April 13, 2001

Mr. Mike Bellamy
Site Vice President
Entergy Nuclear Generation Company
Pilgrim Nuclear Power Station
600 Rocky Hill Road
Plymouth, MA 02360

SUBJECT: PILGRIM NUCLEAR POWER STATION - ISSUANCE OF AMENDMENT RE: PRESSURE-TEMPERATURE LIMIT CURVES (TAC NO. MB0561)

Dear Mr. Bellamy:

The Commission has issued the enclosed Amendment No. 190 to Facility Operating License No. DPR-35 for the Pilgrim Nuclear Power Station. This amendment is in response to your application dated November 22, 2000, as supplemented on January 30 and February 2, 2001.

This amendment will change the pressure-temperature limit curves of Figures 3.6.1, 3.6.2, and 3.6.3 of Pilgrim's Technical Specifications (TS) over operation between 20, 32, and 48 Effective Full Power Years. However, these curves will only apply to the remainder of operating cycle 13 and Operating Cycle 14. The Bases section has been modified to reflect these TS changes.

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

Sincerely,

/RA/

Alan B. Wang, Project Manager, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-293

Enclosures: 1. Amendment No. 190 to License No. DPR-35

2. Safety Evaluation

cc w/encls: See next page

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2. Safety Evaluation
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Accession Number: ML01080
*SE input dated February 9 and February 28, 2001, was provided and no major changes were made.

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ENTERGY NUCLEAR GENERATION COMPANY

DOCKET NO. 50-293

PILGRIM NUCLEAR POWER STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 190
License No. DPR-35

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

   A. The application for amendment filed by the Entergy Nuclear Generation Company (the licensee) dated November 22, 2000, as supplemented on January 30 and February 2, 2001, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations;

   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:

   B. Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 190, 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 issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

James W. Clifford, Chief, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: April 13, 2001

ATTACHMENT TO LICENSE AMENDMENT NO. 190

FACILITY OPERATING LICENSE NO. DPR-35

DOCKET NO. 50-293

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
3/4 6-14	3/4 6-14
3/4 6-15	3/4 6-15
3/4 6-16	3/4 6-16
B3/4 6-2	B3/4 6-2
B3/4 6-3	B3/4 6-3

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 190 TO FACILITY OPERATING LICENSE NO. DPR-35

ENTERGY NUCLEAR GENERATION COMPANY

PILGRIM NUCLEAR POWER STATION

DOCKET NO. 50-293

1.0 INTRODUCTION

By letter dated November 22, 2000, as supplemented on January 30 and February 2, 2001, the Entergy Nuclear Generation Company (Entergy/the licensee) submitted a request for changes to the Pilgrim Nuclear Power Station (Pilgrim) Technical Specifications (TSs). The requested changes would change the pressure-temperature (P-T) limit curves of Figures 3.6.1, 3.6.2, and 3.6.3 of Pilgrim's TSs over operation between 20, 32, and 48 effective full power years (EFPY). By letter dated February 2, 2001, the licensee requested that these curves only apply through the remainder of Operating Cycle 13 and Operating Cycle 14. The Bases section has been modified to reflect these TS changes. The January 30 and February 2, 2001, letters provided clarifying information that was within the scope of the amendment request and did not change the initial proposed no significant hazards consideration determination.

2.0 BACKGROUND

By letter dated November 22, 2000, (Ref. 1), Entergy requested TS changes to update the P-T curves. The 32 EFPYs of operation correspond to the end of the current license. The proposed fluence value for 32 EFPYs was determined by extrapolation from the value used for the current P-T curves. The current value was established from measurements and calculations related to the first surveillance capsule removed in 1980 and described in the Southwest Research Institute (SwRI) report SwRI Project No. 02-5951 and a General Electric supplement to the SwRI report (Refs. 2 and 3).

The proposed methodology for P-T limit calculations is based on the 1995 American Society of Mechanical Engineers (ASME) Appendix G methodology with two modifications. The first modification is the use of Code Case N-588, which permits both the postulation of a circumferentially oriented flaw in lieu of an axially oriented flaw for the evaluation of reactor pressure vessel (RPV) circumferential welds and the use of the revised formula for stress intensity factors due to pressure and thermal gradient for axial flaws. The second modification is the use of Code Case N-640, which permits the use of the plane strain fracture toughness (K_{IC}) curve, instead of the crack arrest fracture toughness (K_{Ia}) curve for RPV materials, in determining the P-T limits. By letter dated January 19, 2001, the licensee requested an exemption from applying the current Appendix G methodology and to use the alternative methodologies of Code Cases N-588 and N-640. The licensee further amended this TS

change request by letter dated February 2, 2001, to limit the applicability of the P-T curves only through Operating Cycle 14. With respect to the licensee's use of ASME Code Case N-588 to calculate the stress intensity factors for axial flaws, this methodology has been incorporated into the 1995 ASME Code currently endorsed by the U.S. Nuclear Regulatory Commission (NRC). Therefore, the licensee would not be required to apply for an exemption for Code Case N-588. Based on discussions with the NRC staff, the licensee withdrew the exemption request for the use of ASME Code Case N-588 by letter dated February 8, 2001.

The NRC has established requirements in Title 10 of the Code of Federal Regulations Part 50 (10 CFR 50) to protect the integrity of the reactor coolant pressure boundary in nuclear power plants. The staff evaluates the P-T limit curves based on the following NRC regulations and guidance: 10 CFR Part 50, Appendix G; Generic Letter (GL) 88-11; GL 92-01, Revision 1; GL 92-01, Revision 1, Supplement 1; Regulatory Guide (RG) 1.99, Revision 2 (Rev. 2); and Standard Review Plan (SRP) Section 5.3.2. Generic Letter 88-11 advises licensees that the staff would use RG 1.99, Rev. 2, to review P-T limit curves. RG 1.99, Rev. 2, contains methodologies for determining the increase in transition temperature and the decrease in upper-shelf energy (USE) resulting from neutron radiation. Generic Letter 92-01, Rev. 1, requested that licensees submit their RPV data for their plants to the staff for review. Generic Letter 92-01, Rev. 1, Supplement 1, requested that licensees provide and assess data from other licensees that could affect their RPV integrity evaluations. These data are used by the staff as the basis for the staff's review of P-T limit curves and as the basis for the staff's review of pressurized thermal shock (PTS) assessments (10 CFR 50.61 assessments). Appendix G to 10 CFR Part 50 requires that P-T limit curves for the RPV be at least as conservative as those obtained by applying the methodology of Appendix G to Section XI of the ASME Code.

SRP Section 5.3.2 provides an acceptable method of determining the P-T limit curves for ferritic materials in the beltline of the RPV based on the linear elastic fracture mechanics (LEFM) methodology of Appendix G to Section XI of the ASME Code. The basic parameter of this methodology is the stress intensity factor K_I, which is a function of the stress state and flaw configuration. Appendix G requires a safety factor of 2.0 on stress intensities resulting from reactor pressure during normal and transient operating conditions, and a safety factor of 1.5 for hydrostatic testing curves. The methods of Appendix G postulate the existence of a sharp surface flaw in the RPV that is perpendicular to the direction of the maximum stress. This flaw is postulated to have a depth that is equal to 1/4 thickness (1/4T) of the RPV beltline thickness and a length equal to 1.5 times the RPV beltline thickness. The critical locations in the RPV beltline region for calculating heatup and cooldown P-T curves are the 1/4T and 3/4 thickness (3/4T) locations, which correspond to the maximum depth of the postulated inside surface and outside surface defects, respectively.

The Appendix G ASME Code methodology requires that licensees determine the adjusted reference temperature (ART or adjusted RT_{NDT}). ART is defined as the sum of the initial (unirradiated) reference temperature (initial RT_{NDT}), the mean value of the adjustment in reference temperature caused by irradiation (ΔRT_{NDT}), and a margin (M) term.

ΔRT_{NDT} is a product of a chemistry factor and a fluence factor. The chemistry factor is dependent upon the amount of copper and nickel in the material and may be determined from tables in RG 1.99, Rev. 2, or from surveillance data. The fluence factor is dependent upon the neutron fluence at the maximum postulated flaw depth. The margin term is dependent upon

whether the initial RT_{NDT} is a plant-specific or a generic value and whether the chemistry factor (CF) was determined using the tables in RG 1.99, Rev. 2, or surveillance data. The margin term is used to account for uncertainties in the values of the initial RT_{NDT}, the copper and nickel contents, the fluence and the calculational procedures. RG 1.99, Rev. 2, describes the methodology to be used in calculating the margin term and the initial RT_{NDT}.

3.0 EVALUATION

3.1 Fluence Calculations

The staff has determined that the following assumptions were used for the fluence calculations submitted in References 1, 2, and 3:

- The plant specific dosimetry and/or pressure vessel calculations are dated.
- The proposed fluence values were derived from a 1981 surveillance capsule report.
- The calculations were performed in 1985 using the one-dimensional code ANISN.
- The dosimeters and their locations have not been specified.
- The results of the dosimetry to flux conversion (E > 1.0 MeV) calculations depend on the neutron energy spectrum. The spectrum calculations are not shown.
- The activation and transport cross sections have changed since these calculations were performed. Iron scattering in particular changed in a non-conservative direction.
- There is no information regarding how the neutron spectrum was derived at the location of the dosimeter, especially in the shadow of the jet pump or the jet pump risers.

Based on the staff's review of the submitted information, the staff believes that the plant-specific dosimetry and/or calculations for the original fluence value are outdated. The staff has determined that the fluence value is not credible and has related its concerns to the licensee in teleconferences on December 8 and 11, 2000, and January 3, 2001. In a letter dated February 2, 2001, (Ref. 4), Entergy proposed to limit the applicability of the proposed P-T curves to the end of the Cycle 14 refueling outage, until the licensee can perform plant-specific calculations and/or dosimetry using a new methodology soon to be approved by the NRC. Based on these calculations, the licensee will propose (if necessary) revised P-T curves for NRC staff review and approval.

The staff's pressure vessel fast neutron fluence evaluation and the justification for the acceptability of the proposed P-T curves for the interim follows. This evaluation is limited to the pressure vessel fluence. There are two conservatisms in the evaluation of the Pilgrim fluence value: (1) the proposed curves were estimated for 32 EFPYs and are to be used to about 19 EFPYs which is a conservatism factor of 1.7; and (2) Reference 3 projects a conservatism of 25 percent in the predicted peak vessel fluence. The 32 EFPYs P-T curves are bounding for operation until the end of the current license. Based on these conservatisms and considering the limited time of applicability of the proposed P-T curves (essentially one cycle), the staff

concludes that there is reasonable assurance of safety and finds the proposed curves acceptable for the period through Cycle 14, which is scheduled to end on May 15, 2003.

3.2 P-T Limit Calculations

Licensee Evaluation

The licensee submitted detailed information for ART and P-T limit curves for the limiting beltline and bottom head materials for 20, 32, and 48 EFPYs. The staff performed a confirmatory analysis for the licensee's P-T limits evaluation related only to 48 EFPYs. The licensee determined that the most limiting beltline material for cooldown curves is the lower intermediate shell axial weld with heat number 27204/12008. The licensee employed the methodology in RG 1.99, Rev. 2 and calculated an ART of 124 °F (48 EFPYs) for this limiting material based on a calculated 1/4 thickness fluence of 0.148E19 n/cm^2, an initial RT_{NDT} of -48 °F, and a margin term of 56 °F (σ_I = 0 °F and σ_Δ = 28 °F). The licensee also calculated an ART of 29 °F for the bottom head, applicable for all EFPYs since the bottom head does not receive significant amounts of neutron radiation.

Based on the beltline ART of 124 °F and the bottom head ART of 29 °F, the licensee used the methodology of Appendix G in Section XI of the ASME Code, as modified by Code Cases N-588 and N-640, to calculate the P-T limits.

Staff Evaluation

The staff compared the licensee's material information, by reference, on page B3/4.6-2 of Attachment 2 of the submittal with that in the NRC's reactor vessel integrity database (RVID) and has determined that the licensee's data for the limiting beltline material is equivalent to RVID with the exception of the surface neutron fluence (0.138E19 n/cm^2 in the RVID and 0.221E19 n/cm^2 from the licensee calculation). Using the methodology in RG 1.99, Rev. 2, the staff performed an independent calculation of ART values for the limiting beltline and the bottom head material to verify the licensee's identification of the limiting materials and their ART values for 20, 32, and 48 EFPYs.

The licensee used the ASME Code Appendix G methodology, as modified by Code Cases N-588 and N-640, to generate the heatup and cooldown P-T limits. The staff performed calculations and confirmed the validity of the proposed P-T limits for beltline materials and the bottom head. Although "HEATUP AND COOLDOWN" appears in captions and titles of the proposed P-T curves, the graph presented in the submittal is the cooldown curve. The staff performed calculations for the heatup curves and confirmed that the cooldown curves are bounding and concluded that the proposed P-T curves are valid for both heatup and cooldown. Further, the staff has reviewed and accepted the licensee's method for calculating the P-T limits for the bottom head by confirming the detailed calculation documented in the submittal as Structural Integrity Associates report SIR-00-108. Therefore, the staff has determined that the licensee's proposed P-T limit curves meet the requirements of the ASME Code as modified by Code Cases N-588 and N-640.

In addition to beltline materials, Appendix G of 10 CFR Part 50 also imposes a minimum temperature at the closure head flange based on the reference temperature for the flange material. Section IV.A.2 of Appendix G states that when the pressure exceeds 20 percent of the preservice system hydrostatic test pressure, the temperature of the closure flange regions

highly stressed by the bolt preload must exceed the reference temperature of the material in those regions by at least 160 °F for criticality, by 120 °F for subcritical heatup/cooldown, and by 90 °F for hydrostatic pressure tests and leak tests. For a critical boiling water reactor, when the pressure is less than or equal to 20 percent of the preservice system hydrostatic test pressure, the temperature of the closure flange regions highly stressed by the bolt preload must exceed the highest reference temperature of the material in the closure flange region that is highly stressed by 60 °F. Based on the flange RT_{NDT} of 10 °F, the staff determined that the straight-line segments determined by the licensee and displayed in Table 1 satisfy the requirements for hydrostatic and leak tests; subcritical heatup and cooldown, and normal operation:

Operating Condition	(°F)
Hydrostatic and Leak Tests > 20% preservice system hydrostatic test pressure	90
Subcritical Heatup and Cooldown	130
Critical Operation @ ≤ 20% preservice system hydrostatic test pressure	70
Critical Operation @ > 20% preservice system hydrostatic test pressure	160

Table 1: Minimum Temperature Requirements for the Pilgrim Reactor Pressure Vessel

The staff concludes that the proposed P-T limits for the reactor coolant system for hydrotesting, heatup, cooldown, and criticality satisfy the requirements in Appendix G to Section XI of the ASME Code, as modified by Code Cases N-588 and N-640, and Appendix G of 10 CFR Part 50 for 20, 32, and 48 EFPYs. The proposed P-T limits also satisfy GL 88-11 since the licensee used the method in RG 1.99, Rev. 2 to calculate ART. However, pending staff review of a new method to calculate neutron fluence, the proposed P-T limit curves may be incorporated into the Pilgrim TSs only through Operating Cycle 14.

3.3 Bases Changes

The Bases for the above TSs would be modified to be consistent with the proposed changes previously discussed. The staff does not object to the proposed TS Bases changes.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Massachusetts State Official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC 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 the amendment involves no significant hazards consideration, and there has been no public comment on such finding (65 FR 81915). Accordingly, the 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 the amendment.

6.0 CONCLUSION

The Commission 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, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

7.0 REFERENCES

1. Letter from M. Bellamy, Entergy Nuclear Generation Company to U.S. NRC “Request for Technical Specification Change Concerning Pressure-Temperature Limit Curves of Figures 3.6.1, 3.6.2, and 3.6.3” dated November 22, 2000.

2. SwRI Project No. 02-5951, “Pilgrim Nuclear Station Unit 1 Reactor Vessel Irradiation Surveillance Program” by E. B. Norris, Southwest Research Institute, July, 1981.

3. MDE Report No. 277-1285, “Pilgrim Nuclear Power Station Reactor Pressure Vessel Fast Neutron Flux as A Function of Fuel Cycle” Revision 1, by L.S. Burns General Electric Company, Palo Alto, CA, November 27, 1985.

4. Letter from M. Bellamy, Entergy Nuclear Generation Company to U.S. NRC, “Modification of Technical Specification Change Submittal Concerning Pressure-Temperature Limit Curves” dated February 2, 2001.

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Date: April 13, 2001