Docket No. 50-293

August 31, 1987

Boston Edison Company
ATTN: Mr. Ralph E. Bird
Senior Vice President - Nuclear
800 Boylston Street
Boston, Massachusetts 02199

Dear Mr. Bird:

SUBJECT: ISSUANCE OF AMENDMENT NO. 105 TO FACILITY OPERATING LICENSE NO. DPR-35 - PILGRIM NUCLEAR POWER STATION (TAC #65491)

The Commission has issued the enclosed Amendment No. 105 to Facility Operating License No. DPR-35 for the Pilgrim Nuclear Power Station. This amendment consists of changes to the Technical Specifications in response to your application dated May 22, 1987 as supplemented by your letter, dated July 22, 1987.

This amendment modifies the Pilgrim Station Technical Specifications to ensure safe operation during Cycle 8 following core reload 7. For the first time, only retrofit fuel will be loaded into the core and all non-retrofit fuel will be discharged to the spent fuel pool.

A copy of the related Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's Bi-Weekly Federal Register Notice.

Sincerely,

/s/

Richard H. Wessman, Senior Project Manager
Project Directorate I-3
Division of Reactor Projects I/II

Enclosures:
1. Amendment No. 105 to License No. DPR-35.
2. Safety Evaluation

cc w/enclosures:
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SUBJECT: AMENDMENT NO. 105 TO PILGRIM FACILITY OPERATING LICENSE DPR-35

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

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Richard H. Wessman, Senior Project Manager
Project Directorate I-3
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See next page

Mr. Ralph G. Bird
Boston Edison Company

Pilgrim Nuclear Power Station

cc:

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

BOSTON EDISON COMPANY

DOCKET NO. 50-293

PILGRIM NUCLEAR POWER STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 105
License No. DPR-35

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

   A. The application for amendment by Boston Edison Company (the licensee) dated May 22, 1987 as supplemented by letter dated July 22, 1987, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 3.B. of Facility Operating License No. DPR-35 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 105, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Carl Stahle

for Victor Nerses, Acting Director
Project Directorate I-3
Division of Reactor Projects I/II

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 31, 1987

ATTACHMENT TO LICENSE AMENDMENT NO. 105

FACILTIY OPERATING LICENSE NO. DPR-35

DOCKET NO. 50-293

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change. The corresponding overleaf pages are provided to maintain document completeness.

Remove Pages	Insert Pages
7	7
8	8
46	46
53	53
59a	59a
68	68
203	203
205A-1	205A-1
205B-2	205B-2
205C-3	205C-3
205E-1	205E-1
205E-2	205E-2
205E-3	205E-3
205E-4	205E-4
205E-5	205E-5
205E-6	205E-6
205F	205F
206m	206m

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.

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

$$S \leq (0.58W + 62\%) \left[\frac{FRP}{MFLPD}\right] \underline{\text{2 Loop}}$$

Where,

FRP = fraction of rated thermal power (1998 MWt)

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

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

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

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

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

c. <u>IRM</u>

The IRM flux scram setting shall be $\leq$120/125 of scale.

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

The APRM rod block trip setting shall be:

$$S_{RB} \leq 0.58W + 50\% \quad \underline{\text{2 Loop}}$$

Where,

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

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

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

$$S \leq (0.58W + 50\%) \left[\frac{FRP}{MFLPD}\right] \quad \underline{\text{2 Loop}}$$

Where,

FRP = fraction of rated thermal power

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

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

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

D. Turbine stop valve closure scram settings 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.

NOTES FOR TABLE 3.2.A

1. Whenever Primary Containment integrity is required by Section 3.7, there shall be two operable or tripped trip systems for each function.

2. Action

   If the first column cannot be met for one of the trip systems, that trip system shall be tripped. If the first column cannot be met for both trip systems, the appropriate action listed below shall be taken.

   A. Initiate an orderly shutdown and have the reactor in Cold Shutdown Condition in 24 hours.

   B. Initiate an orderly load reduction and have Main Steam Lines isolated within eight hours.

   C. Isolate Reactor Water Cleanup System.

   D. Isolate Shutdown Cooling.

3. Instrument set point corresponds to 128.26 inches above top of active fuel.

4. Instrument set point corresponds to 77.26 inches above top of active fuel.

5. Not required in Run Mode (bypassed by Mode Switch).

6. Two required for each steam line.

7. These signals also start SBGTS and initiate secondary containment isolation.

8. Only required in Run Mode (interlocked with Mode Switch).

9. Within 24 hours prior to the planned start of hydrogen injection test with the reactor power at greater than 20% rated power, the normal full power radiation background level and associated trip setpoints may be changed based on a calculated value of the radiation level expected during the test. The background radiation level and associated trip setpoints may be adjusted during the test based on either calculations or measurements of actual radiation levels resulting from hydrogen injection. The background radiation level shall be determined and associated trip setpoints shall be set within 24 hours of re-establishing normal radiation levels after completion of hydrogen injection and prior to establishing reactor power levels below 20% rated power.

NOTES FOR TABLE 3.2.B

1. Whenever any CSCS subsystem is required by Section 3.5 to be operable, there shall be two (Note 5) operable trip systems. If the first column cannot be met for one of the trip systems, that system shall be repaired or the reactor shall be placed in the Cold Shutdown Condition within 24 hours after this trip system is made or found to be inoperable.

2. Close isolation valves in RCIC subsystem.

3. Close isolation valves in HPCI subsystem.

4. Instrument set point corresponds to 77.26 inches of active fuel.

5. RCIC and HPCI have only one trip system for these sensors.

TABLE 3.2-G

INSTRUMENTATION THAT INITIATES RECIRCULATION PUMP TRIP
AND
ALTERNATE ROD INSERTION

Minimum Number of Operable or Tripped Instrument Channels Per Trip System (1)	Trip Function	Trip Level Setting
2	High Reactor Dome Pressure	1175 $\pm$ 15 PSIG
2	Low-Low Reactor Water Level	$\geq$ 77.26 inches above the top of the active fuel

Actions (1) There shall be two (2) operable trip systems for each function.

(a) If the minimum number of operable or tripped instrument channels for one (1) trip system cannot be met, restore the trip system to operable status within 14 days or be in at least hot shutdown within 24 hours.

(b) If the minimum operability conditions (1.a) cannot be met for both (2) trip systems, be in at least hot shutdown within 24 hours.

BASES:

3.2 In addition to reactor protection instrumentation which initiates a reactor scram, protective instrumentation has been provided which initiates action to mitigate the consequences of accidents which are beyond the operator's ability to control, or terminates operator errors before they result in serious consequences. This set of specifications provides the limiting conditions of operation for the primary system isolation function, initiation of the core cooling systems, control rod block and standby gas treatment systems. The objectives of the Specifications are (i) to assure the effectiveness of the protective instrumentation when required by preserving its capability to tolerate a single failure of any component of such systems even during periods when portions of such systems are out of service for maintenance, and (ii) to prescribe the trip settings required to assure adequate performance. When necessary, one channel may be made inoperable for brief intervals to conduct required functional tests and calibrations.

Some of the settings on the instrumentation that initiate or control core and containment cooling have tolerances explicitly stated where the high and low values are both critical and may have a substantial effect on safety. The set points of other instrumentation, where only the high or low end of the setting has a direct bearing on safety, are chosen at a level away from the normal operating range to prevent inadvertent actuation of the safety system involved and exposure to abnormal situations.

Actuation of primary containment valves is initiated by protective instrumentation shown in Table 3.2.A which senses the conditions for which isolation is required. Such instrumentation must be available whenever primary containment integrity is required.

The instrumentation which initiates primary system isolation is connected in a dual bus arrangement.

The low water level instrumentation set to trip at 128.26 inches above the top of the active fuel closes all isolation valves except those in Groups 1, 4 and 5. Details of valve grouping and required closing times are given in Specification 3.7. This trip setting is adequate to prevent core uncovery in the case of a break in the largest line assuming a 60 second valve closing time. Required closing times are less than this.

The low low reactor water level instrumentation is set to trip when reactor water level is 77.26 inches above the top of the active fuel (-49" on the instrument). This trip closes Main Steam Line Isolation

2. The SRM shall have a minimum of 3 cps except as specified in 3 and 4 below.

3. Prior to spiral unloading, the SRM's shall have an initial count rate of ≥ 3 cps. During spiral unloading, the count rate on the SRM's may drop below 3 cps.

4. During spiral reload, each control cell shall have at least one assembly with a minimum exposure of 1000 MWD/ST.

Spiral Reload

During spiral reload, SRM operability will be verified by using a portable external source every 12 hours until the required amount of fuel is loaded to maintain 3 cps. As an alternative to the above, up to two fuel assemblies will be loaded in different cells containing control blades around each SRM to obtain the required 3 cps. Until these assemblies have loaded, the cps requirement is not necessary.

C. Spent Fuel Pool Water Level

Whenever irradiated fuel is stored in the spent fuel pool, the pool water level shall be maintained at or above 33 feet.

C. Spent Fuel Pool Water Level

Whenever irradiated fuel is stored in the spent fuel pool, the water level shall be recorded daily.

D. Multiple Control Rod Removal

Any number of control rods and/or control rod drive mechanisms may be removed from the reactor pressure vessel provided that at least the following requirements are satisfied until all control rods and control rod drive mechanisms are reinstalled and all control rods are fully inserted in the core.

a. The reactor mode switch is operable and locked in the Refuel position per Specification 3.10.A, except that the Refuel position "one rod out" interlock may be bypassed, as required, for those control rods and/or control rod drive mechanisms to be removed, after the fuel assemblies have been removed as specified below.

b. The source range monitors (SRM) are operable per Specification 3.3.B.4.

c. The Reactivity Margin requirements of Specification 3.3.A.1 are satisfied.

D. Multiple Control Rod Removal

Within 4 hours prior to the start of removal of control rods and/or control rod drive mechanisms from the core and/or reactor pressure vessel and at least once per 24 hours thereafter until all control rods and control rod drive mechanisms are reinstalled and all control rods are fully inserted in the core, verify that:

a. The reactor mode switch is operable and locked in the Refuel position per Specification 3.10.A.

b. The SRM channels are operable per Specification 3.3.B.4.

c. The Reactivity Margin requirements of Specification 3.3.A.1 are satisfied.

LIMITING CONDITIONS FOR OPERATION

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 13.4 kw/ft for all fuel.

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.

SURVEILLANCE REQUIREMENTS

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.

TABLE 3.11-1
OPERATING LIMIT MCPR VALUES

A. MCPR Operating Limit from Beginning of Cycle (BOC) to BOC + 7,513 MWD/ST.

	P8x8R/BP8x8R
For all values of τ	1.36

B. MCPR Operating Limit from BOC + 7,513 MWD/ST to End of Cycle.

τ	P8x8R/BP8x8R
$\tau \leq 0$	1.39
$0.0 < \tau \leq 0.1$	1.40
$0.1 < \tau \leq 0.2$	1.40
$0.2 < \tau \leq 0.3$	1.41
$0.3 < \tau \leq 0.4$	1.41
$0.4 < \tau \leq 0.5$	1.42
$0.5 < \tau \leq 0.6$	1.42
$0.6 < \tau \leq 0.7$	1.43
$0.7 < \tau \leq 0.8$	1.43
$0.8 < \tau \leq 0.9$	1.44
$0.9 < \tau \leq 1.0$	1.44

BASES:

3.11C MINIMUM CRITICAL POWER RATIO (MCPR)

Operating Limit MCPR

For any abnormal operating transient analysis evaluation with the initial condition of the reactor being at the steady state operating limit, it is required that the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient assuming instrument trip setting given in Specification 2.1.

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

The evaluation of a given transient begins with the system input parameters shown in Table 5-4, 5-6 and 5-8 of NEDE-24011-P[1], Supplemented by reload unique inputs given in the current Supplemental Reload Licensing Submittal. These values are input to a GE core dynamic behavior transient computer program described in NEDO-10802[2]. The transient code used for all pressurization events is described in NEDE-24154-P (Reference 5). The MCPR analysis for pressurization events is done in accordance with the procedures given in Reference 6.

FIGURE 3.11-1

MAPLHGR Versus Planar Average Exposure

Fuel Type 8DB219L

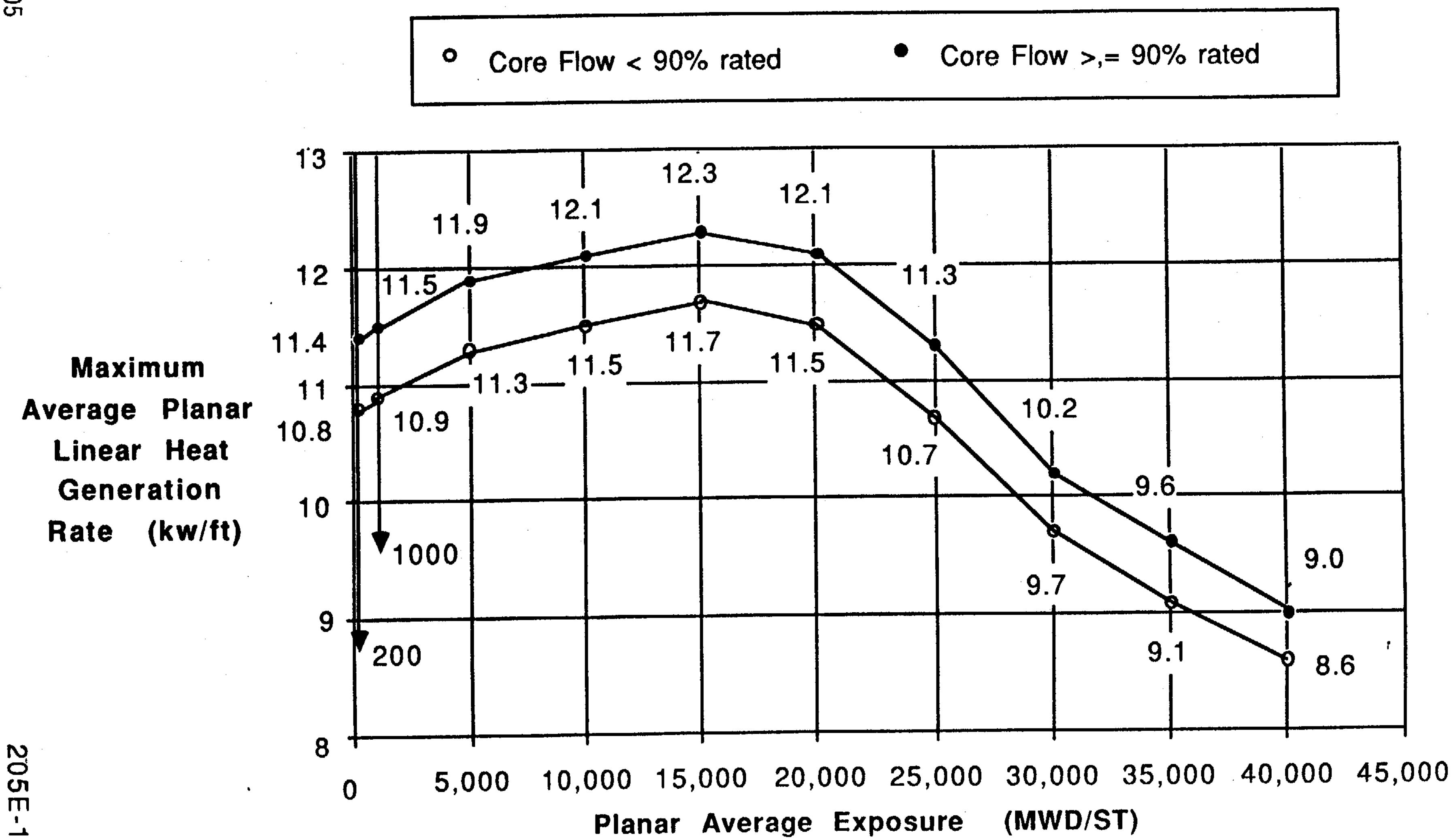
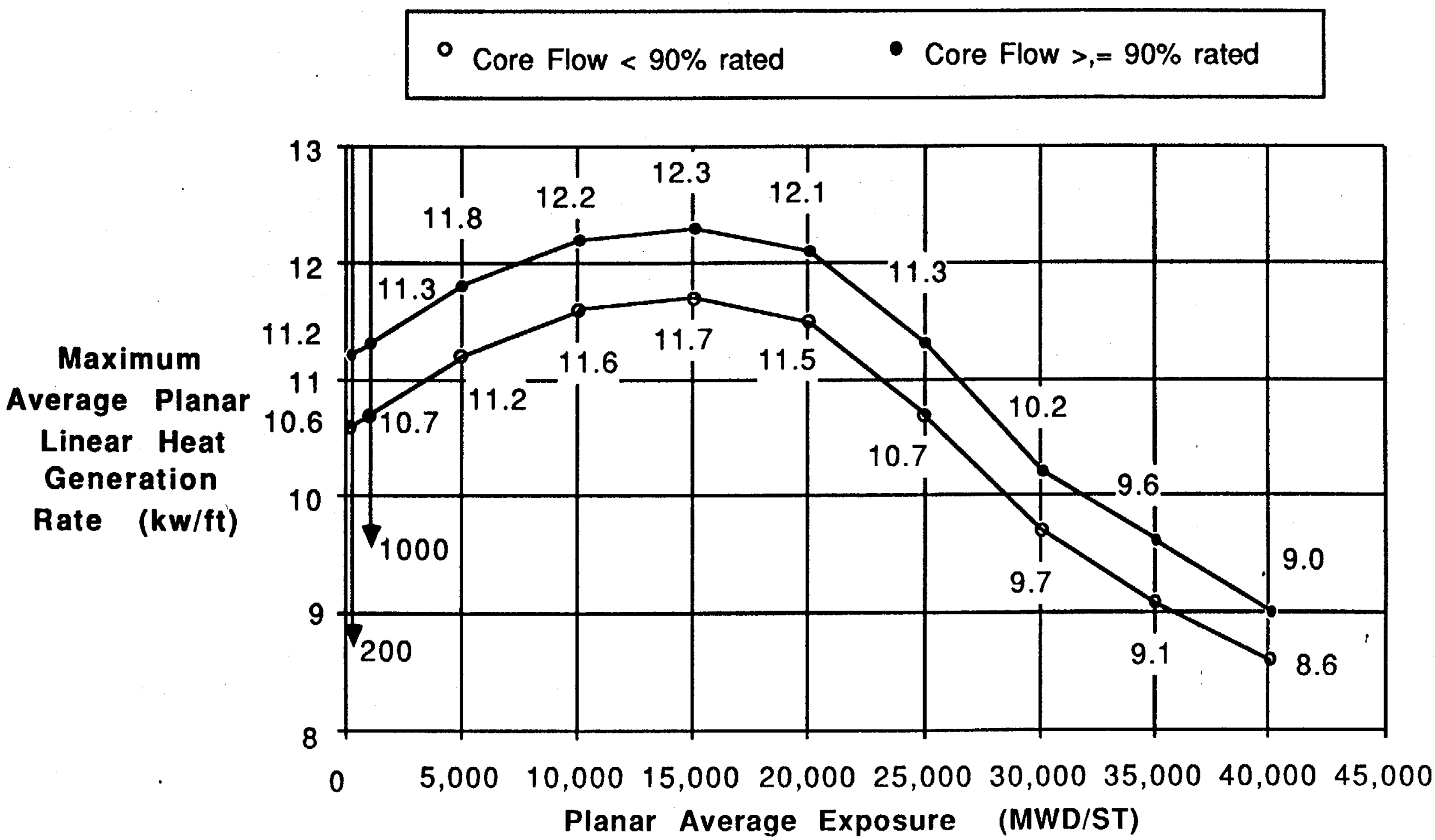


FIGURE 3.11-2

MAPLHGR Versus Planar Average Exposure

Fuel Type 8DB219H



Amendment No. 89, 105

FIGURE 3.11-3

MAPLHGR Versus Planar Average Exposure

Fuel Type 8DB262

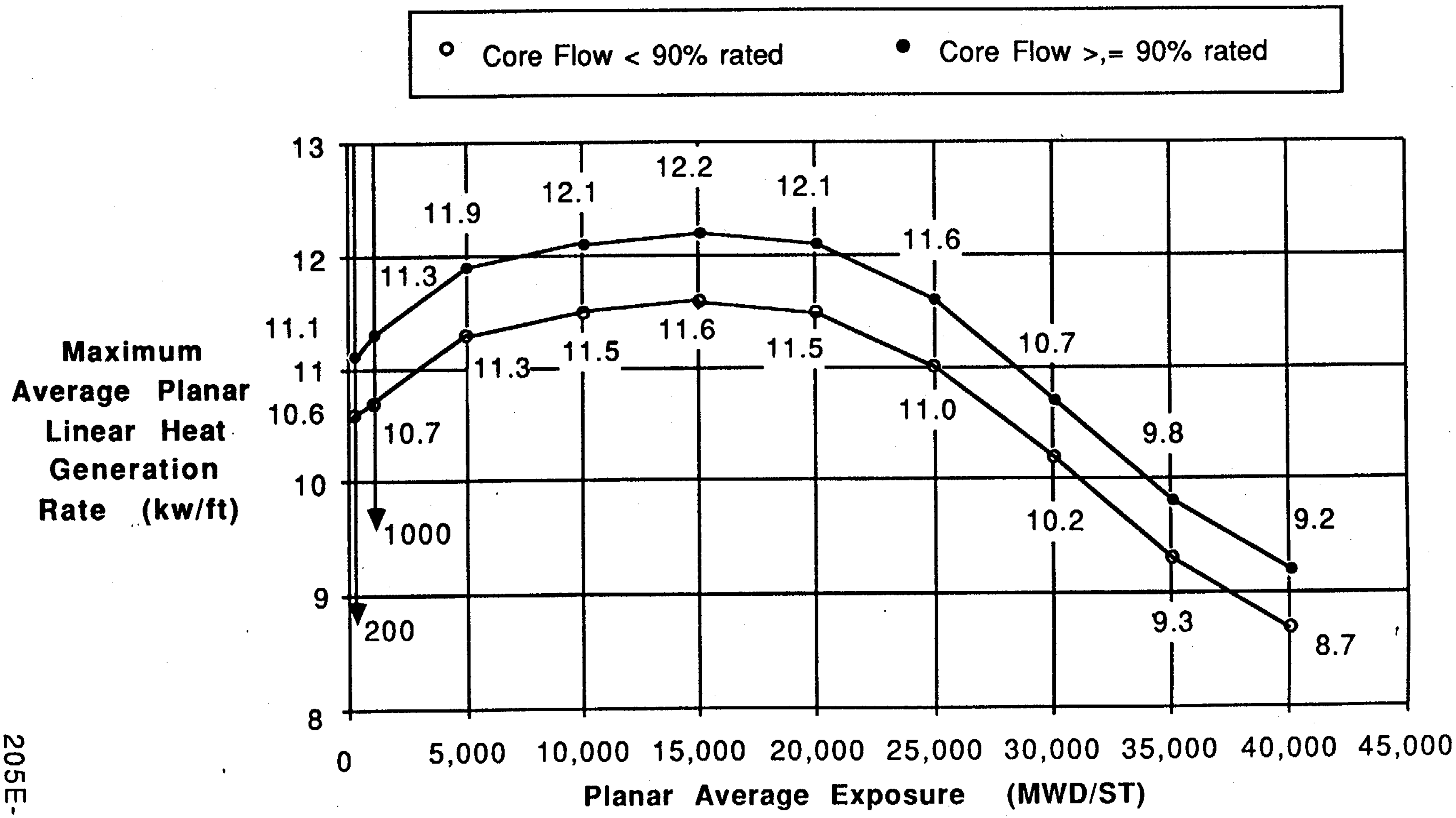


FIGURE 3.11-4

MAPLHGR Versus Planar Average Exposure

Fuel Types P8DRB265L and BPDRB265L

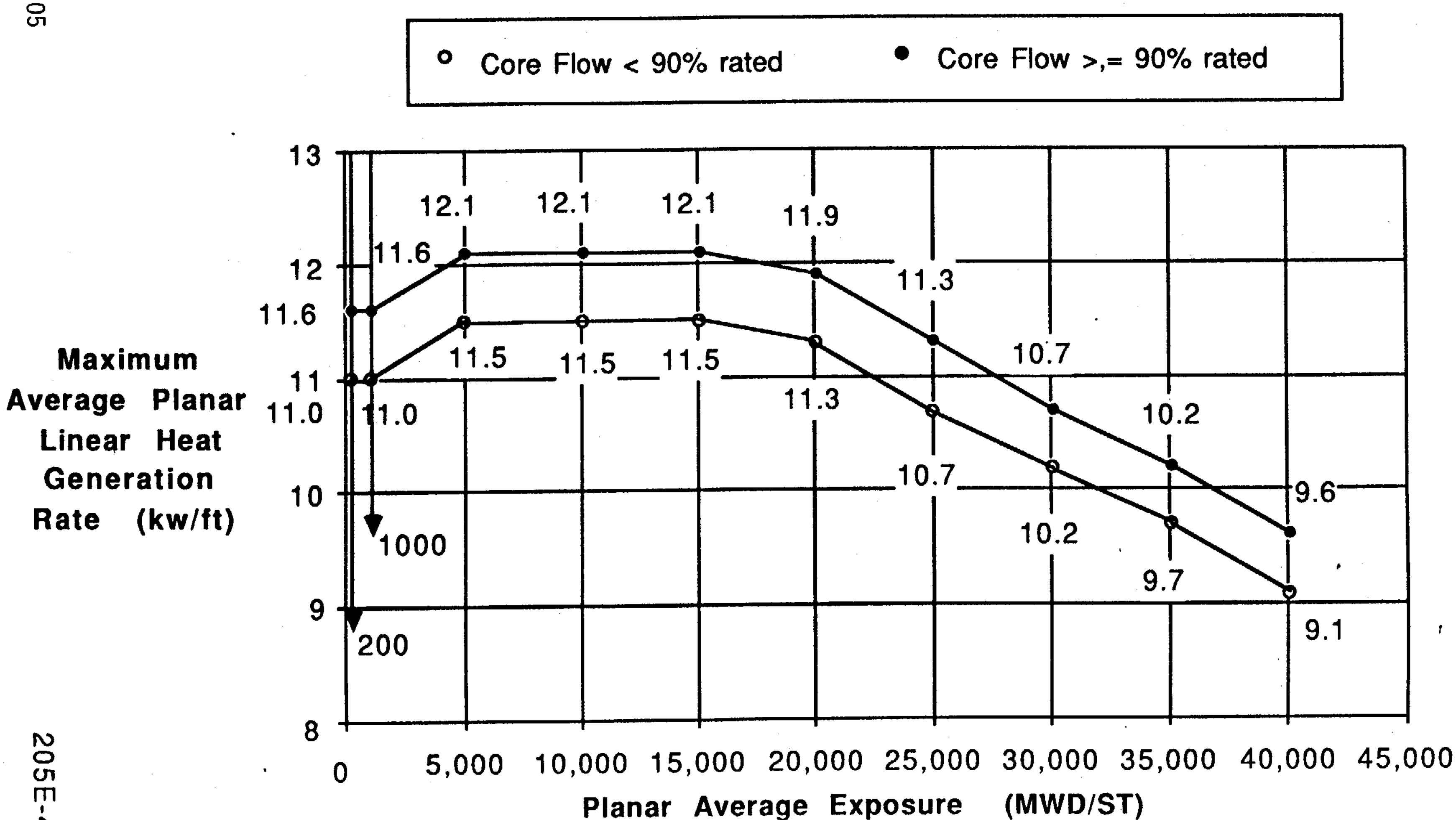


FIGURE 3.11-5

MAPLHGR Versus Planar Average Exposure

Fuel Types P8DRB282 and BP8DRB282

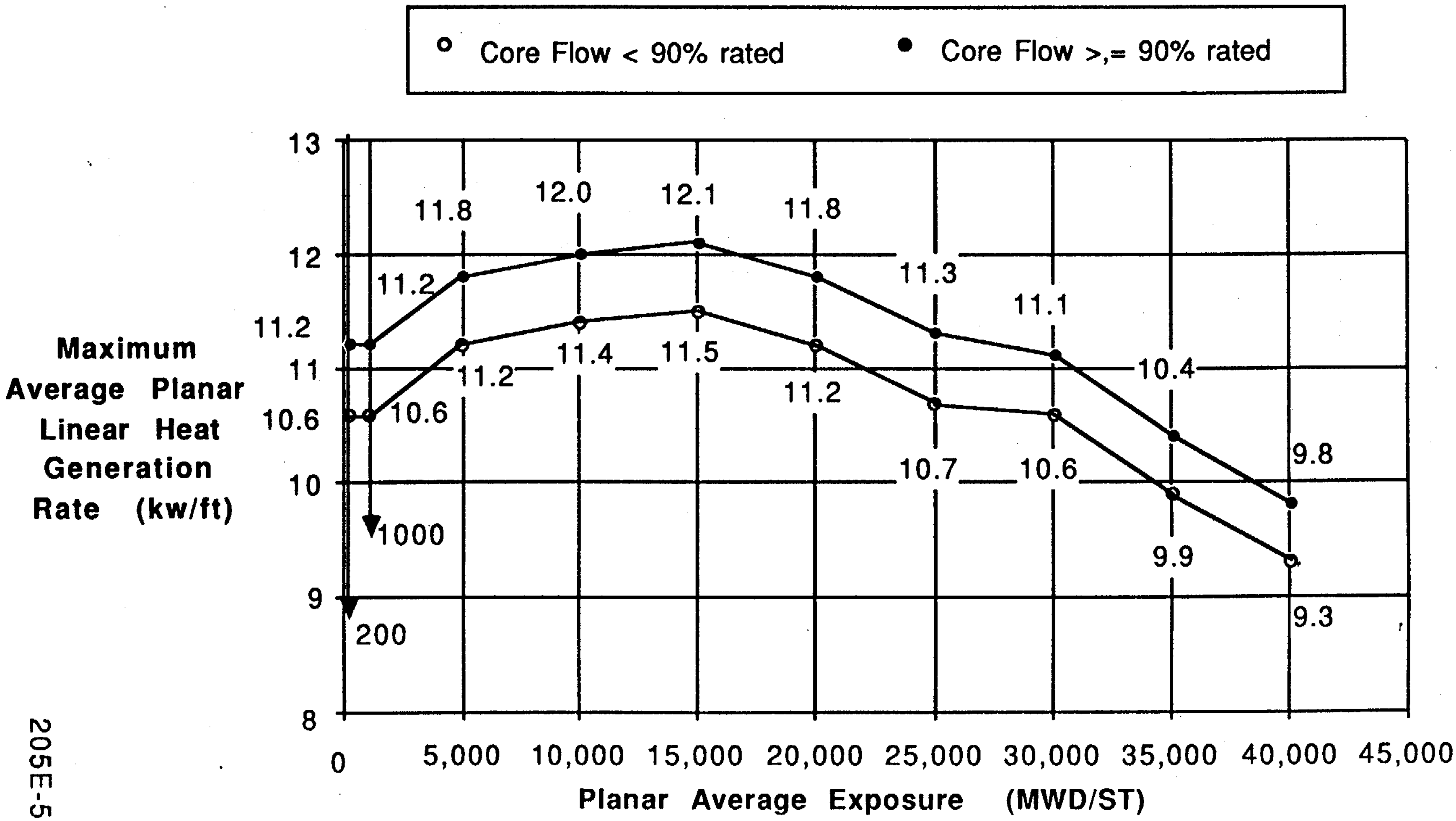
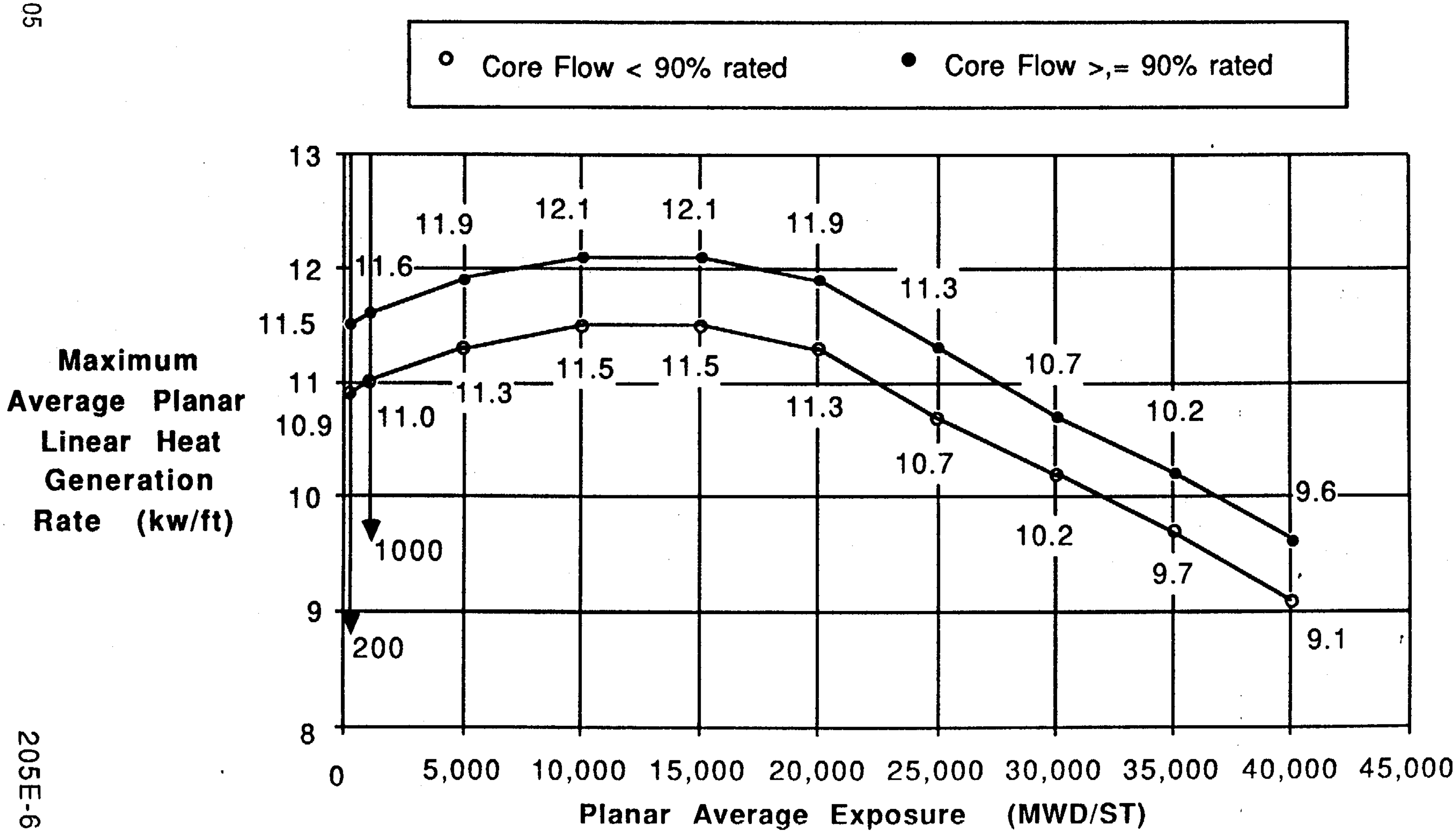


FIGURE 3.11-6

MAPLHGR Versus Planar Average Exposure

Fuel Types P8DRB265H and BP8DRB265H

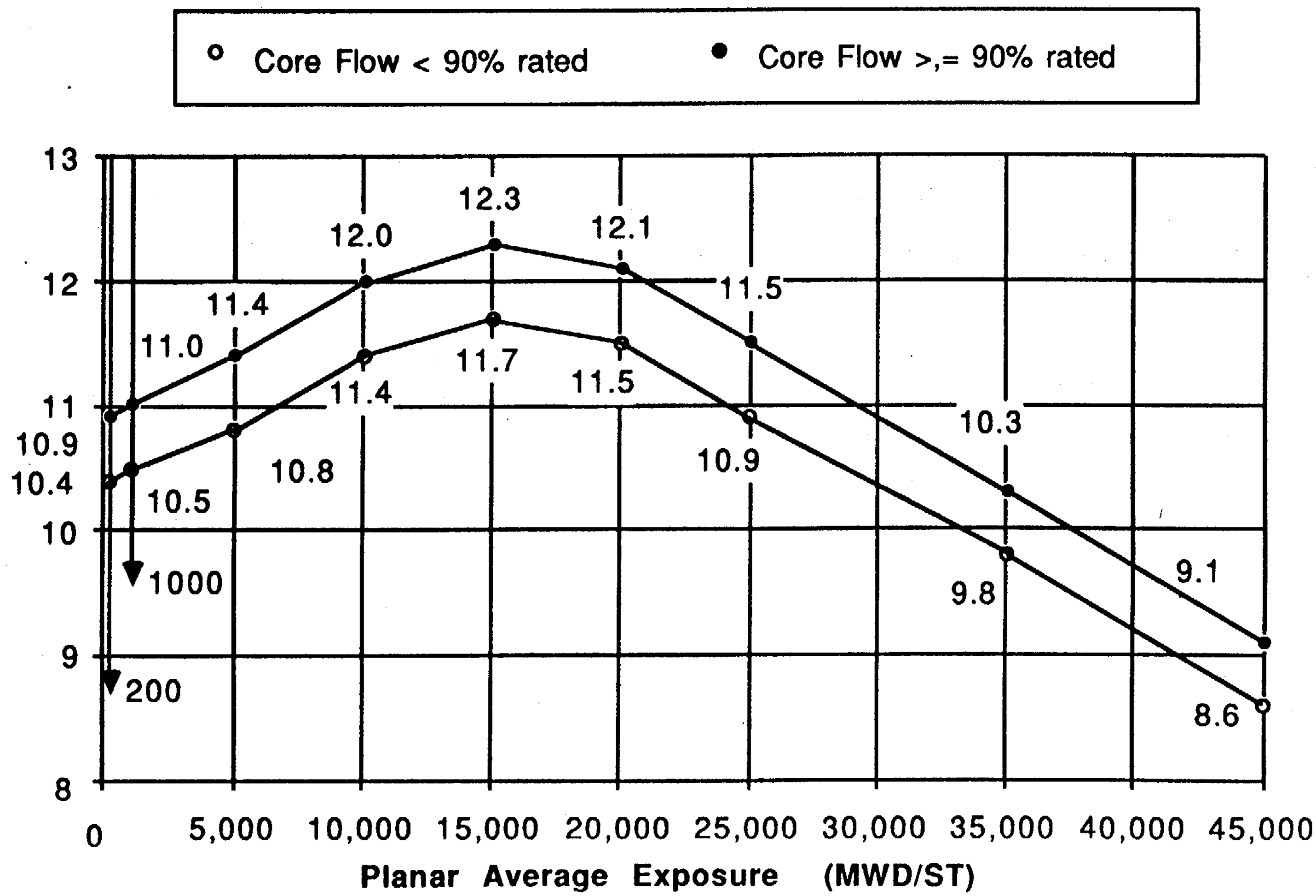


Amendment No. 42, 100, 105

FIGURE 3.11-7

MAPLHGR Versus Planar Average Exposure

Fuel Type BP8DRB300



5.0 MAJOR DESIGN FEATUR

5.1 SITE FEATURES

Pilgrim Nuclear Power Station is located on the Western Shore of Cape Cod Bay in the Town of Plymouth, Plymouth County, Massachusetts. The site is located at approximately 41°51' north latitude and 70°35' west longitude on the Manomet Quadrangle, Massachusetts, Plymouth County 7.5 Minute Series (topographic) map issued by U.S. Geological Survey. UTM coordinates are 19-46446N-3692E.

The reactor (center line) is located approximately 1800 feet from the nearest property boundary.

5.2 REACTOR

A. The core shall consist of not more than 580 fuel assemblies.

B. The reactor core shall contain 145 cruciform-shaped control rods. The control materials shall be either boron carbide powder (B_4C) compacted to approximately 70% of theoretical density or a combination of boron carbide powder and solid hafnium.

5.3 REACTOR VESSEL

The reactor vessel shall be as described in Table 4.2.2 of the FSAR. The applicable design codes shall be as described in Table 4.2.1 of the FSAR.

5.4 CONTAINMENT

A. The principal design parameters for the primary containment shall be as given in Table 5.2.1 of the FSAR. The applicable design codes shall be as described in Section 12.2.2.8 of the FSAR.

B. The secondary containment shall be as described in Section 5.3.2 of the FSAR.

C. Penetrations to the primary containment and piping passing through such penetrations shall be designed in accordance with standards set forth in Section 5.2.3.4 of the FSAR.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO LICENSE AMENDMENT FOR CYCLE 8 (RELOAD 7)

LICENSE NO. NPF-35

BOSTON EDISON COMPANY

PILGRIM NUCLEAR POWER STATION

DOCKET NO. 50-293

1.0 INTRODUCTION

By letter from R. G. Bird, Boston Edison Company (BECo) to U. S. Nuclear Regulatory Commission dated May 22, 1987 (Ref. 1), Technical Specification changes were proposed for the operation of Pilgrim Nuclear Power Station for Cycle 8 (Reload 7) (designated herein as PSC8) with a reload using General Electric (GE) manufactured fuel assemblies and GE analyses and methodologies. The requested Technical Specification (TS) changes and reports (including Reference 2) discussing the reload and analyses done to support and justify the Reload 7 operation were included with the submittal. Additional information regarding thermal-hydraulic stability was provided by the licensee in Reference 6. Some editorial changes to the TS were also requested as part of the proposed amendment.

2.0 EVALUATION

2.1 RELOAD DESCRIPTION

The Pilgrim reload will retain 388 assorted GE P8x8 retrofit fuel assemblies from the previous cycles and add 192 new BP8DRB300 GE fuel assemblies. The reload is based on a previous cycle core average exposure of 16658 Megawatt Days per Standard Ton (MWD/ST) and a Cycle 8 end of cycle exposure of 18682 MWD/ST. The loading will be a conventional scatter pattern with low reactivity fuel on the periphery.

2.2 FUEL DESIGN

The new fuel assembly to be used for PSC8, type BP8DRB300, has been approved for inclusion in NEDE-24011, GESTAR II (Ref. 3). This fuel type has been analyzed for this application with approved methods and meets the approved limits of GESTAR II (Ref. 3), therefore the fuel is acceptable for PSC8. The design analysis for the BP8DRB300 fuel and TS changes related to MAPLHGR curves for the additional fuel assemblies have been addressed and approved in the NRC Safety Evaluation for Amendment No. 100 to the Pilgrim Facility Operating License (Ref. 4).

2.3 NUCLEAR DESIGN

The nuclear design for PSC8 has been performed with methodology described in GESTAR II (Ref. 3). The results of those analyses are given in Reference 2. The shutdown margin (SDM) is 3.8 % delta-k/k at the beginning of cycle and 1.0 % delta-k/k at the exposure of minimum shutdown margin. Therefore, it meets the required 0.29 % delta-k/k shutdown margin. The standby liquid control system also meets shutdown requirements with a shutdown margin of 4.4% delta-k/k. Since these and other PSC8 nuclear design parameters have been obtained with previously approved methods and fall within expected ranges, the nuclear design is acceptable.

The description of low and low-low water level setpoints with respect to the top of the active fuel are revised to reflect the different dimensional length of the retrofit fuel. The trip setting descriptions which are affected are for trips which close isolation valves in certain process lines which penetrate containment. The water level trip settings are adequate to prevent core uncovery in the case of a break in the largest of these lines assuming a 60 second valve closing time. The required closing time for the isolation valves in these lines is less than 60 seconds. Therefore the proposed change does not significantly alter the margin to core uncovery and is acceptable.

2.4 THERMAL HYDRAULIC DESIGN

The thermal-hydraulic design for PSC8 has been performed with the methodology described in GESTAR II (Ref. 3) and the results are given in Reference 2. The parameters used for the analyses are those approved in Reference 3 for the Pilgrim BWR/3 product line.

The Operating Limit Minimum Critical Power Ratio (OLMCPR) values are determined by analysis of the limiting transients, Rod Withdrawal Error (RWE), Feedwater Controller Failure (FWCF) and Load Rejection Without Bypass (LRWBP). The analysis of these events for PSC8, via the ODYN Option A and B approach, provide new Cycle 8 Technical Specification values of OLMCPR as a function of average scram time and exposure ranges. Two exposure ranges from beginning of cycle (BOC) to end of cycle (EOC) were analyzed, 1) BOC to BOC + 7513 MWD/ST and 2) BOC + 7513 MWD/ST to EOC. For standard operating conditions, LRWBP is controlling at both option A and B limits. The resulting OLMCPR values are reflected in the proposed TS changes. For exposure range 1), the OLMCPR is a constant value of 1.36 for all values of average scram speed; for exposure range 2), the OLMCPR varies from 1.39 to a maximum value of 1.44 as a function of average scram time. Approved methods (Ref. 3) were used to analyze the limiting transients and the results fall within expected ranges and are acceptable.

The thermal-hydraulic stability of the Cycle 8 core has been analyzed using approved methods (Ref. 6). The result is a decay ratio of 0.64 at the intersection of the natural circulation line and the 100 percent rod line. Existing technical specifications do not allow continued operation in natural circulation. Operation at the combination of low flow and high power sufficient to produce a high decay ratio is thus limited. Based on the similarity of the Cycle 8 decay ratio to the previously evaluated Cycle 7 (reported as 0.65), we conclude that appropriate consideration has been given to compliance with 10 CFR 50 Appendix A General Design Criterion 12 (Suppression of reactor power oscillations) and the licensee's submittal is responsive to NRC Generic Letter 86-02 (Ref. 7).

2.5 TRANSIENT AND ACCIDENT ANALYSES

The transient and accident analysis methodologies used for PSC8 are described and NRC approval indicated in GESTAR II (Ref. 3). Generally, the ODYN Option

A and B approach was used for transient analyses. The Loss of Feedwater Heating event was analyzed with the GE BWR Simulator code, approved in Reference 3. The limiting MCPR events have been previously indicated in Section 2.4 of this SE. The core wide transient analysis methodologies are acceptable and the results fall within expected ranges.

The RWE was analyzed with a generic bounding analysis and a rod block setpoint of 1.07 was selected to provide an OLMCPR of 1.29 for the retrofit fuel types. The fuel assembly misorientation event was analyzed with standard methods for the PSC8 D lattice fuel, giving a non-limiting MCPR of 1.26. As approved in Reference 3, the mislocated assembly is not analyzed for reload cores on the basis of studies indicating the small probability of an event exceeding MCPR limits.

The limiting pressurization event, the Main Steam Isolation Valve Closure with Flux Scram, was analyzed with standard GESTAR II methods. Results for peak steam dome and vessel pressures were well under required limits. These are acceptable methodologies and results.

LOCA analyses, using approved methodologies and parameters (Ref. 3), were performed to provide MAPLHGR values for the new reload fuel assemblies (BP8DRB300). These analyses and results are acceptable. The resulting LOCA results and MAPLHGR values are documented in Attachment 4 (Addendum to Ref. 5) of the base Reload 7 submittal (Ref. 1).

2.6 TECHNICAL SPECIFICATIONS

The Technical Specification changes are for the most part minor and provide for MCPR changes due to Cycle 8 parameter changes, MAPLHGR values for the new fuel assemblies, and editorial changes. Details of the specification changes follow:

(1) Specification 2.1, LCO pages 203, 205A-1, 205B-2, 205C-2, and Major Design Features page 206m

References to non-retrofit fuel are removed since this fuel type is not to be used in Cycle 8. These changes are acceptable.

(2) Pages 46, 53, 59a and 68

Descriptions of water level setpoints are changed to reflect dimensional changes for retrofit fuel assemblies. The changes are acceptable as discussed in Section 2.3.

(3) Table 3.11-1, page 205B-2 and Bases page 205C-3

Operating limit MCPR values are revised for Cycle 8 operation. The revised information is acceptable as discussed in Sections 2.4 and 2.5.

(4) Figures 3.11-1 through 3.11-7: MAPLHGR versus Planar Average Exposure Curves

A new MAPLHGR curve is provided for the new fuel. A standard unit of fuel exposure, megawatt days per standard ton (MWD/ST), was established as uniform nomenclature. These changes are acceptable as discussed in Section 2.5.

ENVIRONMENTAL CONSIDERATION

This amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

3.0 CONCLUSION

As a result of our review, which is described in Section 2.0 of this evaluation, we conclude that the proposed reload and technical specification changes are acceptable.

The staff has concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the health and safety of the public.

Date: August 31, 1987

Principal Contributor: Michael McCoy

4.0 REFERENCES

1. Letter from R. G. Bird, BECo, to US NRC, dated May 22, 1987 "Reload 7 Licensing Submittal and Proposed Change to Technical Specifications (with Attachments)

2. GE Report 23A4800, dated December 1986, "Supplemental Reload Licensing Submittal for Pilgrim Nuclear Power Station, Reload 7"

3. NEDE-24011-P-A-8, May 1986, "General Electric Standard Application for Reactor Fuel," (GESTAR II)

4. Safety Evaluation by the Office of Nuclear Reactor Regulation Supporting Amendment No. 100 to Facility Operating License No. DPR-35, Pilgrim Nuclear Power Station, April 9, 1987

5. NEDO-30767, September 1984, "Pilgrim Nuclear Power Station Loss of Coolant Accident (LOCA) Analysis Update"

6. Letter, R. G. Bird BECo, to U. S. NRC, dated July 22, 1987 "Results of Thermal-Hydraulic Stability Analysis for Cycle 8 Operation".

7. Technical Resolution of Generic Issue B-19 - Thermal Hydraulic Stability (Generic Letter NO. 86-02), dated January 23, 1986.