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

March 28, 2011

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 REVISED TECHNICAL SPECIFICATIONS FOR SETPOINT AND SETPOINT TOLERANCE INCREASES FOR SAFETY RELIEF VALVES AND SPRING SAFETY VALVES (TAC NO. ME3453)

Dear Sir or Madam:

The Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 235 to Facility Operating License No. DPR-35 for the Pilgrim Nuclear Power Station (Pilgrim), in response to your application dated March 15, 2010 (Agencywide Documents Access Management System (ADAMS) Accession No. ML100770450), as supplemented by letters dated August 30, 2010 (ML102510137), September 21, 2010 (ML102740069), January 31, 2011 (ML110420305), and February 18, 2011 (ML110610397).

This amendment modifies the Pilgrim Technical Specifications to revise the setpoint and setpoint tolerances for Safety Relief Valves (SRVs) and Spring Safety Valves (SSVs) and to support the plant modifications associated with the replacement of (1) four Target Rock two-stage SRVs with three-stage SRVs, and (2) two existing Dresser 3.749 inch throat diameter SSVs with Dresser 4.956 inch throat diameter SSVs.

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. 235 to DPR-35
2. Safety Evaluation

cc w/encls: Distribution via Listserv

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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. 235
License No. 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 March 15, 2010, as supplemented by letters dated August 30, 2010, September 21, 2010, January 31, 2011, and February 18, 2011, 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. <u>Technical Specifications</u>

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 235, 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

Nancy L. Salgado

Nancy L. Salgado, 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: March 28, 2011

ATTACHMENT TO LICENSE AMENDMENT NO. 235

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
2-1	2-1
3/4.2-26	3/4.2-26
3/4.2-28	3/4.2-28
3/4.2-29	3/4.2-29
3/4.5-7	3/4.5-7
3/4.5-8	3/4.5-8
3/4.6-6	3/4.6-6
3/4.6-7	3/4.6-7

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 235, are hereby incorporated in the 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 FR27817 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.

2.0 SAFETY LIMITS

2.1 Safety Limits

2.1.1 With the reactor steam dome pressure < 785 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 ≥ 785 psig and core flow ≥ 10% of rated core flow:

MINIMUM CRITICAL POWER RATIO shall be ≥ 1.08 for two recirculation loop operation or ≥ 1.11 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.

PNPS
TABLE 3.2.F (Cont)

SURVEILLANCE INSTRUMENTATION

Minimum # of Channels	Operable Instrument Instrument #	Parameter	Type Indication and Range	Notes
2	TI-5021-2A TRU-5021-1A TI-5022-2B TRU-5022-1B	Suppression Chamber Water Temperature Suppression Chamber Water Temperature	Indicator/ Multipoint Recorder 30-230°F (Bulk) Indicator/ Multipoint Recorder 30-230°F (Bulk)	(1) (2) (3) (4)) (1) (2) (3) (4)
1	PID-5021	Drywell/Torus Diff. Pressure	Indicator - .25 - +3.0 psig	(1) (2) (3) (4)
1	PID-5067A PID-5067B	Drywell Pressure Torus Pressure	Indicator -.25 - +3.0 psig Indicator -1.0 - +2.0 psig	(1) (2) (3) (4)
1/Valve	(a) Primary or (b) Backup	Safety/Relief Valve Position	a) Acoustic monitor b) Thermocouple	(5)
1/Valve	(a) Primary or (b) Backup	Safety Valve Position Indicator	a) Acoustic monitor b) Thermocouple	(5)
2	LI-1001-604A LR- 1001-604A LI-1001- 604B LR-1001- 604B	Torus Water Level (Wide Range) Torus Water Level (Wide Range)	Indicator /Multipoint Recorder 0 - 300"H_2O Indicator /Multipoint Recorder 0 - 300"H_2O	(1) (2) (3) (4) (1) (2) (3) (4)

NOTES FOR TABLE 3.2.F

(1) With less than the minimum number of instrument channels, restore the inoperable channel(s) within 30 days.

(2) With the instrument channel(s) providing no indication to the control room, restore the indication to the control room within seven days.

(3) If the requirements of notes (1) or (2) cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the Cold Shutdown Condition within 24 hours.

(4) These surveillance instruments are considered to be redundant to each other.

(5) At a minimum, the primary or backup parameter indicators shall be operable for each valve when the valves are required to be operable. With both primary and backup instrument channels inoperable either return one (1) channel to operable status within 31 days or be in a shutdown mode within 24 hours.

The following instruments are associated with the safety/relief and safety valves:

Valve	Primary Acoustic Monitor	Backup Tail Pipe Temperature Thermocouple
203-3A	ZT-203-3A	TE6285
203-3B	ZT-203-3B	TE6286
203-3C	ZT-203-3C	TE6287
203-3D	ZT-203-3D	TE6288
203-4A	ZT-203-4A	TE6274-B
203-4B	ZT-203-4B	TE6275-B

(6) Deleted.

(7) With less than the minimum number of operable instrument channels, restore the inoperable channels to operable status within 7 days or prepare and submit a special report to the Commission within 14 days of the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the channels to operable status.

PNPS
TABLE 3.2-G

INSTRUMENTATION THAT INITIATES RECIRCULATION PUMP TRIP AND ALTERNATE ROD INSERTION

Minimum Number of Operable or Tripped Instrument Channels Per Trip System (1)	Trip Function	Trip Level Setting
2	High Reactor Dome Pressure	≤1210 psig
2	Low-Low Reactor Water Level	≥-46.3" indicated level

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

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

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

LIMITING CONDITIONS FOR OPERATION

3.5 CORE AND CONTAINMENT COOLING SYSTEMS

C. HPCI System

1. The HPCI system shall be operable whenever there is irradiated fuel in the reactor vessel, reactor pressure is greater than 150 psig., and reactor coolant temperature is greater than 365°F, except as specified in 3.5.C.2 below.

2. From and after the date that the HPCI system is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding 14 days unless such system is sooner made operable, providing that during such 14 days all active components of the ADS system, the RCIC system, the LPCI system and both core spray systems are operable.

3. If the requirements of 3.5.C cannot be met, an orderly shutdown of the reactor shall be initiated and the reactor shall be in the Cold Shutdown Condition within 24 hours.

SURVEILLANCE REQUIREMENTS

4.5 CORE AND CONTAINMENT COOLING SYSTEMS

C. HPCI System

1. HPCI system testing shall be as follows:

a. Simulated Automatic Actuation Test	Once/ Operating Cycle

---------------------- Note -----------------

Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform test.

----

b. Pump Operability	When tested as specified in 3.13, verify with reactor pressure ≤1035 and ≥ 940 psig, the HPCI pump can develop a flow rate ≥ 4250 gpm against a system head corresponding to reactor pressure.
c. Motor Operated Valve Operability	As Specified in 3.13

---------------------- Note -----------------

Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform test.

----

d. Flow Rate at 150 psig.	Once/ Operating Cycle, verify with reactor pressure ≤ 150 psig, the HPCI pump can develop a flow rate ≥ 4250 gpm against a system head corresponding to reactor pressure.

LIMITING CONDITIONS FOR OPERATION

3.5 CORE AND CONTAINMENT COOLING SYSTEMS

D. Reactor Core Isolation Cooling (RCIC) System

1. The RCIC system shall be operable whenever there is irradiated fuel in the reactor vessel, reactor pressure is greater than 150 psig, and reactor coolant temperature is greater than 365°F, except as specified in 3.5.D.2 below.

2. From and after the date that the RCIC system is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding 14 days unless such system is sooner made operable, providing that during such 14 days the HPCIS is operable.

3. If the requirements of 3.5.D cannot be met, an orderly shutdown of the reactor shall be initiated and the reactor shall be in the Cold Shutdown Condition within 24 hours.

SURVEILLANCE REQUIREMENTS

4.5 CORE AND CONTAINMENT COOLING SYSTEMS

D. Reactor Core Isolation Cooling (RCIC) System

1. RCIC system testing shall be as follows:

a. Simulated Automatic Actuation Test	Once/ Operating Cycle

---------------------- Note -----------------

Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform test.

----

b. Pump Operability	When tested as specified in 3.13, verify with reactor pressure ≤ 1035 and ≥ 940 psig, the RCIC pump can develop a flow rate ≥ 400 gpm against a system head corresponding to reactor pressure.
c. Motor Operability Valve Operability	As Specified in 3.13

---------------------- Note -----------------

Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform test.

----

d. Flow Rate at 150 psig.	Once/Operability Cycle verify with reactor pressure ≤ 150 psig, the RCIC pump can develop a flow rate ≥ 400 gpm against a system head corresponding to reactor pressure.

LIMITING CONDITIONS FOR OPERATION

3.6 PRIMARY SYSTEM BOUNDARY (Cont)

c. With no required leakage detection systems Operable, be in Hot Shutdown within the next 12 hours and in Cold Shutdown within the following 24 hours.

D. Safety and Relief Valves

1. During reactor power operating conditions and prior to reactor startup from a Cold Condition, or whenever reactor coolant pressure is greater than 104 psig and temperature greater than 340°F, both safety valves and the safety modes of all relief valves shall be operable.

2. If Specification 3.6.D.1 is not met, an orderly shutdown shall be initiated and the reactor coolant pressure shall be below 104 psig within 24 hours.

SURVEILLANCE REQUIREMENTS

4.6 PRIMARY SYSTEM BOUNDARY (Cont)

D. Safety and Relief Valves

1. As specified in accordance with 3.13, verify the safety function lift setpoints of the safety and relief valves as follows:

No. of S/R Valves	Setpoint (psig)
2 Safety	1280 ± 38.4
4 Relief	1155 ± 34.6

Following testing, lift setting shall be within ± 1%.

---------------- Note ------------------
Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform test.
----

2. Once/ Operating Cycle, verify each relief valve opens when manually actuated.

LIMITING CONDITIONS FOR OPERATION

3.6 PRIMARY SYSTEM BOUNDARY (Cont)

E. Jet Pumps

1. Whenever the reactor is in the Startup or Run Modes, all jet pumps shall be Operable. If it is determined that a jet pump is inoperable, the reactor shall be in Hot Shutdown within 12 hours.

SURVEILLANCE REQUIREMENTS

4.6 PRIMARY SYSTEM BOUNDARY (Cont)

E. Jet Pumps

NOTES

1. Not required to be performed until 4 hours after the associated recirculation loop is in operation.
2. Not required to be performed until 24 hours after >25% Rated Thermal Power.

Whenever there is recirculation flow with the reactor in the Startup or Run Modes, jet pump operability shall be checked daily by verifying at least one of the following criteria (1, 2, or 3) is satisfied for each operating recirculation loop:

1. Recirculation pump flow to speed ratio differs by ≤ 5% from established patterns, and jet pump loop flow to recirculation pump speed ratio differs by ≤ 5% from established patterns.
2. Each jet pump diffuser to lower plenum differential pressure differs by ≤ 20% from established patterns.
3. Each jet pump flow differs by ≤ 10% from established patterns.

UNITED STATES
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 235

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 March 15, 2010 (Agencywide Documents Access Management System (ADAMS) Accession No. ML100770450), as supplemented by letters dated August 30, 2010 (ML102510137), September 21, 2010 (ML102740069), January 31, 2011 (ML110420305), and February 18, 2011 (ML110610397), Entergy Nuclear Operations, Inc. (the licensee) submitted a request for changes to the Pilgrim Nuclear Power Station (Pilgrim) Technical Specifications (TSs). The supplements dated August 30, 2010, September 21, 2010, January 31, 2011, and February 18, 2011, 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 May 4, 2010 (75 FR 23812).

The proposed amendment would modify the Pilgrim TSs to revise the setpoint and setpoint tolerances for safety relief valves (SRVs) and spring safety valves (SSVs) and support the plant modifications associated with the replacement of (1) four Target Rock two-stage SRVs with three-stage SRVs, and (2) two existing Dresser 3.749 inch throat diameter SSVs with Dresser 4.956 inch throat diameter SSVs.

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, "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." Additionally, 10 CFR 50.36(c)(2)(ii) sets forth four criteria to be used in determining whether a limiting condition for operation (LCO) is required to be included in the TS. These criteria are: (1) installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant system (RCS) pressure boundary; (2) initial plant conditions that are assumed in a design-basis transient and accident analysis; (3) components or systems that are used for mitigating consequences of the design-basis transient and accident; and (4) components or systems which probabilistic risk assessment has shown to be significant to public health and safety. Furthermore, 10 CFR 50.36(c)(3) states, "Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions of operation will be met."

General Design Criterion (GDC) 13, "Instrumentation and Control," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50 requires that instrumentation be provided to monitor variables and systems and that controls be provided to maintain these variables and systems within prescribed operating ranges.

GDC 20, "Protection System Functions," of Appendix A to 10 CFR Part 50 requires that the protection system be designed to initiate the operation of appropriate systems to ensure that specified acceptable fuel design limits are not exceeded.

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. RG 1.105 endorses Part I of Instrument Society of America (ISA)-S67.04-1994, "Setpoints for Nuclear Safety Instrumentation," subject to NRC staff clarifications.

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 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. The NRC encourages the licensee to upgrade their TS to be consistent with those criteria, conforming to the extent practical, and consistent with the licensing basis for the plant to the current STS.

Pilgrim uses boiling water type nuclear steam supply systems furnished by General Electric. As noted above, since the STS were developed based on the criteria in 10 CFR 50.36(c)(2)(ii), 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 TECHNICAL EVALUATION

3.1 Licensee's Proposed Changes

In its application dated March 15, 2010, the licensee stated that the proposed modifications will result in the following system changes:

- Replace the Target Rock 2-stage SRVs with Target Rock 3-stage SRVs to improve the set pressure and leakage performance of the SRVs.

- Replace the SSVs with a similar larger capacity model to lower peak Anticipated Transient Without Scram (ATWS) pressure sufficiently to accommodate the setpoint and tolerance increases.

- Increase the as-found setpoint tolerance for the SRVs and SSVs from ± 1% to ± 3% to provide greater assurance that the valves will meet setpoint requirements.

- Raise the nominal set pressure of the SRVs by 40 psig (1115 psig to 1155 psig) to increase the simmer margin to 120 psi to reduce the vulnerability to SRV pilot leakage.

- Raise the nominal set pressure of the SSVs by 40 psig (1240 psig to 1280 psig) to maintain the current difference between the SRV and SSV setpoints to ensure that the SSVs will not open during any anticipated operational transients.

- Raise the High Reactor Steam Dome Pressure Trip Level Setpoint for Alternate Rod Insertion (ARI) and Recirculation Pump Trip (RPT) by 40 psig (1175 ± 5 psig to ≤ 1210 psig) to reduce likelihood of unnecessary ARI actuation and recirculation pump trips.

- Lower the High Reactor Pressure Feedwater Pump Trip Setpoint from 1415 psig to 1315 psig to support required ATWS mitigating functions.

- Increase the Reactor Steam Dome Pressure Limit from 1325 psig to 1340 psig to provide sufficient margin to accommodate variation in reactor pressure for future cycle specific over-pressure protection analysis.

- Increase Reactor Core Isolation Cooling (RCIC) System turbine and pump speed, power, and steam flow requirements to account for elevated reactor steam pressure. In addition, an increase is required to the RCIC Steam Line Break Detection Instrument Setpoint.

The licensee's proposed TS and Bases changes are summarized as follows:

TS 2.1.4 (TS page 2-1)

Reactor steam dome pressure is revised from ≤ 1325 to ≤ 1340 psig at any time when irradiated fuel is present in the reactor vessel.

BASES: 2.0 SAFETY LIMITS (Cont.) (Page B2-4)

BASES for REACTOR STEAM DOME PRESSURE (2.1.4) are revised consistent with TS 2.1.4.

TS Table 3.2.F (TS Page 3/4.2-26), and NOTES FOR TABLE 3.2.F (TS Page 3/4.2-28)

Tailpipe Temperature Indication; asterisk in Note (5) is removed; Note (6) is revised and relocated to Final Safety Analysis Report (FSAR), and the backup tailpipe temperature thermocouple. The existing SRV thermocouples (TE-6271, TE-6272, TE-6273, and TE-6276) contained in the current TSs 3/4.2 and Table 3.2.F and TS "NOTES FOR TABLE 3.2.F' are replaced with the functionally equivalent new thermocouples (TE-6285, TE-6286, TE-6287, and TE-6288). The new thermocouples will be listed in the NOTES FOR TABLE 3.2.F as a backup instrument for the acoustic monitors.

TABLE 3.2-G (TS Page 3/4.2-29), High Reactor Dome Pressure Trip Level Setting, and BASES (page B3/4.2-6)

High Reactor Dome Pressure Trip Level Setting "1175 ± 5 psig" is revised to "≤ 1210 psig."

SR 4.5.C (TS Page 3/4.5-7)

High-Pressure Coolant Injection (HPCI) system testing for HPCI pump operability surveillance is revised consistent with BWR Standard TS methodology.

SR 4.5.D.1 (TS Page 3/4.5-8) is revised

RCIC system testing for RCIC pump operability surveillance is revised consistent with BWR Standard TS methodology.

TS BASES 3/4.5 BASES (Page B3/4.5-21)

BACKGROUND, 3rd paragraph is revised for the Automatic Depressurization System (ADS) to reflect new values for SRVs.

TS 3.6.D.1, 3.6.D.2 NOTE and 3.6.D.3 (TS Page 3/4.6-6, TS 3.6.D.4 and D.5 (TS Page 3/4.6-7)

TS 3.6.D.1, 3.6.D.2 NOTE and 3.6.D.3 are revised to reflect the new SRV and SSV setpoints and adoption of BWR Standard TS surveillance methodology.

BASES: 3/4.6.D (Pages B3/4.6-7 and B3/4.6-8)

SRV and SSV BASES are revised to reflect the three stage SRVs and new SSVs and new BWR Standard TS surveillance methodology.

3.2 NRC Staff Evaluation

3.2.1 Background

The American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) require the reactor pressure vessel be protected from overpressure during upset conditions by self-actuated safety valves. The safety objective of the Pressure Relief System is to prevent over pressurization of the reactor vessel and the attached piping which forms the RCS. The Pressure Relief System at Pilgrim includes four (4) SRVs and two (2) SSVs. The SSVs discharge directly into the interior space of the drywell. The SRVs discharge through their individual discharge piping, terminating below the minimum suppression pool (torus) water level. All the SRVs and the SSVs will be replaced. These valves are installed on the main steam lines located in primary containment between the reactor pressure vessel and the main steam line flow restrictors.

The licensee will be replacing the existing four (4) Target Rock Model 7567F two-stage SRVs with Target Rock Model 0867F three-stage SRVs. The setpoints for the SRVs will be increased from 1115 psig to 1155 psig, and the allowable as-found setpoint tolerance will be increased from the current ±1% to ±3%. The new SRVs have the same bore size as the existing two-stage SRVs. Following as-found testing, the SRV setpoint will be restored to the current setpoint tolerence of ±1%. The licensee will also replace the existing two Dresser type 3777Q SSVs with new Dresser type 3707RR SSVs. The SSV setpoints will be increased from 1240 psig to 1280 psig, and the allowable as-found setpoint tolerance will be increased from the current ±1% to ±3%. The new SSV throat diameter will be increased from 3.749 inches to 4.956 inches for increased relief capacity. Following as-found testing, the SSV setpoint will be restored to the current setpoint tolerence of ±1%.

The use of ±1% allowable as-found SRV/SSV safety function lift setpoint tolerance in plant TS has been a generic industry issue. Nuclear power plant licensees have experienced difficulty in meeting the typical ±1% setpoint for SRV/SSVs. As a result, the BWR Owners' Group (BWROG) developed NEDC-31753P, "BWROG In-Service Pressure Relief Technical Specification Revision Licensing Topical Report," (Reference 3), to support the use of ±3% setpoint tolerance, which is consistent with ASME Code requirements (formally Section XI requirements). NEDC-31753P was reviewed and approved by the NRC staff as documented in its SE March 8, 1993 (Reference 4). In its SE (generic), the NRC staff determined that it is acceptable for licensees to submit TS amendment requests to revise the SRV code safety function lift setpoint tolerance to ±3%, provided that the setpoints for those SRVs tested are restored to ±1% prior to reinstallation. In addition, the SE required the following plant-specific analyses that must be performed by the licensees as conditions to implement TS changes to increase the SRV setpoint tolerances:

1. Transient analysis, using NRC approved methods, of abnormal (anticipated) operational occurrences (AOOs) as described in NEDC-31753P utilizing ±3% setpoint tolerance for the safety mode of the SRVs.

2. Analysis of the design basis overpressure event using the ±3% tolerance limit for the SRV setpoints to confirm that the vessel pressure does not exceed ASME pressure vessel code upset limits.

3. Plant specific analyses described in Items 1 and 2 should assure that the number of SRVs included in the analyses corresponds to the number of valves required to be operable in the TS.

4. Re-evaluation of the performance of high pressure systems (pump capacity, discharge pressure, etc.), motor-operated valves, and vessel instrumentation and associated piping considering the ±3% tolerance limit.

5. Evaluation of the ±3% tolerance on any plant specific alternate operating modes (e.g., increased core flow, extended operating domain, etc.).

6. Evaluation of the effects of the ±3% tolerance limit on the containment response during loss-of-coolant accidents (LOCAs) and the hydrodynamic loads on the SRV discharge lines and containment.

GE Hitachi Nuclear Energy (GEH) and Entergy Nuclear Operations, Inc. performed plant-specific analyses and evaluations for Pilgrim, and the results were provided in GEH Report, NEDC-33532P, "Pilgrim Nuclear Power Station Safety Valve Setpoint Increase," (Reference 5). This report was originally provided in the licensee's March 15, 2010, submittal as Revision 0, dated March 2010, and was later updated, in part, to include revised information on the Anticipated Transient Without Scram (ATWS) Trip Level Setting in response to NRC staff's RAI dated January 10, 2011 (Reference 7). Reference 5 addressed the impacts of the proposed changes, including (a) increased setpoints for SRVs and SSVs, (b) increased SRV and SSV ± 3% tolerance limits, and (c) increased SSV capacity. The results of these analyses are discussed below.

3.2.2 Anticipated Operational Occurrence (AOO)

A plant-specific review of the events in the updated final safety analysis report (UFSAR) was performed to determine if any of the AOOs are impacted by change to the safety and relief valve configuration (Reference 5). Based on the generic evaluation in NEDC-31753P (Reference 3) and the review of the transient and accident events, the overpressure analysis was evaluated with the SSVs and SRVs at the +3 % limit, and the Loss of Feedwater Event was reviewed with the SSVs and SRVs setpoints at the -3% limit. All other events were determined to be unaffected by the change in setpoint tolerance.

Reference 5 describes the effect of increasing both the SRV and SSV setpoints and setpoint tolerance and increasing the high pressure recirculation pump trip setpoint (high reactor pressure recirculation pump trip). The FSAR transient and accident event review demonstrate that if there is no SRV actuation, there will be no high reactor pressure recirculation pump trip. This is because the high reactor pressure recirculation pump trip was moved above the new SRV setpoint to be consistent with the current setpoints. The most limiting event for overpressure is the Main Steam Isolation Valve Closure with Flux scram (MSIVF); this event was analyzed, as discussed in Section 4.2 of this SE.

The evaluation considered the effect of lowering the ATWS high pressure feedwater pump trip. The feedwater pump trip has the potential to reduce the peak pressure due to the reduction in vessel level. None of the AOOs has a peak dome pressure high enough to reach the new ATWS high pressure feedwater pump trip setpoint. Thus, these events are not impacted by lowering that setpoint. The only event that can reach the high pressure feedwater pump trip is the MSIVF event. The MSIVF event is not an AOO, and is less likely to occur because of the additional failure of the MSIV position scram. For the MSIVF, the peak vessel and dome pressure occur so rapidly that the feedwater pump trip has no beneficial effect for this event.

The NRC staff reviewed the licensee's evaluation of the effects of increased SRV/SSV setpoints, increased SRV/SSV setpoint tolerance limit, increased SSV capacity, and the related changes on AOOs. Based on the above evaluation and on GEH's generic evaluation of AOOs included in NRC-approved NEDC-31753P, the proposed changes are acceptable for AOO events. This satisfies the first requirement of plant-specific analyses to be performed, as listed in the generic NRC staff SE.

3.2.3 Reactor Vessel Overpressure Protection

The ASME Code requires the reactor pressure vessel to be protected from overpressure during transient conditions by self-actuated safety valves. Pilgrim assumed all four SRVs and two SSVs operable in the safety mode. The safety limit for the reactor steam dome pressure was selected such that it is at a pressure below which it can be shown that the integrity of the RCS system is not endangered. The reactor pressure limit of 1340 psig as measured in the vessel steam dome was derived from the design pressures of the reactor vessel. The peak pressures for the piping systems connected to the reactor vessel was recalculated based on a reactor steam dome peak pressure of 1340 psig. These peak pressures for Pilgrim are below the lowest of the transient pressures permitted by the applicable design code: ASME Code (1965 Edition, including January 1966 Addendum) for the pressure vessel, USAS Piping Code Section B31.1 for the steam space piping, and ASME Code Section III for the RCS recirculation piping. The ASME Code permits pressure transients up to 10% over the vessel design pressure (110% x 1250 = 1375 psig). The USAS Piping Code Section B31.1 and ASME Code Section III permit pressure transients and other occasional loads whose combined effect do not exceed stress levels based on the duration of the loads and the applicable service limit.

The parameters of the SRVs which impact the transients include pressure setpoints, allowable tolerance limit of the setpoints, and rated relief capacity at set pressure. The as-left setpoint tolerance (± 1% of nominal setpoint) will not be changed by this modification. The nominal set pressure is increased by 40 psig to 1155 psig and the analyzed as-found setpoint tolerance is increased to ± 3%. The delay and stroke times will remain unchanged. The parameters of the SSVs include set pressure with tolerance and rated relief capacity at set pressure. The as-left setpoint tolerance (± 1% of nominal setpoint) will not be changed by this modification. The nominal set pressure is increased by 40 psig to 1280 psig and the analyzed setpoint tolerance is increased to ± 3%. The opening response time is unchanged. The new SSVs will have an increased bore size and larger outlet than the existing SSVs. The increased throat diameter of the SSVs increases the valve relief capacity. All SRVs and SSVs are required to be operable to satisfy the ASME over pressure analyses. The setpoints are established to ensure that the applicable code limits for peak reactor and coolant piping pressure are satisfied.

The most recent MSIVF transients (Cycle 18 Reload Licensing) for Pilgrim were analyzed with the current SSV/SRV configuration and high pressure recirculation pump trip setpoint. The reload evaluation includes both GE14 and GNF2 fuel. These results and those with the new configuration with the increase to the high pressure recirculation pump trip were provided in Reference 5. These results demonstrate that the reactor vessel dome pressure safety limit (1325 psig) and the peak vessel pressure limit (1375 psig) are met when analyzed with a 3% setpoint tolerance. The overpressure analyses were performed consistent with the methodologies described in GE Nuclear Energy, "General Electric Standard Application for Reactor Fuel (GESTAR), NEDE-24011-P-A-16, (Reference 6). These results show that the margin to the current dome pressure safety limit is less than 5 psi, but there is still sufficient margin to the ASME overpressure limit of 1375 psig with over 30 psi of margin. This shows that the dome pressure safety limit has excess conservatism. The safety limit is placed on the dome pressure to have a plant measurable parameter to demonstrate compliance with the vessel pressure limit of 1375 psig. These results show that there is less than a 20 psi difference between the peak vessel pressure and the peak dome pressure. The increase in dome pressure safety limit to 1340 psig has been proposed to assure that margin is available for cycle-to-cycle variation in cycle specific overpressure calculations while retaining margin in the vessel to dome pressure difference. Establishing a safety limit at 1340 psig provides margin to the ASME overpressure analysis and provides for a 35 psi pressure difference between the vessel bottom to dome. This 35 psi is 75% higher than the observed pressure difference (~20 psi) for the limiting ASME overpressure event.

The NRC staff reviewed the licensee's evaluation of the effects of increased SRV/SSV setpoints, increased SRV/SSV setpoint tolerance limit, increased SSV capacity, and the related changes on reactor vessel overpressure protection and finds that these changes do not prevent Pilgrim from maintaining reactor pressure below the ASME Code limit of 110% of vessel design pressure. Based on the above evaluation and NRC-approved methodology NEDE-24011-P-A, the proposed changes are acceptable for reactor vessel overpressure protection and shall accommodate the most severe pressurization transient. This satisfies the second and third requirements of plant-specific analyses to be performed, as listed in the generic NRC staff SE.

3.2.4 Impact on High Pressure Systems

The HPCI system ensures that the reactor core is adequately cooled to limit fuel clad temperature in the event of a small break in the nuclear system and loss of coolant which does not result in rapid depressurization of the reactor vessel. GEH analyses of the HPCI system for increased SRV setpoint of 1155 psig and ± 3% setpoint tolerance concluded that it is not necessary to increase the HPCI turbine and pump design speed and that adequate system margin exists to accommodate the SRV/SSV setpoint changes. The licensee proposes that since there are no system changes required, the current design conditions for the HPCI system are also unchanged.

The RCIC system provides makeup water at reactor pressures up to the new SRV setpoint. The Pilgrim-specific evaluation of the RCIC system due to the increased SRV setpoint of 1155 psig and ±3% drift concluded that it is required to increase the speed to 4628 RPM. Based on the evaluations of GEH, a design speed increase of up to 4700 RPM is acceptable. This speed increase requires more steam flow which impacts the high steam flow isolation setpoints. The increase in pressure and flow impacts the steam supply isolation, steam supply admission, and pump injection motor operated valves.

The licensee stated in its submittal, dated March 15, 2010, that RCS boundary valves are under review to assure that they will perform their function at the increased design pressure resulting from the SRV/SSV setpoint and setpoint tolerance increase and that all valves reviewed to date were successfully screened with the exception of the RCIC Pump Injection Valve (MO1301-49). This motor-operated valve (MOV) does not demonstrate sufficient margin based on a review of the weak link and torque/thrust analyses.

The NRC staff requested the licensee to provide a more specific plan to implement modification of MOVs. In its response dated September 21, 2010, the licensee stated that a design change package has been prepared to modify MO-1301-49 in Refueling Outage (RFO)18 to achieve increased thrust margin at torque switch trip. The existing weak link analysis for MO-1301-49 was reviewed in more detail, and it was determined that some excessively conservative assumptions could be revised in the calculation of the maximum allowed thrust for the limiting valve components. The result is an increase in the maximum allowed thrust. The overall gear ratio for the Limitorque actuator will be changed to increase the torque output, and therefore, the operator output thrust margin. The valve will continue to function within its prescribed operating times. Diagnostic testing will be performed to confirm the adequacy of the torque switch setting and the resultant margin at torque switch trip. The performance of MO-1301-49 will continue to be evaluated as part of Pilgrim Station's MOV periodic verification program. No other MOVs require modification in order to meet the new requirements imposed by the change in SRV and SSV setpoints and setpoint tolerances. The NRC staff found the licensee's response acceptable.

The licensee further stated that they will modify valve components or replace MOVs as a part of the SRV/SSV modification package to assure that sufficient margin exists prior to implementing the full modification package. Interfacing piping is also under review as a part of this project. To date, there are no interfacing piping issues identified that require modification. However, should any modifications be required they will be performed in accordance with the Pilgrim design and 10 CFR 50.59 processes.

The NRC staff reviewed the licensee's evaluation of the proposed changes on the impact on high pressure systems, and found that it was acceptable. This satisfies the fourth requirement of plant-specific analyses to be performed, as listed in the generic NRC staff SE.

3.2.5 Alternate Operating Modes

The licensee has evaluated the impact of increased setpoints, setpoint tolerance, and SSV capacity on Pilgrim-specific alternate operating modes (Reference 5). The alternate operating modes, including the Maximum Extended Load Line Limit Allowed (MELLLA), Increased Core Flow (ICF), Feedwater Temperature Reductions; Turbine Bypass Out-Of-Service (OOS), Main Steam Isolation Valve (MSIV) OOS and Single Loop Operation (SLO) were considered in determining the most restrictive analytical conditions (i.e., the most limiting operating mode) for performing the analysis associated with the proposed TS change. Therefore, the impact of the proposed change on the Pilgrim-specific alternate operating modes has been explicitly addressed and determined to be acceptable. This satisfies the fifth requirement of plant-specific analyses to be performed, as listed in the generic NRC staff SE.

3.2.6 Containment Response Due to Dynamic Loads

The licensee evaluated the impact of SRV and SSV discharge loads on primary containment. The SRV discharge loads are defined by parameters that include:

- SRV discharge line (SRVDL) and containment geometry
- Water leg length in the SRVDL at the time of SRV opening
- SRV flow capacity and SRV opening pressure

Since an SRV setpoint increase and the setpoint tolerance will increase the SRV valve opening pressure, the SRV discharge dynamic loads will increase. The licensee has evaluated the SRV dynamic load increases for the associated piping and torus submerged structures and the evaluation concluded that all piping and structures were found to meet Code requirements.

The SSV discharge loads are defined by parameters that include the SSV flow capacity and SSV opening pressure. Since an SSV setpoint increase and setpoint tolerance will increase the SSV safety valve opening pressure, and an increase in the SSV throat size will increase the SSV flow capacity, the SSV dynamic loads are expected to increase. The licensee has evaluated the SSV dynamic loads for the associated piping and the evaluation concluded that all piping and structures were found to meet Code requirements.

The NRC staff reviewed the licensee's evaluation of the proposed changes on the impact on the containment response due to dynamic loads and finds that these changes do not invalidate the Code requirements for all piping and structures, and hence, are acceptable. This satisfies the sixth requirement of plant-specific analyses to be performed, as listed in the generic NRC staff SE.

3.2.7 ATWS Analysis

In addition to the analyses discussed above, the licensee performed the ATWS analysis to support the proposed TS changes, as discussed below. As described in Reference 5, the GE computer model ODYN is used for the reactor transient analysis, and the NRC has approved the application of ODYN to ATWS evaluations. The GE computer model STEMP04 is used for the suppression pool heatup analysis. The STEMP analytical models have been accepted by the NRC in previous applications and other ATWS analyses. STEMP calculates the temperature rise of the suppression pool due to SRV and SSV discharge.

ATWS analyses for the MSIVC and the Pressure Regulator Failure - Open (PRFO) events were evaluated at the Beginning of Cycle (BOC) and at the End of Cycle (EOC) exposure conditions at rated core power (2028 MWt) and minimum core flow (76.7% of rated). ATWS analyses were performed for the proposed SRV and SSV configuration changes, which include:

1. SRV setpoint increase of 40 psi and tolerance relaxation to +/- 3%
2. SSV setpoint increase of 40 psi and tolerance relaxation to +/- 3%
3. High reactor pressure recirculation pump trip setpoint increase of 40 psi
4. High reactor pressure feedwater pump trip setpoint decrease of 100 psi
5. SSV reference capacity increase from 644,501 lbm/hr to 1,126,200 ibm/hr at 1240 psig plus 3% accumulation.

The ATWS analysis for the proposed TS changes yielded similar results to previous ATWS analyses. The limiting peak vessel pressure was 1478 psig which is less than the ASME limit of 1500 psig (120% x 1250 = 1500 psig) for PRFO/BOC event, and the peak suppression pool temperature was 175.9 F for PRFO/EOC which is less than the allowed value of 185°F. The ATWS analysis was based on the standby liquid control (SLC) system delivering 39 gallons per minute (gpm) of 8.42% sodium pentaborate solution with a minimum Boron-10 (B-10) enrichment of 54.5 atom-percentage. These SLC system equipment parameters are the minimum required by Pilgrim TSs and provide a hot shutdown capability equivalent to 10 CFR 50.62 requirements to inject 86 gpm of 13% sodium pentaborate solution with a B-10 enrichment of 19.8 atom-percentage (natural enrichment). The SLC system includes two SLC pumps and each pump discharge has a separate relief valve installed to prevent system over-pressurization.

Only one out of the two SLC pumps is operated for ATWS mitigation. The licensee has reviewed both relief valve setpoints and verified that each of the relief valves will remain closed when the associated pump is in operation with the maximum lower plenum pressure of 1212 psig, as reported in Reference 5.

The NRC staff reviewed the licensee's evaluation of the effects of the proposed changes on the ATWS events and finds that these changes do not prevent Pilgrim from maintaining reactor pressure below the ASME Code limit of 120% of vessel design pressure and other requirements delineated in 10 CFR 50.62. Based on the above evaluation, the proposed changes are acceptable for ATWS events and shall accommodate the most severe ATWS event.

3.2.8 Setpoint Calculation for Trip Level Setting

In its letter dated January 31, 2011, the licensee responded to the NRC staff's request for additional information (Reference 7), and submitted the setpoint calculation for Trip Level Setting (usually called the allowable value (AV)), calculated field setpoint (usually called the nominal trip setpoint (NTSP)), and the no-adjust limit (usually called the as-left limit).

The as-found limit is usually calculated adding the effects of drift to the as-left limit. For this setpoint calculation, the licensee uses the as-found limit as the no-adjust limit, and hence, it is a conservative approach. The licensee has used 1220 psig as the analytical limit (AL) for this setpoint calculation.

The licensee calculated the total loop uncertainty (TLU) taking the square-root of the sum-of-the-squares (SRSS) of all random terms and adding the bias terms algebraically. By License Amendment No. 151, dated April 6, 1994, the NRC approved Pilgrim for a 24-month fuel cycle in accordance with Generic Letter (GL) 91-04, "Changes in Technical Specification Surveillance intervals to accommodate a 24-month Fuel Cycle," RG 1.105, and ISA-S67.04-1982. The licensee committed that the Pilgrim setpoint calculations and drift evaluations provide 95/95 tolerance limit as acceptance criteria for uncertainties used and are valid for 24-month fuel cycles. The licensee used a 25% extension in selecting the surveillance test intervals. Based on vendor supplied information, the licensee used $\pm$ 0.25% of span as the no-adjust limit (as-left) for pressure transmitters and $\pm$ 0.13% for master trip units and derived a combined no-adjust limit of 1.9 psig, which the NRC staff finds to be a reasonably low number for application range at about 1210 psig. Based on the above design guidance, the licensee calculated TLU as 16.43 psig, and by subtracting this number from the AL, the licensee derived the calculated field setpoint as 1203.6 psig. By adding drift and no-adjust limit terms by the SRSS methodology to NTSP, the licensee arrived at the Trip Level Setting (AV) of $\leq$ 1210 psig.

The NRC staff reviewed the setpoint calculation methodology submitted by the licensee and finds that the licensee has properly applied: (1) two-sided distribution in calculating TLU which is valid for tolerance distribution from -2σ to $+ 2\sigma$ and ensures 95/95 confidence level, (2) vendor supplied tolerance figures, and (3) the NRC-approved methodology for 24-month fuel cycle with a 25% extension. Based on these findings, the NRC staff concludes that the revised setpoint complies with the requirements of 10 CFR 50.36, RG 1.105, and RIS 2006-17, and is therefore, acceptable.

The licensee also provided plant procedures on surveillance tests of this instrumentation which indicates that the plant has adequate corrective actions when the instruments are found outside the no-adjust limit. Considering that the licensee is using the no-adjust limit as the as-found limit and that these limits have been established at a reasonably low value, the NRC staff finds the plant procedures to be consistent with the guidance provided in RIS 2006-17. The requirements of GDC 13 and 20 have also been satisfied.

The NRC staff evaluated the licensee's calculations and plant procedures for surveillance testing of the instruments affected by the proposed TS change to increase the High Reactor Dome Pressure from "1175± 5 psig" to ≤ 1210 psig" and finds that the proposed TS change will ensure proper operation of the affected instrumentation to initiate RPT and ARI before reaching the AL of 1220 psig Reactor Dome pressure. The requirements specified in Section 2.0 of this SE have therefore been satisfied.

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 (May 4, 2010 (75 FR 23812)). 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 K. Bronson (Entergy) to USNRC, "Revised Technical Specification for Setpoint and Setpoint Tolerance Increases for Safety Relief Valves (SRV) and Spring Safety Valves (SSV), and Related Changes," March 15, 2010 (ML100770450).

2. Letter from K. Bronson (Entergy) to USNRC, "Entergy Response to NRC Request for Additional Information dated July 21, 2010, in support of Proposed License Amendment

for Pilgrim Setpoint and Setpoint Tolerance Increases for Safety Relief Valves (SRV) and Spring Safety Valves (SSV), and Related Changes," September 21, 2010 (ML102740069).

3. NEDC-31753P, "BWROG In-Service Pressure Relief Technical Specification Revision Licensing Topical Report," dated February 1990.

4. Letter from A.C. Thadani (NRC) to C. L. Tully (BWR Owners' Group), "Acceptance for Referencing of Licensing Topical Report NEDC-31753P, 'BWROG In-Service Pressure Relief Technical Specification Revision Licensing Topical Report,' (TAC No. M79265) dated March 8, 1993.

5. GE Hitachi Nuclear Energy Report, NEDC-33532P, "Pilgrim Nuclear Power Station Safety Valve Setpoint Increase", Rev. 2, January 2011 (ML110420307).

6. GE Nuclear Energy, "General Electric Standard Application for Reactor Fuel (GESTAR), NEDE-24011-P-A-16, October 2007.

7. Letter from R.V. Guzman (NRC) to Site Vice President, Pilgrim Nuclear Power Station, Entergy Nuclear Operations, Inc., "Request for Additional Information-Pilgrim Nuclear Power Station-To Support the Review of Setpoint and Setpoint Tolerance Increases for Safety Relief Valve and Spring Safety Relief Valve, (TAC No. ME3543), dated January 10, 2011 (ML110090027).

Principal Contributors: M. Razzaque
S. Mazumdar

Date: March 28, 2011

March 28, 2011

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 REVISED TECHNICAL SPECIFICATIONS FOR SETPOINT AND SETPOINT TOLERANCE INCREASES FOR SAFETY RELIEF VALVES AND SPRING SAFETY VALVES (TAC NO. ME3453)

Dear Sir or Madam:

The Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 235 to Facility Operating License No. DPR-35 for the Pilgrim Nuclear Power Station (Pilgrim), in response to your application dated March 15, 2010 (Agencywide Documents Access Management System (ADAMS) Accession No. ML100770450), as supplemented by letters dated August 30, 2010 (ML102510137), September 21, 2010 (ML102740069), January 31, 2011 (ML110420305), and February 18, 2011 (ML110610397).

This amendment modifies the Pilgrim Technical Specifications to revise the setpoint and setpoint tolerances for Safety Relief Valves (SRVs) and Spring Safety Valves (SSVs) and to support the plant modifications associated with the replacement of (1) four Target Rock two-stage SRVs with three-stage SRVs, and (2) two existing Dresser 3.749 inch throat diameter SSVs with Dresser 4.956 inch throat diameter SSVs.

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. 235 to DPR-35
2. Safety Evaluation

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ADAMS Accession No.: ML110650009
(*)SE provided by memo. No substantial changes made. NRR-106

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