal is available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C., and at the Plymouth Public Library on North Street, Plymouth, Massachusetts 02360.

Dated at Bethesda, Maryland, this 21st day of May, 1976.

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by
Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors

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DATE➤						

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

ENVIRONMENTAL IMPACT APPRAISAL BY THE DIVISION OF OPERATING REACTORS SUPPORTING AMENDMENT NO. 15 TO DPR-35

BOSTON EDISON COMPANY

PILGRIM NUCLEAR POWER STATION UNIT NO. 1

ENVIRONMENTAL IMPACT APPRAISAL

1. Description of Proposed Action

By letters dated July 9, 1975 and July 29, 1975 and supplements thereto dated October 3, October 31, November 10, November 17, December 8, 1975 and March 1, March 19, and April 12, 1976, the Boston Edison Company (BECo) submitted proposed changes to the Technical Specifications Appendix A to License No. DPR-35. The proposed changes were requested to incorporate limiting conditions for operation associated with fuel assembly specific power (average planar linear heat generation rate) resulting from the application of the Acceptance Criteria for Emergency Core Cooling System (ECCS). The NRC staff has reviewed this proposed action to determine whether any environmental impact is associated with these proposed changes and the conclusions are set forth below.

The licensee is presently licensed to possess and operate Pilgrim Nuclear Power Station Unit No. 1 located in the State of Massachusetts, County of Plymouth, at power levels up to 1,998 megawatt thermal (MWt) using a core consisting of 7 x 7 and 8 x 8 fuel assemblies (containing U-235). The proposed change to incorporate the ECCS Acceptance Criteria does not result in an increase or decrease in power levels of the unit. The restrictions on heat generation rates will require careful control of fuel operating history. However, there should be no reduction on total burnup resulting from the revised ECCS evaluation methods. Since neither power level nor fuel burnup is affected by the action, the action does not affect the benefits of electric power production considered in the Commission's Final Environmental Statement (FES) for Pilgrim Nuclear Power Station, Docket No. 50-293 dated May 1972.

The proposed action would result in a reduction in power level from 100% to 92% because of non ECCS related considerations. This reduction will begin at a point in cycle 2 where the burnup is equal to 2600 MWd/t and will continue to the end of cycle 2. This 4% average reduction of power during the later portion of the cycle will not result in a significant reduction in the total energy produced during cycle 2. Therefore the action does not significantly affect the benefits of electrical power production considered in the Commission's Final Environmental Statement (Commission's FES) for the Pilgrim Nuclear Power Station Unit No. 1 Docket No. 50-293 dated May 1972.

2. Environmental Impacts of Proposed Action

Potential environmental impacts associated with the proposed action are those which may be associated with incorporation of the ECCS Acceptance Criteria and utilization of nuclear fuel for this facility.

It is particularly noted that in the absence of any significant change in power levels, there will be no change in cooling water requirements and consequently no increase in environmental impact from radioactive effluents and thermal effluents for normal operation or post-accident conditions which in turn could not lead to significant increases in radiation doses to the public or thermal stress to biota in the environment.

For normal operating conditions, no environmental impact other than as described in the Commission's Final Environmental Statement (FES) for Pilgrim Nuclear Power Station, Docket No. 50-293 dated May 1972, can be predicted for the proposed action. The Commission's calculated releases for radioactive effluents, both gaseous and liquid, are based on expected release rates to the environment and are quantified on the basis of the total quantity of nuclear fuel within the reactor. The estimates of radionuclides and release rates will not be affected by the proposed action, and since the total quantity of nuclear fuel is unchanged, no increase in the calculated release of radioactive effluents is predicted. Consequently, no increases in radiation doses to man or other biota are predicted.

3. Conclusion and Basis for Negative Declaration

On the basis of the foregoing analysis, it is concluded that there will be no environmental impact attributable to the proposed action other than has already been predicted and described in the Commission's FES for Pilgrim Nuclear Power Station Unit No. 1. Having made this conclusion, the Commission has further concluded that no environmental impact statement for the proposed action need be prepared and that a negative declaration to this effect is appropriate.

Date: **MAY 21 1976**

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-293

BOSTON EDISON COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

Notice is hereby given that the U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 15 to Facility Operating License No. DPR-35, issued to Boston Edison Company (the licensee), which revised Technical Specifications for operation of Unit No. 1 of the Pilgrim Nuclear Power Station (the facility) located near Plymouth, Massachusetts. The amendment is effective as of its date of issuance.

The amendment (1) authorizes operation with additional 8 x 8 fuel assemblies, (2) establishes operating limits based upon the General Electric Thermal Analysis Basis (GETAB), and (3) incorporates operating limits in the Technical Specifications for the facility based on an acceptable evaluation model that conforms with the requirements of Section 50.46 of 10 CFR Part 50.

The applications for the amendment comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Notice of Proposed Issuance of Amendment to Facility Operating License in connection with items (2) and (3) above was published in the FEDERAL REGISTER on

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October 17, 1975 (40 FR 48735). No request for a hearing or petition for leave to intervene was filed following notice of the proposed action on items (2) and (3) above. Prior public notice of item (1) above was not required since this action does not involve a significant hazards consideration.

In connection with the issuance of this amendment, the Commission has issued a Negative Declaration and Environmental Impact Appraisal.

For further details with respect to this action, see (1) the applications for amendment dated July 9, 1975 and July 29, 1975, and supplements thereto dated October 3, October 31, November 10, November 17, December 8, 1975 and March 1, March 19, and April 12, 1976, (2) Amendment No. 15 to License No. DPR-35, (3) the Commission's concurrently issued related Safety Evaluation, and (4) the Commission's Negative Declaration dated May 21, 1976 (which is also being published in the FEDERAL REGISTER), and associated Environmental Impact Appraisal. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Plymouth Public Library on North Street in Plymouth, Massachusetts 02360.

A single copy of items (2) through (4) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 21st day of May, 1976

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by
Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors

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SURNAME ➤						
DATE ➤						

May 20, 1976

Note to Paul O'Connor
Project Manager for Pilgrim Station Unit No. 1

CONCURRENCE, SUBJECT TO CONDITIONS - LICENSE AMENDMENT NO. 15 TO PILGRIM UNIT NO. 1 - FUEL ASSEMBLIES AND OPERATING LIMITS

I concur in the issuance of Amendment No. 15 to the Pilgrim Unit No. 1 operating license with the following conditions:

1. The reference to channel box integrity on page 5 of the SER should be eliminated.

2. The environmental appraisal be made consistent with the amendment.

The SER presently contains the following language:

> "...[w]e conclude that for Pilgrim Unit 1 Reload 2:
>
> 3. Fuel rod and <u>channel box integrity</u> will remain satisfactory <u>during transients</u> and accidents." [Emphasis added] Page 5, SER.

As we discussed on May 19, 1976, the conclusion made concerning the channel box integrity at best is supported in the SER by some vague references to reports filed by the Licensee and/or GE, and the conclusion is surplusage since the conclusion is not necessary to this amendment.

The Environmental Considerations on page 19 of the SER read in part:

> 6.0 <u>Environmental Considerations</u>
>
> "The Commission's Staff has evaluated the potential for environmental impact associated with operation of Pilgrim Unit 1 in the proposed manner. From this evaluation, the Staff had determined that there will be no change in effluent types or total amounts, no change in <u>authorized power level</u>...."

The Environmental Impact Appraisal reads in part:

> "The proposed change to incorporate the ECCS Acceptance Criteria does not result in an increase or decrease in power levels of the Unit."

These statements are not consistent with the operating restrictions found in the amendment. To correct this situation, it is necessary for you to state that there is a change in the power level but it is not significant and explain why.

Barry H Smith

Barry H. Smith
Attorney, OELD