December 23, 2004

Mr. Michael R. Kansler, President
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

SUBJECT: PILGRIM NUCLEAR POWER STATION - ISSUANCE OF AMENDMENT RE: ENGINEERING EVALUATION SUBMITTED PURSUANT TO TECHNICAL SPECIFICATIONS SECTIONS 3.6.D.3 AND 3.6.D.4 (TAC NO. MC4651)

Dear Mr. Kansler:

The Commission has issued the enclosed Amendment No. 208 to Facility Operating License No. DPR-35 for the Pilgrim Nuclear Power Station (Pilgrim). This amendment is in response to Entergy Nuclear Operations, Inc.'s (Entergy's) application dated October 12, 2004.

This amendment approves an engineering evaluation performed in accordance with the Pilgrim Technical Specifications (TSs). TS 3.6.D.3 requires Entergy to perform an engineering evaluation when safety relief valve discharge pipe temperatures exceed 212 EF during normal reactor power operation for a period greater than 24 hours, and TS 3.6.D.4 further requires that power operation may not continue beyond 90 days from the initial discovery of discharge pipe temperatures in excess of 212 EF, without prior Nuclear Regulatory Commission (NRC) approval of the engineering evaluation. The NRC staff has reviewed the engineering evaluation and has determined that you have adequately justified power operations beyond the end of the TS-required 90-day period for plant shutdown, until the next cold shutdown of 72 hours or more.

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/

Robert J. Fretz, Project Manager, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Reactor Regulation

Docket No. 50-293

Enclosures: 1. Amendment No. 208 to License No. DPR-35
2. Safety Evaluation

cc w/encl: See next page

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cc w/encl: See next page
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ENTERGY NUCLEAR GENERATION COMPANY

ENTERGY NUCLEAR OPERATIONS, INC.

DOCKET NO. 50-293

PILGRIM NUCLEAR POWER STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 208
License No. DPR-35

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

   A. The application for amendment filed by Entergy Nuclear Operations, Inc. (the licensee) dated October 12, 2004, 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 authorizing changes to your licensing basis approving the engineering evaluation performed pursuant to Pilgrim Technical Specifications, Section 3.6.D.3, as set forth in the licensee's application dated October 12, 2004.

3. This license amendment is effective as of the date of issuance and shall be implemented within 30 days. Implementation of the amendment is the incorporation of the engineering evaluation into the Pilgrim licensing basis, and is effective until the next cold shutdown of 72 hours or more.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Darrell J. Roberts, Chief, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Date of Issuance: December 23, 2004

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 208 TO FACILITY OPERATING LICENSE NO. DPR-35

ENTERGY NUCLEAR GENERATION COMPANY

ENTERGY NUCLEAR OPERATIONS, INC.

PILGRIM NUCLEAR POWER STATION

DOCKET NO. 50-293

1.0 INTRODUCTION

By letter dated October 12, 2004, Entergy Nuclear Operations, Inc. (Entergy or the licensee) requested that the Nuclear Regulatory Commission (NRC or the Commission) staff review and approve an engineering evaluation performed in accordance with the Pilgrim Nuclear Power Station (Pilgrim) Technical Specifications (TSs). TS 3.6.D.3 requires the licensee to perform an engineering evaluation when safety relief valve (SRV) discharge pipe temperatures exceed 212 EF during normal reactor power operation for a period greater than 24 hours, and TS 3.6.D.4 further requires that power operation may not continue beyond 90 days from the initial discovery of discharge pipe temperatures in excess of 212 EF without prior NRC approval of the engineering evaluation. The licensee believes that the SRV is leaking small amounts of steam past its pilot stage.

The discharge pipe temperature for SRV-3C exceeded 212 EF on October 6, 2004, and by 3:09 a.m. on October 7, 2004, the discharge pipe temperature had exceeded 212 EF for greater than 24 hours.

2.0 REGULATORY EVALUATION

Pilgrim TS Section 3.6.D.3 states:

> If the temperature of any safety relief valve discharge pipe exceeds 212 EF during normal reactor power operation for a period greater than 24 hours, an engineering evaluation shall be performed justifying continued operation for the corresponding temperature increases.

Furthermore, Pilgrim TS Section 3.6.D.4 states:

> Power Operation shall not continue beyond 90 days from the initial discovery of discharge pipe temperatures in excess of 212 EF for more than 24 hours without prior NRC approval of the engineering evaluation delineated in 3.6.D.3.

The licensee requested that the NRC approve its engineering evaluation for continued plant operation with the leakage past SRV-3C. Entergy requested this approval in accordance with Title 10 of the *Code of Federal Regulations*, Section 50.90, and Pilgrim TS 3.6.D.4. In its letter dated October 12, 2004, the licensee also committed to enforce a more restrictive surveillance limit on continued operation with higher discharge pipe temperatures for SRV-3C, as follows:

1. If the discharge pipe temperature for the SRV exceeds 235 EF for 24 hours an orderly shutdown of the reactor shall commence and the reactor pressure shall be less than 104 psig within 24 hours. In addition, if the discharge pipe temperature for the SRV exceeds 250 EF an orderly shutdown of the reactor shall commence and the reactor pressure shall be less than 104 psig within 24 hours.

2. Technical Specifications surveillance 4.6.D.3 requires that SRV discharge pipe temperature be logged daily. This surveillance shall be performed at an increased frequency of once per hour while the discharge pipe temperatures are greater than 212 EF for SRV-3C.

The staff also notes that TS 3.7 requires measurement of torus and drywell temperatures and limits plant operations for temperatures above specific limits, and that TS 3.6.D.4 requires the SRVs to be removed for testing and recalibration at the next cold shutdown greater than 72 hours in duration. The testing and recalibration must meet the provisions of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (the Code), which requires that the leak be repaired prior to returning the valve to service.

3.0 TECHNICAL EVALUATION

As previously discussed, TS 3.6.D.3 requires that an engineering evaluation be performed to support continued operation if the temperature of any SRV discharge pipe exceeds 212 EF for a period greater than 24 hours during normal reactor power operation. The TS Bases states that minimal leakage exists when the discharge pipe temperature is 215 EF; and, therefore, a conservative temperature limit of 212 EF was chosen.

Following an evaluation of the SRV-3C temperature profiles, the licensee determined that the leakage is likely due to pilot stage (pilot seat area) leakage because of the similarity of the increase in discharge pipe temperatures to previous leaking pilots. On the basis of their engineering evaluation, the licensee further concluded that the temperature is expected to remain within the range of 215 to 230 EF from now until the next refueling outage.

The SRVs are part of the reactor coolant pressure boundary and operate by power actuation (i.e., automatic depressurization system) or self-actuation by process high pressure. The SRVs limit peak vessel pressure during overpressure transients to satisfy ASME Code requirements. The automatic depressurization system provides a means to rapidly depressurize the primary system down to a pressure at which low-pressure cooling systems can provide makeup. In the event of a small or medium break loss of coolant accident, this function would be required if the high pressure coolant injection system is unable to maintain vessel water level. The consequences of leakage across either the pilot or main stage boundary for SRV-3C were addressed in the submittal, and are discussed below.

As previously noted, the licensee believes that the most likely path of the leakage is across the pilot stage of the SRV. Even if the leakage is across the main stage, the consequences would be expected to be minimal. The licensee states that the main stage leakage does not affect valve lift set point, or the valve response time. The licensee concluded that leakage across the main disc will not affect the ability of the SRV to operate in either the pressure actuated, or power actuated modes. In addition, leakage across the main stage should not cause the SRV to inadvertently open and cause a rapid depressurization or fail to reclose after operating.

Pilot stage leakage, on the other hand, affects the valve lift set point, as well as the valve response time. Pilot stage leakage can affect the performance of the two stage Target Rock SRV in the pressure-actuated mode (i.e., safety mode). The effects of leakage on valve performance, such as set point drift and response time changes, were extensively studied (Reference 2). On the basis of these test results, pilot stage leakage up to 1000 lbs/hr does not significantly affect the SRV set point. An SRV discharge pipe temperature of approximately 255 EF can be correlated to a steam leakage flow rate of approximately 225 lbs/hr, while steam leakage of 1000 lbs/hr can be correlated to a discharge pipe temperature of approximately 275 EF. Therefore, the staff agrees with the licensee that it is acceptable to continue operation with a discharge pipe temperature of less than or equal to 255 EF since test data has demonstrated that the possible relief valve set point drift at this temperature is equivalent to +1% (Reference 2).

Valve response time is defined as the interval from pilot actuation to main disc lift. The normal response time for a two stage Target Rock SRV is approximately 0.4 seconds. Response time varies with leakage rate. A slower response time results in a higher peak reactor vessel pressure during the safety mode, and a faster response time results in a lower peak reactor pressure. A slower response time results when discharge pipe temperature increases. A combined effect of a longer response time and setpoint drift due to leakage and the resulting impact on the peak vessel pressure was evaluated by the licensee. The evaluation and the results were presented in the submittal (Reference 1), which is summarized as follows: The peak vessel pressure is estimated at 1335 psig which is significantly below the ASME Code allowable of 1375 psig. The corresponding peak dome pressure of 1317 psig is less than the TS Safety Limit of 1325 psig. The following conservative bounding assumptions were made for the analysis:

- A 10% increase in the nominal setpoint of each of the four SRVs results in a peak pressure increase of 30 psig (Reference 2).

- The effect of a response time delay of 0.9 seconds is 5 psig (Reference 2).

These analyses demonstrate that maximum system pressure remains below the upset limit of 1375 psig. In addition, the licensee stated that there is only a minimal effect on critical power thermal margin.

The installed relief and spring safety valves protect the reactor coolant pressure boundary from exceeding the ASME Code Level C limit of 1500 psig during a full power anticipated transient without scram (ATWS). The limiting ATWS is a pressure regulator failure that causes the turbine control and bypass valves to fail open, leading to vessel depressurization and main steam isolation valve closure on low steamline pressure. This event was re-analyzed for the recent Thermal Power Optimization uprate from 1998 MWth to 2028 MWth as documented in

Reference 3. The reanalysis for the leaking valve credited the currently installed larger throat sizes of 5.125 inches and assumed one SRV experienced a 21 psig setpoint drift (equivalent to 1.8%) that caused the valve to open at 1136 psig (1115 psig + 21 psi drift). The remaining three SRVs were assumed to open at 1126 psig (1115 psig + 1% (11 psi)). This reanalysis also assumes a 0.55-second response time delay for all four SRVs, which bounds the response time predicted for the selected discharge pipe temperature limit. The calculated peak vessel pressure for the limiting ATWS is 1495 psig which meets the acceptance criteria of 1500 psig. The selected discharge pipe temperature limit is chosen to limit setpoint drift to less than 1%; and, therefore, the ATWS analysis described above bounds the conditions permitted by this operability evaluation and ensures the ASME Code Level C limit is not exceeded during the worst case ATWS. The effect of the leakage on torus water temperature is expected to be small because the evaluated leakage is appreciably small relative to the mass of the torus water volume. Similarly, any effects on drywell air temperature or containment pressure are also expected to be insignificant. No other systems are expected to be impacted by this amount of leakage.

3.1 NRC Staff's Conclusion

The licensee has satisfied TS 3.6.D.3 and TS 3.6.D.4 requirements that an engineering evaluation be performed for the leaking SRV conditions. The NRC staff has reviewed the evaluation and has found the proposed temperature limits identified in Entergy's engineering evaluation for the SRV-3C tailpipe to be conservative such that the SRV will remain capable of performing its intended function. If the tailpipe temperature exceeds these limits, the licensee will conduct an orderly shutdown as described in Reference 1. Further, the NRC staff has determined that the licensee's proposed increased temperature monitoring frequency will be adequate to determine the quantity of the SRV leakage in a timely manner. Therefore, the licensee's engineering evaluation adequately justifies plant operation beyond January 4, 2005, (the end of the TS-required 90-day period) until the next cold shutdown of 72 hours or more.

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 approves the licensee's engineering evaluation with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. 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 (69 FR 61695). 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 dated October 12, 2004 to NRC from Entergy, “Request for NRC Approval of Engineering Evaluation for Elevated Discharge Pipe Temperature of Safety Relief Valve 203-3C.”

2. General Electric Report NEDE-30476, “Set point Drift Investigation of Target Rock Two-Stage Safety/Relief Valve (Final Report)”, dated February 1984.

3. GENE-0000-0000-6653, “Project Task Report T0902, ‘Anticipated Transients Without Scram”, Revision 0, January 2002.

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