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

May 6, 2015

Mr. John A. Dent, Jr.
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 THE MINIMUM CRITICAL POWER RATIO LICENSE AMENDMENT REQUEST (TAC NO. MF5431)

Dear Mr. Dent:

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 243 to Renewed Facility Operating License No. DPR-35 for the Pilgrim Nuclear Power Station. This amendment consists of changes to the Technical Specification (TS) in response to your application transmitted by letter dated December 10, 2014, as supplemented by letters dated February 13, and March 11, 2015.

This amendment revises the minimum critical power ratio from $\geq$ 1.08 to $\geq$ 1.10 for two recirculation loop operation and from $\geq$ 1.11 to $\geq$ 1.12 for single recirculation loop operation in TS 2.1, "Safety Limits."

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

Sincerely,

Nadiyah S. Morgan, 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. 243 to License No. DPR-35
2. Safety Evaluation

cc w/encls: Distribution via Listserv

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

ENTERGY NUCLEAR GENERATION COMPANY

AND ENTERGY NUCLEAR OPERATIONS, INC.

PILGRIM NUCLEAR POWER STATION

DOCKET NO. 50-293

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 243
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 December 10, 2014, as supplemented by letters dated February 13, 2015, and March 11, 2015, 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 Renewed 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. 243, are hereby incorporated in the renewed operating 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 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael I. Dudek, Acting 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: May 6, 2015

ATTACHMENT TO LICENSE AMENDMENT NO. 243

RENEWED FACILITY OPERATING LICENSE NO. DPR-35

DOCKET NO. 50-293

Replace the following page of the Renewed 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	Insert
3	3

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
2-1	2-1

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. 243, 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 (50 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, as supplemented by changes approved by :License Amendment Nos. 238 and 241.

2.0 SAFETY LIMITS

2.1 Safety Limits

2.1.1 With the reactor steam dome pressure < 685 psig or core flow < 10% of rated core flow:

THERMAL POWER shall be ≤ 25% of RATED THERMAL POWER.

2.1.2 With the reactor steam dome pressure ≥ 685 psig and core flow ≥ 10% of rated core flow:

MINIMUM CRITICAL POWER RATIO shall be ≥ 1.10 for two recirculation loop operation or ≥ 1.12 for single recirculation loop operation.

2.1.3 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 inches above the top of the normal active fuel zone.

2.1.4 Reactor steam dome pressure shall be ≤ 1340 psig at any time when irradiated fuel is present in the reactor vessel.

2.2 Safety Limit Violation

With any Safety Limit not met within two hours the following actions shall be met:

2.2.1 Restore compliance with all Safety Limits, and

2.2.2 Insert all insertable control rods.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 243

TO RENEWED FACILITY OPERATING LICENSE NO. DPR-35

ENTERGY NUCLEAR GENERATION COMPANY

AND ENTERGY NUCLEAR OPERATIONS, INC.

PILGRIM NUCLEAR POWER STATION

DOCKET NO. 50-293

1.0 INTRODUCTION

By application dated December 10, 2014 (Agencywide Document Access and Management System (ADAMS) Accession Nos. ML14349A495 and ML14349A496), as supplemented by letters dated February 13, 2015 (ADAMS Accession No. ML15050A245), and March 11, 2015 (ADAMS Accession No. ML15075A033), Entergy Nuclear Operations, Inc. (ENO, the licensee), submitted a request to the U.S. Nuclear Regulatory Commission (NRC) for changes to the Pilgrim Nuclear Power Station (Pilgrim) Technical Specification (TS). The supplements dated February 13, and March 11, 2015, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* (FR) on March 12, 2015 (80 FR 13030).

The proposed changes would revise the TS associated with the safety limit minimum critical power ratio (SLMCPR) for single and two recirculation loop operations.

2.0 REGULATORY EVALUATION

2.1 Description of System

The licensee stated in its letter dated December 10, 2014 (Reference 1), that:

> The fuel cladding integrity safety limit is set such that no mechanistic fuel damage is calculated to occur if the limit is not violated. Since the parameters which result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions resulting in a departure from nucleate boiling have been used to mark the beginning of the region where fuel damage could occur. Although it is recognized that a departure from nucleate boiling

would not necessarily result in damage to boiling-water reactor [BWR] fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. The uncertainties in monitoring the core operating [condition] and in the procedures used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore, the fuel cladding integrity safety limit is defined as the minimum critical power ratio (MCPR) in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition considering the power distribution within the core and all uncertainties.

2.2 Proposed TS Change

In TS 2.1, "Safety Limits," the MCPR would change from $\geq$ 1.08 to $\geq$ 1.10 for two recirculation loop operation and from $\geq$ 1.11 to $\geq$ 1.12 for single recirculation loop operation.

2.3 Regulatory Requirements and Guidance

The construction permit for Pilgrim was issued by the Atomic Energy Commission (AEC) on August 26, 1968, a low-power license was issued on June 8, 1972, and a full-power license was issued on September 15, 1972. The plant design approval for the construction phase was based on the draft General Design Criteria (GDC) published by the AEC in the *Federal Register* (32 FR 10213) on July 11, 1967. The AEC published the final rule that added Title 10 of *Code of Federal Regulations* (10 CFR) Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," in the *Federal Register* (36 FR 3255) on February 20, 1971, with the rule effective on May 21, 1971. As stated in SECY-92-223, dated September 18, 1992 (ADAMS Accession No. ML003763736), the Commission decided not to apply the Appendix A GDC to plants with construction permits issued prior to May 21, 1971.

The Pilgrim Updated Final Safety Analysis Report (UFSAR) states that the AEC concluded in a May 20, 1968, safety evaluation that Pilgrim conforms to the draft GDC. The plant GDC is discussed in the UFSAR, Appendix F, "Comparison of Pilgrim Nuclear Power Station with the Proposed General Design Criteria Published by AEC for Public Comment in the *Federal Register*, July 11, 1967" (32 FR 10213).

Section 182a of the Atomic Energy Act requires applicants for nuclear power plant operating licenses to include TSs as part of the license application. The Commission's regulatory requirements related to the content of the TS are contained in 10 CFR 50.36(c). Section 50.36(c) of 10 CFR requires that the TSs include, among other things, items in the following categories: (1) safety limits, limiting safety systems settings, and limiting control settings; (2) limiting conditions for operation; (3) surveillance requirements; (4) design features; and (5) administrative controls.

The regulation at 10 CFR 50.36(c)(1)(i)(A) states, in part, that if any safety limit is exceeded, the reactor must be shut down. The safety limit MCPR is calculated on a cycle-specific basis because it is necessary to account for the core configuration-specific neutronic and thermal-hydraulic response.

The GDC 10, "Reactor Design," states that "the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences."

NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," provides guidance on the acceptability of the reactivity control systems, the reactor core, and fuel system design. Specifically, Section 4.4, "Thermal Hydraulic Design," provides guidance on the review of thermal-hydraulic design in meeting the requirement of GDC 10.

3.0 TECHNICAL EVALUATION

The licensee stated in its letter dated February 13, 2015 (Reference 2), that:

> Fuel design limits can be exceeded if the core exceeds critical power. Critical power is a term used for the power at which the fuel departs from nucleate boiling and enters a transition to film boiling. For BWRs, the critical power is predicted using a correlation known as the General Electric (GE) critical quality boiling length correlation (GEXL correlation). Due to core wide and operational variations, the margin to boiling transition is most easily described in terms of a critical power ratio (CPR), which is defined as the rod critical power as calculated by GEXL divided by the actual rod power. The more a CPR value exceeds 1.0, the greater the margin to boiling transition. The SLMCPR is calculated using a statistical process that takes into account operating parameters and uncertainties. The operating limit MCPR (OLMCPR) is equal to the SLMCPR plus a CPR margin for transients. At the OLMCPR, at least 99.9 percent of the rods avoid boiling transition during steady state operation and transients caused by a single operator error or equipment malfunction.

3.1 Cycle 21 Core

Pilgrim is a General Electric Type 3 BWR design. Pilgrim Cycle 21 core loading consists of 144 fresh Global Nuclear Fuel 2 (GNF2) fuel bundles, 152 once-burnt GNF2 fuel bundles, 152 twice-burnt GNF2 fuel bundles, and 132 thrice-burnt GNF2 fuel bundles.

3.2 Methodology

Global Nuclear Fuel developed the Pilgrim Cycle 21 SLMCPR values using the following NRC-approved methodologies and uncertainties:

- NEDC-32601-A, "Methodology and Uncertainties for Safety Limit MCPR Evaluations," (Reference 4)
- NEDC-32649P-A, "Power Distribution Uncertainties for Safety Limit MCPR Evaluations," (Reference 5)
- NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel (GESTAR II)" (Reference 6)

- NEDC-32505-A, "R-Factor Calculation Method for GE11, GE12 and GE13 Fuel," (Reference 7)

Plant specific use of these methodologies must adhere to certain restrictions.

3.2.1 Methodology Restrictions

Based on the review of topical reports NEDC-32601-A and NEDC-32649P-A and Amendment 25 to NEDE-24011-P-A, the NRC staff identified the following restrictions for the use of these topical reports:

- The TGBLA (lattice physics code) fuel rod power calculational uncertainty should be verified when applied to fuel designs not included in the benchmark comparisons of Table 3.1 of NEDC-32601-A, since changes in fuel design can have a significant effect on calculation accuracy.

- The effect of the correlation of rod power calculation uncertainties should be reevaluated to insure the accuracy of R-Factor uncertainty when the methodology is applied to a new fuel lattice.

- In view of the importance of MCPR Importance Parameter (MIP) criterion and its potential sensitivity to changes in fuel bundle designs, core loading and operating strategies, the MIP criterion should be reviewed periodically as part of the procedural review process to insure that the specific value recommended in NEDC-32601-A is applicable to future designs and operating strategies.

3.2.1.1 Restrictions (1) and (2)

In addressing restrictions (1) and (2) in a September 24, 2001, letter (ADAMS Accession No. ML012710272) from GNF to the NRC "Confirmation of 10x10 Fuel Design Applicability to Improved SLMCPR, Power Distribution and R-Factor Methodologies" (Reference 8), GNF stated that the rod power calculational uncertainties are dominated by geometrical considerations in which GE14 is identical to GE12. Global Nuclear Fuel has stated in the fuel bundle design information notice (Reference 9) that GNF2 fuel is designed for mechanical, nuclear, and thermal-hydraulic compatibility with the other 10x10 GNF fuel designs. The design has features of the currently operating GE10, GE11/13, and GE12/14 fuel including pellet-cladding interaction resistant barrier cladding, high performance spacers, part length rods, interactive thick corner/thin wall channel, and axial enrichment loading. The GNF2 design is a 10x10 array with 92 fuel rods and two large central water rods, fourteen part length fuel rods. The part length rod configuration improves efficiency and reactivity margins. Tables 2.1 and 3.1 of NEDC-32601-A provide a summary of SLMCPR uncertainties and pin power comparisons for a typical GE fuel design. The values given in the tables are representative of the values being calculated for GE14 and GNF2.

By letter dated February 25, 2015 (ADAMS Accession No. ML15051A032), the NRC staff asked the licensee to explain the differences in design and geometrical considerations between GNF2 and GE14 fuel. In response to the NRC staff's question (Reference 3), the licensee stated that GNF2 is an evolutionary fuel product based on GE14 and that it is not considered a new fuel design as it maintains the previously established 10x10 array. The NRC staff finds the licensee's response acceptable because there are only minor design differences, which does not affect the implementation of the NRC-approved methodology.

Based on the above discussion, the NRC staff has determined that the rod power calculational uncertainties used by GNF to develop the Pilgrim Cycle 21 SLMCPR values are valid for GNF2 fuel.

3.2.1.2 Restriction (3)

For Pilgrim Cycle 21, the minimum core flow SLMCPR calculation performed at 76.7 percent core flow and rated core power condition was limiting as compared to the rated core flow and rated core power condition. The analysis indicates that, at low core flows, the limiting rod pattern and the nominal rod pattern are essentially the same. In its report, GNF-001N8659-R1-NP, "GNF Additional Information Regarding the Requested Changes to the TS SLMCPR, Pilgrim Cycle 21," GNF determined that the rod pattern used to calculate the SLMCPR at 100 percent rated power and 76.7 percent rated flow reasonably assures that at least 99.9 percent of fuel rods in the core would not be expected to experience boiling transition during normal operation or anticipated operational occurrences during the operation of Pilgrim Cycle 21.

The NRC staff finds that the rod patterns used to calculate the SLMCPR at 76.7 percent of rated core flow and 100 percent of rated core power produce a limiting MCPR distribution that reasonably bounds the MCPR distributions that would be expected during the operation of the Pilgrim core throughout Cycle 21. Therefore, the NRC staff finds that the licensee's application demonstrates the validity of the criterion in restriction (3) (MIP criterion) for GNF2 fuel and the minimum core flow condition.

In summary, the NRC staff finds that the licensee has adequately addressed the restrictions of topical reports NEDC-32601-A and NEDC-32649P-A and Amendment 25 to NEDE-24011-P-A; and that the use of these reports to evaluate the Pilgrim Cycle 21 SLMCPR is acceptable.

3.3 Departures from NRC-Approved Methodology

No departures from NRC-approved methodologies were identified in the Pilgrim Cycle 21 SLMCPR calculations.

3.4 Deviations from the NRC-Approved Calculational Uncertainties

3.4.1 R-Factor

The R-Factor is an input into the GEXL correlation used to describe the local pin-by-pin power distribution and the fuel assembly and channel geometry on the fuel assembly critical power. The R-Factor uncertainty analysis includes an allowance for power peaking modeling uncertainty, manufacturing uncertainty and channel bow uncertainty. Global Nuclear Fuel has increased this uncertainty for all SLMCPR calculations to account for the potential impact of control blade shadow corrosion induced bow. Global Nuclear Fuel has generically increased the GEXL R-Factor uncertainty (Reference 7) to account for an increase in channel bow due to the emerging unforeseen phenomenon called control blade shadow corrosion-induced channel bow, which is not accounted for in the channel bow uncertainty component of the approved R-Factor uncertainty. The Pilgrim Cycle 21 analysis shows an expected channel bow which is bounded by a GEXL R-Factor uncertainty that accounts for channel bow uncertainty. Thus, the NRC staff finds that the use of a GEXL R-Factor uncertainty adequately accounts for the expected control blade shadow corrosion-induced channel bow for Pilgrim Cycle 21.

3.4.2 Core Flow Rate and Random Effective Traverse In-Core Probe Reading

Global Nuclear Fuel has committed to the expansion of the statepoints used in the determination of the SLMCPR (Reference 10). Consistent with the Reference 10 commitments, GNF performs analyses at the rated core power and minimum licensed core flow point in addition to analyses at the rated core power and rated core flow point. The NRC-approved SLMCPR methodology is applied at each statepoint that is analyzed.

The core flow and random traverse in-core probe (TIP) reading uncertainties used in single loop operation (SLO) minimum core flow SLMCPR analysis remain the same as in the rated core flow SLO SLMCPR analysis because these uncertainties, which are substantially larger than used in two loop operation (TLO) analysis, already account for the effects of operating at reduced core flow.

For TLO calculations performed at 76.7 percent core flow, the NRC-approved uncertainty values for the core flow rate (2.5 percent) and the random effective TIP reading (1.2 percent) are adjusted by dividing them by 76.7/100. The treatment of the core flow and random effective TIP reading uncertainties is based on the assumption that the signal to noise ratio deteriorates as core flow is reduced. Global Nuclear Fuel stated that this increase is conservative based on the expectation that the variability in the absolute flow will decrease as flow decreases. The NRC staff finds that this increase in the uncertainty bounds the original non-flow dependent uncertainties and, therefore, the NRC staff finds it acceptable for Pilgrim Cycle 21.

3.5 Core Monitoring System

For Pilgrim Cycle 21, the GNF 3D MONICORE System (Reference 11) will be used as the core monitoring system. The 3D MONICORE system is in widespread use throughout the GNF fueled fleet of BWRs like Pilgrim. Use of the current version of 3D MONICORE provides the plant capability to perform the reactivity anomaly surveillance. Use of 3D MONICORE has been previously evaluated and accepted by the NRC staff (Reference 11).

Therefore, the NRC staff finds the use of the GNF 3D MONICORE system for Pilgrim Cycle 21 to be acceptable.

3.6 NRC Staff's Evaluation

The NRC staff has determined that the proposed SLMCPR values were developed through the appropriate use of NRC-approved methodologies and guidelines. Therefore, the NRC staff finds that the SLMCPR values of $\geq$ 1.10 for two recirculation loop operation and $\geq$ 1.12 for single recirculation loop operation are acceptable.

4.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The Commission has made a final determination that the license amendment involves no significant hazards consideration. Under the Commission's regulations in 10 CFR 50.92, this means that operation of the facility, in accordance with the proposed amendment does not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. As required by 10 CFR 50.91 (a) in its application dated December 10, 2014, the licensee provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed SLMCPR, and its use to determine the Operating Cycle 21 thermal limits, have been derived using NRC approved methods specified in the Reference section of the Technical Specification Bases Section for 2.0 SAFETY LIMITS. These methods do not change the method of operating the plant and have no effect on the probability of an accident initiating event or transient.

   The basis of the SLMCPR is to ensure no mechanistic fuel damage is calculated to occur if the limit is not violated. The new SLMCPR preserves the margin to transition boiling, and the probability of fuel damage is not increased.

   Therefore, the proposed changes to Technical Specifications do not involve an increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed changes result only from the analysis for the Cycle 21 core reload using methods described in NEDE24011 P-A (GESTAR II). These methods have been reviewed and approved by the NRC, do not involve any new or unapproved method for operating the facility, and do not involve any facility modifications. No new initiating events or transients result from these changes.

   Therefore, the proposed changes to Technical Specifications do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   The margin of safety as defined in the TS bases will remain the same. The new SLMCPR was derived using NRC approved methods which are in accordance with the current fuel design and licensing criteria. The SLMCPR remains high enough to ensure that greater than 99.9% of all fuel rods in the core will avoid transition boiling if the limit is not violated, thereby preserving the fuel cladding integrity.

Therefore, the proposed changes to technical specifications do not involve a significant reduction in the margin of safety.

Based on the above evaluations, the NRC staff has concluded that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff has made a final determination that no significant hazards consideration is involved for the proposed amendment and that the amendment should be issued as allowed by the criteria contained in 10 CFR 50.91.

5.0 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.

6.0 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. 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 published in the *Federal Register* on March 12, 2015 (80 FR 13030). 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.

7.0 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.

8.0 REFERENCES

1. Letter from John A. Dent, Jr. (ENO) to NRC, "Proposed Change to the Pilgrim's Technical Specification Concerning the Safety Limit Minimum Critical Power Ratio," December 10, 2014 (ADAMS Accession Nos. ML14349A495 and ML14349A496).

2. Letter from John A. Dent, Jr. (ENO) to NRC, "Request for Supplemental Information Needed For Acceptance of Requested Licensing Action Regarding the Safety Limit

Minimum Critical Power Ratio License Amendment Request," February 13, 2015 (ADAMS Accession No. ML15050A245).

3. Letter from John A. Dent, Jr. (ENO) to NRC, "Response to Request for Additional Information Regarding the Safety Limit Minimum Critical Power Ratio License Amendment," March 11, 2015 (ADAMS Accession No. ML15075A033).

4. General Electric Nuclear Energy Topical Report, NEDC-32601-A, "Methodology and Uncertainties for Safety Limit MCPR Evaluations," August 1999 (ADAMS Accession No. ML14093A216).

5. General Electric Nuclear Energy Topical Report, NEDC-32649P-A, Power Distribution Uncertainities for Safety Limit MCPR Evaluations," August 31, 1999.

6. Letter from Brian R. More (GNF) to NRC, "Accepted Proprietary and Non-Proprietary Versions of Revision 20 to NEDE-24011–P, General Electric Standard Application for Reactor Fuel (GESTAR II), Main and United States Supplement," December 2013 (ADAMS Accession No. ML13352A465).

7. General Electric Nuclear Energy Topical Report, NEDC-32505-A, Revision 1, "R-Factor Calculation Method for GE11, GE12 and GE13 Fuel," July 1999 (ADAMS Accession No. ML060520636).

8. Letter from Glen A. Watford (GNF) to NRC, "Confirmation of 10x10 Fuel Design Applicability to Improved SLMCPR, Power Distribution and R-Factor Methodologies," September 24, 2001 (ADAMS Accession No. ML012710272).

9. Global Nuclear Fuels Topical Report, NEDO-31152, "Global Nuclear Fuels Fuel Bundle Designs", May 2007, Revision 9 (ADAMS Accession No. ML071510287).

10. MFN 04-081, Letter from J.S. Post (GE) to NRC, "Part 21 Reportable Condition and 60-Day Interim Report Notification: Non-Conservative SLMCPR," August 24, 2004 (ADAMS Accession No. ML042720293).

11. MFN-003-99, Letter, Frank Akstulewicz (NRC) to Glen A. Watford (GE), "Acceptance for Referencing of Licensing Topical Reports NEDC-32601P, Methodology and Uncertainties for Safety Limit MCPR Evaluations, NEDC-32694P, Power Distribution Uncertainties for Safety Limit MCPR Evaluation; and Amendment 25 to NEDE-24011-P-A on Cycle Specific Safety Limit MCPR," March 11, 1999 (TAC Nos. M97490, M99069, and M9749).

Principal Contributors: F. Forsaty
M. Chernoff

Date: May 6, 2015

May 6, 2015

Mr. John A. Dent, Jr.
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 THE MINIMUM CRITICAL POWER RATIO LICENSE AMENDMENT REQUEST (TAC NO. MF5431)

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

Sincerely,
/RA/
Nadiyah S. Morgan, Project Manager
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Docket No. 50-293

Enclosures:
1. Amendment No. 243 to License No. DPR-35
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cc w/encls: Distribution via Listserv

ADAMS Accession No.: ML15114A021 *See dated memo **NLO via email

OFFICE	DORL/LPLI-1/PM	DORL/LPLI-1/LA	DSS/SRXB/BC
NAME	NMorgan	KGoldstein	CJackson*
DATE	04/28/2015	04/28/2015	04/23/2015
OFFICE	DSS/STSB/BC	OGC	DORL/LPLI-1/BC (A)
NAME	RElliott	AGhosh**	MDudek
DATE	04/24/2015	05/01/2015	05/06/2015

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