UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

August 7, 2012

Site Vice President
Entergy Nuclear Operations, Inc.
Pilgrim Nuclear Power Station
600 Rocky Hill Road
Plymouth, MA 02360-5508

SUBJECT: PILGRIM NUCLEAR POWER STATION - ISSUANCE OF AMENDMENT REGARDING CONDENSATE STORAGE TANK LOW LEVEL TRIP SETPOINT REVISION (TAC NO. ME7545)

Dear Sir or Madam:

The Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 237 to Facility Operating License No. DPR-35 for the Pilgrim Nuclear Power Station (Pilgrim), in response to your application dated October 28, 2011 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML11312A051), as supplemented by letter dated May 16, 2012 (ADAMS Accession No. ML12139A304).

This amendment revises the Pilgrim Technical Specifications (TSs) to increase the condensate storage tank low water level setpoint for the interlock to the high pressure coolant injection pump suction valves. Additionally, the amendment corrects typographical errors in TS numbering and referencing made in prior license amendment nos. 223 and 228.

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

Sincerely,

RGuzman

Richard V. Guzman, Senior Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-293

Enclosures:
1. Amendment No. 237 to DPR-35
2. Safety Evaluation

cc w/encls: Distribution via Listserv

ENTERGY NUCLEAR GENERATION COMPANY

ENTERGY NUCLEAR OPERATIONS, INC.

DOCKET NO. 50-293

PILGRIM NUCLEAR POWER STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 237
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 28, 2011, as supplemented on May 16, 2012, 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. 237, are hereby incorporated into this 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 and shall be implemented within 60 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

George A. Wilson, Chief
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the License and
Technical Specifications

Date of Issuance: August 7, 2012

ATTACHMENT TO LICENSE AMENDMENT NO. 237

TO FACILITY OPERATING LICENSE NO. DPR-35

DOCKET NO. 50-293

Replace the following page of the Facility Operating License with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

Remove Page	Insert Page
3	3

Replace the following pages of 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 Pages	Insert Pages
3/4.2-16	3/4.2-16
3/4.3-2	3/4.3-2
3/4.13-1	3/4.13-1
3/4.13-2	3/4.13-2

provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified below:

A. Maximum Power Level

ENO is authorized to operate the facility at steady state power levels not to exceed 2028 megawatts thermal.

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 237, are hereby incorporated in the renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

C. Records

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

D. Equalizer Valve Restriction – DELETED

E. Recirculation Loop Inoperable – DELETED

F. Fire Protection

ENO shall implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report for the facility and as approved in the SER dated December 21, 1978 as supplemented subject to the following provision:

ENO may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

G. Physical Protection

The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans, which contain Safeguards Information protected under 10 CFR 73.21, is entitled: "Pilgrim Nuclear Power Station Physical Security, Training and Qualification, and Safeguards Contingency Plan, Revision 0" submitted by letter dated October 13, 2004, as supplemented by letter dated May 15, 2006.

The licensee shall fully implement and maintain in effect all provisions of the Commission-approved cyber security plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The licensee's CSP was approved by License Amendment No. 236.

TABLE 3.2.B (CONT)

INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

Minimum # of Operable Instrument Channels Per Trip System (1)	Trip Function	Trip Level Setting	Remarks
2	Condensate Storage Tank Low Level	≥ 46" above tank zero (a)	Provides interlock to HPCI pump suction valves.
2	Suppression Chamber High Level	≤ 1'11" below torus zero	
1	RCIC Turbine Steam Line High Flow	≤ 300% of rated steam flow	(2)
2	RCIC Turbine Compartment Wall	≤ 168°F	(2)
2	RCIC Exhaust Duct Torus Cavity	≤ 148°F	(2)
2	RCIC Valve Station Area Wall	≤ 198°F	(2)
4	RCIC Steam Line Low Pressure	77 > P > 63 psig	(2)(5)(6)
1	HPCI Turbine Steam Line High Flow	≤ 296% of rated flow	(3)
2	HPCI Turbine Compartment Exhaust Duct	≤ 168°F	(3)
2	HPCI Exhaust Duct Torus Cavity	≤ 198°F	(3)
2	HPCI/RHR Valve Station Area Exhaust Duct	≤ 168°F	(3)

(a) If the as-found channel setpoint is outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service. The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Nominal Trip Setpoint (NTSP) at the completion of the surveillance; otherwise the channel shall be declared inoperable. Setpoints more conservative than NTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the Surveillance procedure to confirm channel performance. The NTSP and the methodologies used to determine the as-found and as-left tolerances are specified in the Pilgrim Setpoint calculation for Condensate Tank Low Level Instrumentation, IN1-245.

LIMITING CONDITIONS FOR OPERATION

3.3 REACTIVITY CONTROL (continued)

B. Control Rod Operability

LCO 3.3.B.1

Each control rod shall be OPERABLE.

APPLICABILITY:

RUN and STARTUP MODES; REFUEL MODE when the reactor vessel head is fully tensioned. (See also 3/4.14.E)

ACTIONS

----------------NOTE--------------------
Separate condition entry is allowed for each control rod.

A. One withdrawn control rod stuck.

------------------NOTE------------------
Rod Worth Minimizer (RWM) may be bypassed as allowed by LCO 3.3.F.
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1. Verify stuck control rod separation criteria are met immediately.

AND

2. Disarm the associated control rod drive (CRD) within 2 hours.

AND

3. Perform SR 4.3.B.1.1 and SR 4.3.B.1.2 for each withdrawn OPERABLE control rod within 24 hours from discovery of condition A concurrent with thermal power greater than the Low Power Setpoint (LPSP) of the RWM.

AND

4. Verify LCO 3.3.A.1 is met within 72 hours.

AND

SURVEILLANCE REQUIREMENTS

4.3 REACTIVITY CONTROL (continued)

B. Control Rod Operability

SR 4.3.B.1.1

----------------------NOTE----------------------
Not required to be performed until 7 days after the control rod is withdrawn and thermal power is greater than the LPSP of the RWM.

Insert each fully withdrawn OPERABLE control rod at least one notch once per 7 days.

SR 4.3.B.1.2

----------------------NOTE----------------------
Not required to be performed until 31 days after the control rod is withdrawn and thermal power is greater than the LPSP of the RWM.

Insert each partially withdrawn OPERABLE control rod at least one notch once per 31 days.

SR 4.3.B.1.3

Verify each withdrawn control rod does not go to the withdrawn overtravel position.

a. Each time the control rod is withdrawn to "full out" position.

AND

b. Prior to declaring control rod OPERABLE after work on control rod or CRD system that could affect coupling.

SR 4.3.B.1.4

Verify each control rod scram time from fully withdrawn to notch position 04 is $\leq$ 7 seconds in accordance with SR 4.3.C.1, SR 4.3.C.2, SR 4.3.C.3 or SR 4.3.C.4

SR 4.3.B.1.5

Determine the position of each control rod once per 24 hours.

LIMITING CONDITIONS FOR OPERATION

3.13 INSERVICE CODE TESTING

Applicability:

Applies to ASME Code Class 1, 2 and 3 or pumps and valves.

Objective:

To assure the operational readiness of ASME Code Class 1, 2, and 3 pumps and valves.

Specification:

A. Inservice Code Testing of Pumps and Valves

1. Based on the Facility Commercial Operation Date, Inservice Code Testing of ASME Code Class 1, 2 and 3 pumps and valves shall be performed in accordance with the Inservice Code Testing Program.

SURVEILLANCE REQUIREMENTS

4.13 INSERVICE CODE TESTING

Applicability:

Applies to the periodic testing requirements of ASME Code Class 1, 2 and 3 pumps and valves.

Objective:

To assess the operational readiness of ASME Code Class 1, 2, and 3 pumps and valves by performance of inservice tests.

Specification:

A. Inservice Code Testing of Pump and Valves

1. The ASME OM Code terminology for Inservice Test activities is as follows.

Code Terminology	Frequencies
Weekly	7 Days
Monthly	31 Days
Quarterly or 3 Mths	92 Days
Semiannually/ 6 Mths	184 Days
9 Months	276 Days
Yearly/Annually	366 Days
Biannual/2 Yrs	732 Days

2. The provisions in Definitions (1.0) for REFUELING INTERVAL, SURVEILLANCE FREQUENCY, and SURVEILLANCE INTERVAL are applicable to Code testing and to the above frequencies for performing Code testing activities.

3. Performance of Code testing shall be in addition to other specified Surveillance Requirements.

4. Nothing in the Inservice Code Testing Program shall supersede the requirements of Technical Specifications.

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 237

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 28, 2011 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML11312A051), as supplemented by letter dated May 16, 2012 (ADAMS Accession No. ML12139A304), Entergy Nuclear Operations, Inc. (the licensee) proposed a license amendment to revise the technical specifications (TSs) of Pilgrim Nuclear Power Station (Pilgrim). The supplement dated May 16 , 2012, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination noticed in the *Federal Register* on January 10, 2012 (77 FR 1517).

The proposed amendment would modify the Pilgrim TSs to increase the condensate storage tank (CST) low water level setpoint for the interlock to the High Pressure Coolant Injection (HPCI) pump suction valves to account for a more conservative calculation of the vortex formation at the suction of the HPCI pump. The proposed amendment would also modify the TSs to correct numbering and referencing typographical errors made in the prior license amendments nos. 223 and 228.

2.0 REGULATORY EVALUATION

The regulatory requirements and guidance which the NRC staff considered in assessing the proposed TS change are as follows:

Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.36, "Technical specifications," of 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," requires that "each applicant for a license authorizing operation of a production or utilization facility shall include in his application proposed technical specifications in accordance with the requirements of this section." Specifically, 10 CFR 50.36(c)(1)(ii)(A) states, "Limiting safety system settings for nuclear reactors are settings for automatic protective devices related to those variables having significant safety functions. Where a limiting safety system setting is specified for a variable on which a safety limit has been placed, the setting must be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded. If, during operation, it is determined that the automatic safety system does not function as required, the licensee shall take appropriate action, which may include shutting down the reactor."

Regulatory Guide (RG) 1.105, "Setpoints for Safety-Related Instrumentation," Revision 3, describes a method that the NRC staff finds acceptable for use in complying with the NRC's regulations for ensuring that setpoints for safety-related instrumentation are initially within, and will remain within the TS limits.

In Regulatory Issue Summary (RIS) 2006-17, "NRC Staff Position on the Requirements of 10 CFR 50.36, "Technical Specifications," regarding Limiting Safety System Settings during Periodic Testing and Calibration of Instrument Channels," dated August 24, 2006 (ADAMS Accession No. ML051810077), the NRC addresses requirements on limiting safety system settings that are assessed during the periodic testing and calibration of instrumentation. RIS 2006-17 discusses issues that could occur during the testing of limiting safety system settings and that, therefore, may have an adverse effect on equipment operability.

The Standard Technical Specifications (STS) were developed based on the criteria in 10 CFR 50.36(c)(2)(ii). Existing limiting conditions for operations (LCOs) and related surveillance requirements (SRs) included as TS requirements which satisfy any of the criteria specified in 10 CFR 50.36(c)(2)(ii) must be retained in the TS. Pilgrim uses boiling water type nuclear steam supply systems furnished by General Electric. The NRC staff reviewed the licensee's proposed TS changes in accordance with NUREG-1433, "Standard Technical Specifications General Electric Plants, [boiling-water reactor] BWR/4."

3.0 <u>TECHNICAL EVALUATION</u>

3.1 <u>Licensee's Proposed Changes</u>

In its application dated October 28, 2011, the licensee stated that the proposed modifications will result in the following TS changes:

- Condensate Storage Tank Low Level Trip Level Setting is changed from "≥18" above tank zero" to "≥ 46" above tank zero" on Table 3.2B, "INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS," page 3/4.2-16.

- The "Inservice Code Testing" surveillance requirements are sequentially numbered as 1, 2, 3, and 4 under TS Section 4.13, TS pages 3/4.13-1, and TS page 3/4.13-2 is marked as "Intentionally Left Blank." The purpose of this change is to correct the typographical errors made in prior license amendment no. 223.

- The TS LCO 3.3.B.1 Applicability is being corrected to state in parenthesis from "(See also 3.10.D)" to "(See also 3/4.14.E)" on TS page 3/4.3-2. The purpose of this change is to correct the typographical errors made in prior license amendment no. 228.

The proposed TS changes would also result in a change to the Pilgrim TS Bases. Specifically, the reference to LCO 3.10.D in the BASES section 3.10, "Core Alterations" is corrected to LCO "3/4.14.E" on pages B3/4.10-1. On page B3/4.10-3, references to "Specification to 3.10.D" is corrected to Specification "3/4.14.E." The licensee included in its submittal, proposed marked-up pages to the affected TS Bases section 3.10 pages, for NRC information only. All other licensing bases changes associated with the revision to the condensate storage tank low level trip level setting from "≥18" above tank zero" to "≥ 46 above tank zero" will be formally addressed by the licensee in accordance with TS 5.5.6, "Technical Specification (TS) Bases Control Program."

3.2 Background

In 2006, the NRC inspection team identified a low safety significance non-cited violation that the licensee used a non-conservative calculation method to determine the critical CST water level which would preclude vortex formation at the suction of the HPCI pump (Reference 1). The formation of vortex at the intake of the HPCI suction line could result in air entrainment, which in turn, could cause pulsating pump flow and/or reduction in HPCI pump performance. The licensee's corrective actions for the inspection finding included testing on a scale model of a single CST at Alden Research Laboratory to establish the critical water level to preclude air ingestion from vortices. The licensee calculated a trip setpoint of 58 inches and TS allowable value (AV) of ≥ 46 inches for the HPCI suction transfer on low CST water level based on the model test results, flow rate uncertainties, initiating signal time delay, valve operating times, and instrument setpoint methodology and implemented an engineering change that increased the setpoint from 36 inches to 58 inches. The purpose of this amendment request is to increase, in accordance with these test results, the HPCI low CST level allowable value to ≥ 46 inches from the current value of ≥ 18 inches in TS Table 3.2.B.

At Pilgrim, HPCI and the reactor core isolation cooling (RCIC) are normally aligned to both CSTs. HPCI and RCIC share a common suction located on the bottom of each CST which contains reactor quality water and are the normal water source for reactor makeup. Two CST tanks are installed to allow maintenance on one tank when HPCI and RCIC are required to be operable. The proposed TS allowable value is selected based on operation of both HPCI and RCIC with a single CST in-service.

When the CST water level reaches a low limit, HPCI transfers its suction from the CST to the torus. There are three valves in the HPCI suction. One valve lines up pump suction from the CST, the other two from the torus. If water level in the CST falls below the minimum level, the torus suction valves automatically open after a time delay. When both torus suction valves are fully open, the CST suction valve automatically closes.

Pilgrim uses two pressure switches (PS-2390A and B) to detect the CST low water level condition and initiate a HPCI suction transfer. The pressure switches sense pressure in the HPCI suction line from the CST. Either switch can cause the torus suction valves to open. A short time delay is included in the torus suction valves opening circuit to prevent false/transient signals from initiating suction valve transfer.

Entergy contractor, AREVA, conducted a physical hydraulic model study of a single CST at Alden Research Laboratory to establish the critical tank water level to avoid vortexing/ air-entrainment. The testing evaluated the potential for vortexing/air-entrainment at specified flow rates as the tank drains and the water level is lowered. Entergy then adjusted (increased) the critical tank water level from the model testing to account for flow measurement uncertainty of the HPCI and RCIC system flow controls and the CST level decrease that will occur at maximum flow rate while the suction valves change position after the pressure switches initiate HPCI suction transfer to the torus. The critical tank water level of 43 inches was established as the analytical limit (AL) used in the instrument uncertainty calculation.

3.3 NRC Staff's Evaluation

3.3.1 CST Low Water Level Trip Setting

The licensee proposed to revise the Trip Level Setting for the CST Low Water Level from "≥18" above tank zero" to "≥ 46" above tank zero" in TS Table 3.2.B, page 3/4.2-16.

The licensee submitted a copy of its setpoint calculation methodology and a summary of its setpoint calculation for the proposed setpoint change in an attachment to the application (Reference 2). The NRC staff noted that the licensee established the AL of 43 inches and calculated the channel uncertainty as ±15 inches for the CST low level setpoint. Based on the AL and channel uncertainty values, the licensee established the Nominal Trip Setpoint (NTSP) of 58 inches.

Noting that the CST low level setpoint is a limiting safety system settings (LSSS) setpoint, the NRC staff requested the licensee, by letter dated March 26, 2012 (ADAMS Accession No. ML120860147), to provide the basis for the as-left tolerance (ALT) determination and to indicate whether it will apply the TSTF-493, Rev. 4, "Clarify Application of Setpoint Methodology for LSSS Functions," for this setpoint change. The licensee submitted its request for additional information (RAI) response on May 16, 2012 (Reference 3). In its RAI response, the licensee states that it typically uses the accuracy of the device based on the surveillance interval for the ALT (the licensee also uses the term "No-Adjust Limit") and follows the Pilgrim Procedure No. 8.M.2-2.5.6 (Reference 4) to check/calibrate this level instrument. The licensee states that when the technician checks/calibrates the device, the expectation is that the desired value will fall within the specified no-adjust limits. If the technician finds the value outside the no adjust limits, as specified in the procedure, a correction report is written so that the engineering staff can review the results to determine if the instrument is malfunctioning or if the calculation should be revised using a different no-adjust limit.

The NRC staff noted that the licensee calculated the as-found tolerance (AFT) which consists of the square-root-of-the-sum-of-squares (SRSS) of the values of the drift and the ALT. The licensee then subtracted AFT from NSTP to obtain the AV. The NRC staff verified that the calculated values of ALT and AFT were determined to be ±5.5 inches and ±12 inches, respectively. As a result, the licensee established the AV for low-level CST as 46 inches. The NRC staff noted that the licensee includes only reference accuracy in its determination of ALT, which is more conservative than the guidance specified in the RIS 2006-017. The licensee includes measurement and test equipment uncertainty as part of its determination of drift, which is included in the determination of AFT. The NRC staff found that the method used for Pilgrim to calculate these terms follows the principles of the NRC staff guidance in the RIS 2006-017.

Regarding the NRC staff's concerns on the operability of the instrument loop, as described by TSTF-493, the licensee stated in its RAI response that Pilgrim is not adopting TSTF-493, Rev. 4, but will follow the guidance as prescribed through RG 1.105, Revision 3, and RIS 2006-017 for applicable instruments within the scope of the proposed license amendment.

In order to define the criteria for evaluating operability of the instrument loop to the Trip Function "Condensate Storage Tank Low Level" within Pilgrim TS Table 3.2.B, the licensee added Note (a) for this Function. The NRC staff finds that the licensee's proposed Note (a) is consistent with the two footnotes specified in TSTF-493, Rev. 4, and follows the NRC guidance in RIS 2006-017. The proposed Note (a) enhances safety by ensuring that unexpected as-found conditions are evaluated before returning the channel to service and by ensuring that as-left settings provide sufficient margin for uncertainties.

The NRC staff noted that Pilgrim TS Table 4.2.B specifies that the Calibration Frequency for CST Low Level is once per 3 months. The NRC staff also noted that the 92-day surveillance frequency is consistent with the current frequency specified in the surveillance requirements (SR) bases in NUREG-1433, Revision 3, "Standard Technical Specifications - General Electric Plant, BWR 4," issued June 2004, and has been shown to be acceptable through operating experience.

Based on the above discussion, the NRC staff finds that the proposed TS change to the CST low level trip setpoint follows the principles of RIS 2006-017 and RG 1.105, Revision 3, and meets the requirements of 10 CFR 50.36(c)(1)(ii)(A). Therefore, the NRC staff concludes that the proposed TS change is acceptable.

3.3.2 Typographical Error in License Amendment No. 223

A typographical error in License Amendment No. 223 was introduced on the licensee's marked-up TS pages 3/4.13-1 and 3/4.13-2 in the associated license amendment request dated December 14, 2004 (ADAMS Accession No. ML043560279). This error was carried forth to the re-typed TS pages, and ultimately issued by the NRC as License Amendment 223, "Administrative Changes and Relocation of Certain Technical Specifications", dated August 6, 2006 (ADAMS Accession No. ML061720068). Specifically, the error is under TS section 4.13.A, "Inservice Code Testing of Pump and Valves," where the surveillances were numbered as 1, 3, 4, and 5 (omitting 2). The deletion of chronological number "2" in the December 14, 2004, license amendment request, resulted in a numbering sequence error. This error is being corrected in TS pages 3/4.13-1 by sequentially numbering the surveillances as 1, 2, 3, and 4, and marking TS 3/4.13-2 as "Page Intentionally Left Blank."

The NRC staff has confirmed that the related administrative changes are appropriate, editorial in nature, and do not affect the technical requirements of TS 4.13, "Inservice Code Testing." Therefore, the NRC staff finds this change acceptable.

3.3.3 Typographical Error in License Amendment No. 228

A typographical error in License Amendment No. 228 was introduced on the licensee's marked-up TS pages and bases pages for the associated license amendment request dated October 18, 2005 (ADAMS Accession No. ML053040450). Prior to the proposed license amendment change, TS Section 3.10.D addressed TSs related to refueling conditions, which allowed multiple control rod removal. By License Amendment 228, the requirements for refueling conditions were relocated to TS Section 3/4.14.E, "Multiple Control Rod Removal." However, all necessary mark-ups were not submitted by the licensee in its October 18, 2005, license amendment request. This error was carried forth in the issuance of License Amendment 228, "Single Control Rod Withdrawal Allowances, TS 3/4.14, "Special Operations," dated April 25, 2007 (ADAMS Accession No. ML070740419). Specifically, the TS LCO 3.3.B.1 APPLICABILITY statement and TS BASES pages B3/4.10-1 and B3/4.10-3 makes an incorrect reference to TS Section 3.10.D. This error is being corrected in TS page 3/4.3-2 by making reference to the appropriate TS section, 3/4.14.E. The reference to TS Section 3.10.D in the BASES section 3.10, "Core Alterations," is corrected to LCO "3/4.14.E" on page B3/4.10-1, and to Specification "3/4.14.E" on page B3/4.10-1.

The NRC staff has confirmed that the related administrative changes are appropriate, editorial in nature, and do not affect the technical requirements. Therefore, the NRC staff finds this change acceptable.

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 SRs. 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 (January 10, 2012 (77 FR 1517)). 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) there is reasonable assurance that 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. Pilgrim Nuclear Power Station – NRC Inspection Report 05000293/2006006, dated June 29, 2006 (ADAMS Accession No. ML061800215).

2. Calculation IN1-245, "Setpoint Calculation for PS-2390A & B, Condensate Tank Low Level Transfer," Revision 2, dated October 7, 2009 (ADAMS Accession No. ML11312A052).

3. Letter from R. G. Smith (Entergy) to USNRC, "Entergy Response to NRC Request for Additional Information dated March 26, 2012, in support of Proposed License Amendment: Revision to Condensate Storage (CST) Low Level Trip Setpoint (TAC No. ME7545)," May 16, 2012 (ADAMS Accession No. ML12139A304)

4. Attachment 2 of May 16, 2012, Letter from R. G. Smith (Entergy) to USNRC, Pilgrim Procedure No. 8.M.2-2.5.6, "HPCI Condensate Storage Tank Level", Revision 35, November 2009 (ADAMS Accession No. ML12139A304).

Principal Contributors: P. Chung

Date: August 7, 2012

August 7, 2012

Site Vice President
Entergy Nuclear Operations, Inc.
Pilgrim Nuclear Power Station
600 Rocky Hill Road
Plymouth, MA 02360-5508

SUBJECT: PILGRIM NUCLEAR POWER STATION - ISSUANCE OF AMENDMENT REGARDING CONDENSATE STORAGE TANK LOW LEVEL TRIP SETPOINT REVISION (TAC NO. ME7545)

Dear Sir or Madam:

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

Sincerely,

/ra/

Richard V. Guzman, Senior Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-293

Enclosures:
1. Amendment No. 237 to DPR-35
2. Safety Evaluation

cc w/encls: Distribution via Listserv

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