May 9, 2003

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

SUBJECT: PILGRIM NUCLEAR POWER STATION - ISSUANCE OF AMENDMENT RE: APPENDIX K MEASUREMENT UNCERTAINTY RECOVERY - POWER UPRATE REQUEST (TAC NO. MB5603)

Dear Mr. Bellamy:

The Commission has issued the enclosed Amendment No. 201 to Facility Operating License (FOL) No. DPR-35 for the Pilgrim Nuclear Power Station (Pilgrim). This amendment is in response to your application dated July 5, 2002, as supplemented by letters dated September 27, November 6, November 21, and December 30, 2002; February 4, February 10, March 17, and April 14, 2003.

This amendment increases the licensed power level for Pilgrim by 1.5% from 1998 megawatts thermal (MWt) to 2028 MWt. The change is based on the installation of the Westinghouse/AMAG CROSSFLOW ultrasonic flow measurement instrumentation resulting in improved feedwater flow measurement accuracy. The amendment also approves application of the Independent Support Motion methodology for design modifications to the safety relief valve discharge line piping. The amendment changes the FOL and the Technical Specifications (TSs) to reflect the increased licensed power level. The associated TS Bases will also be revised.

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

Sincerely,

/RA/

Travis L. Tate, Project Manager, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-293

Enclosures: 1. Amendment No. 201 to License No. DPR-35
2. Safety Evaluation

cc w/encls: See next page

cc:

Resident Inspector
U. S. Nuclear Regulatory Commission
Pilgrim Nuclear Power Station
Post Office Box 867
Plymouth, MA 02360

Chairman, Board of Selectmen
11 Lincoln Street
Plymouth, MA 02360

Chairman, Duxbury Board of Selectmen
Town Hall
878 Tremont Street
Duxbury, MA 02332

Office of the Commissioner
Massachusetts Department of
Environmental Protection
One Winter Street
Boston, MA 02108

Office of the Attorney General
One Ashburton Place
20th Floor
Boston, MA 02108

Dr. Robert M. Hallisey, Director
Radiation Control Program
Commonwealth of Massachusetts
Executive Offices of Health and
Human Services
174 Portland Street
Boston, MA 02114

Regional Administrator, Region I
U. S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

John M. Fulton
Assistant General Counsel
Pilgrim Nuclear Power Station
600 Rocky Hill Road
Plymouth, MA 02360-5599

Mr. C. Stephen Brennion
Licensing Superintendent
Pilgrim Nuclear Power Station
600 Rocky Hill Road
Plymouth, MA 02360-5599

Mr. Jack Alexander
Manager, Reg. Relations and
Quality Assurance
Pilgrim Nuclear Power Station
600 Rocky Hill Road
Plymouth, MA 02360-5599

Mr. David F. Tarantino
Nuclear Information Manager
Pilgrim Nuclear Power Station
600 Rocky Hill Road
Plymouth, MA 02360-5599

Ms. Jane Perlov
Secretary of Public Safety
Executive Office of Public Safety
One Ashburton Place
Boston, MA 02108

Mr. Stephen J. McGrail, Director
Attn: James Muckerheide
Massachusetts Emergency Management
Agency
400 Worcester Road
Framingham, MA 01702-5399

Chairman
Nuclear Matters Committee
Town Hall
11 Lincoln Street
Plymouth, MA 02360

Mr. William D. Meinert
Nuclear Engineer
Massachusetts Municipal Wholesale
Electric Company
P.O. Box 426
Ludlow, MA 01056-0426

May 9, 2003

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

SUBJECT: PILGRIM NUCLEAR POWER STATION - ISSUANCE OF AMENDMENT RE: APPENDIX K MEASUREMENT UNCERTAINTY RECOVERY - POWER UPRATE REQUEST (TAC NO. MB5603)

Dear Mr. Bellamy:

The Commission has issued the enclosed Amendment No. 201 to Facility Operating License (FOL) No. DPR-35 for the Pilgrim Nuclear Power Station (Pilgrim). This amendment is in response to your application dated July 5, 2002, as supplemented by letters dated September 27, November 6, November 21, and December 30, 2002; February 4, February 10, March 17, and April 14, 2003.

This amendment increases the licensed power level for Pilgrim by 1.5% from 1998 megawatts thermal (MWt) to 2028 MWt. The change is based on the installation of the Westinghouse/AMAG CROSSFLOW ultrasonic flow measurement instrumentation resulting in improved feedwater flow measurement accuracy. The amendment also approves application of the Independent Support Motion methodology for design modifications to the safety relief valve discharge line piping. The amendment changes the FOL and the Technical Specifications (TSs) to reflect the increased licensed power level. The associated TS Bases will also be revised.

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

Sincerely,
/RA/
Travis L. Tate, Project Manager, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-293
Enclosures: 1. Amendment No. 201 to License No. DPR-35
2. Safety Evaluation
cc w/encls: See next page

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ADAMS Accession Numbers: ML031220007,
TS(s): ML031320786 , Package: ML031320794
*SE input provided - no major changes made. ** See previous concurrence.

OFFICE	PDI-2/PM	PDI-2/LA	EMCB/SC*	EMCB/SC**	EMEB/SC*	SRXB/SC*
NAME	TTate	CRaynor	LLund	SCoffin	KManoly	RCaruso
DATE	5/9/03	5/9/03	12/2/02	5/5/03	2/11/03	2/24/03

OFFICE	SPSB/SC*	EEIB/SC*	EEIB/SC**	OGC**	PDI-2/SC	PDI/PD	DLPM/AD
NAME	FReinhart	CHolden	EMarinos	RWeisman	JClifford	RLaufer	TMarsh
DATE	3/19/03	3/24/03	5/6/03	5/8/03	5/9/03	5/9/03	5/9/03

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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. 201
License No. DPR-35

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

   A. The application for amendment filed by Entergy Nuclear Operations, Inc. (the licensee) dated July 5, 2002, as supplemented on September 27, November 6, November 21, and December 30, 2002; February 4, February 10, March 17, and April 14, 2003, 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 Operating License and 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. 201, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of issuance and shall be implemented within 60 days. Prior to implementation of the license amendment, the licensee shall:

   A. Install the Westinghouse/AMAG Crossflow ultrasonic flow measurement (UFM) instrumentation system under the supervision of the vendor and in accordance with the vendor's recommendations as described in the July 5, 2002, application and evaluated in the NRC staff's associated Safety Evaluation dated May 9, 2003. The installation will be in accordance with the ABB Combustion Engineering Nuclear Power Topical Report CENP-397-P, Revision 1. The Crossflow UFM will be calibrated in accordance with the plant-specific piping configuration. The Updated Final Safety Analysis Report (UFSAR) shall be revised to reflect this in the next update of the UFSAR submitted to the NRC pursuant to 10 CFR 50.71(e).

   B. The Crossflow UFM system software will be controlled in accordance with the Entergy Software Quality Assurance program for Level B software. Plant procedures will be developed or revised, where appropriate, for the Crossflow UFM operation; maintenance; calibration; control of software and hardware system configuration; identifying, reporting, and correcting system deficiencies; and for addressing manufacturer-identified deficiencies as described in the July 5, 2002, application and evaluated in the NRC staff's associated Safety Evaluation dated May 9, 2003.

   C. Finalize the feedwater flow measurement uncertainty calculation based on installed operational data as described in the July 5, 2002, application and evaluated in the NRC staff's associated Safety Evaluation dated May 9, 2003. The UFSAR shall be revised to reflect this in the next update of the UFSAR submitted to the NRC pursuant to 10 CFR 50.71(e).

D. Modify the Safety Parameter Display System as described in the licensee's November 21, 2002, letter, and evaluated in the NRC staff's associated Safety Evaluation dated May 9, 2003. The UFSAR shall be revised to reflect this in the next update of the UFSAR submitted to the NRC pursuant to 10 CFR 50.71(e).

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Ledyard B. Marsh, Acting Director
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: May 9, 2003

ATTACHMENT TO LICENSE AMENDMENT NO. 201

FACILITY OPERATING LICENSE NO. DPR-35

DOCKET NO. 50-293

Replace the following pages of the Facility Operating License No. DPR-35 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	2

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
1-1	1-1
3/4.1-3	3/4.1-3
B3/4.1-2	B3/4.1-2
B3/4.1-3	B3/4.1-3

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 201 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 application dated July 5, 2002, as supplemented by letters dated September 27, November 6, November 21, and December 30, 2002; February 4, February 10, March 17, and April 14, 2003, Entergy Nuclear Operations, Inc. (ENO or the licensee), requested changes to the Pilgrim Nuclear Power Station (PNPS or Pilgrim), Facility Operating License (FOL) and Technical Specifications (TSs). The supplements dated September 27, November 6, November 21, and December 30, 2002; February 4, February 10, March 17, and April 14, 2003, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on September 3, 2002 (67 FR 56322).

The proposed changes would increase the licensed power level by 1.5% from 1998 megawatts thermal (MWt) to 2028 MWt. The proposed changes are based on installation of the Westinghouse/AMAG CROSSFLOW ultrasonic flow measurement (UFM) instrumentation, resulting in improved feedwater (FW) flow measurement accuracy. Specifically, the proposed changes would revise Paragraph 3.A in the FOL and the definition of DESIGN POWER in TS 1.0 to reflect the increased licensed power level. TS Table 3.1.1, Note 1D would also be revised to provide instructions for required power reductions related to turbine control valve fast closure and turbine stop valve closure trip functions. The associated TS Bases 3.1 would also be revised to reflect the changes associated with the new instrumentation and uprated power.

In addition, the application included a request for approval of the Independent Support Motion (ISM) methodology to address increased discharge loads associated with design modifications to the safety relief valves (SRVs) and associated piping.

2.0 REGULATORY EVALUATION

Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix K, as originally issued, required licensees to base their transient and accident analyses on an assumed power level of at least 102% of the licensed thermal power level. The original uncertainty assumption was mandated to account for uncertainties in determining thermal power. Specifically, the 2%

margin was intended to address uncertainties related to heat sources in addition to instrument measurement uncertainties.

On June 1, 2000, the U.S. Nuclear Regulatory Commission (NRC or Commission) issued a revision to 10 CFR Part 50, Appendix K, (65 FR 34913). The NRC staff concluded, at the time of the original emergency core cooling system (ECCS) rulemaking, that the 2% power margin requirement was based solely on considerations associated with power measurement uncertainty, as is reflected in Appendix K. The original regulation did not require licensees to demonstrate the power measurement uncertainty, but rather mandated a 2% margin, notwithstanding that the instruments used to calibrate neutron flux instrumentation may be more accurate than originally assumed in the ECCS rulemaking. The revised rule allows licensees to justify a smaller margin for power measurement uncertainty and to use an assumed power level less than 102% of the licensed power level, provided the new power level is demonstrated to account for uncertainties due to power level instrument error.

In its application, ENO requested approval to increase the PNPS licensed thermal power level based on the installation of the Westinghouse/AMAG CROSSFLOW UFM instrumentation for FW flow measurement. Use of the CROSSFLOW UFM provides more accurate measurements of the FW flow due to a reduced core-thermal-power uncertainty. The CROSSFLOW UFM system was previously reviewed and approved by the staff in the NRC's safety evaluation (SE) report for ABB Combustion Engineering Nuclear Power (ABB-CE) Topical Report CENP-397-P, Revision 1, "Improved Flow Measurement Accuracy Using Crossflow Ultrasonic Flow Measurement Technology," dated March 20, 2000. The staff concluded in its review of the generic topical report that the reduction in power measurement uncertainty does not constitute a significant change to the ECCS evaluation model as defined in 10 CFR 50.46(a)(3)(i).

3.0 TECHNICAL EVALUATION

The staff has reviewed the licensee's regulatory and technical analyses in support of its proposed license amendment which are described in Sections 3.0, 4.0, and 5.0 of the licensee's submittal.

The staff's review of the licensee's application is organized as follows:

3.1 Reactor Core and Fuel Performance
3.2 Reactor Coolant and Connected Systems
3.3 Engineered Safety Features
3.4 Instrumentation and Control
3.5 Electrical Power and Auxiliary Systems
3.6 Power Conversion Systems
3.7 Radwaste and Radiation Sources
3.8 Reactor Safety Performance Evaluations
3.9 Other Evaluations
3.10 FOL and TS Changes

3.1 Reactor Core and Fuel Performance

The licensee evaluates core thermal-hydraulic design and fuel performance characteristics for each reload fuel cycle. The following sections address the effect of the 1.5% thermal power optimization uprate on fuel performance, thermal limits, the power/flow map, and stability.

3.1.1 Fuel Design and Operation

Fuel bundles are designed to ensure that the following criteria are met:

(1) Fuel bundles are not damaged during normal steady-state operation and anticipated operational occurrences (AOOs).
(2) Any damage to fuel bundles will not be so severe as to prevent control rod insertion when required.
(3) The number of fuel rod failures during accidents is not underestimated.
(4) The coolability of the core is always maintained.

The use of NRC-approved fuel design acceptance criteria and analysis methodologies ensure that fuel bundles perform in a manner consistent with the applicable general design criteria (GDC) of Appendix A to 10 CFR Part 50. Fuel vendors perform thermal-mechanical, thermal-hydraulic, neutronic, and material analyses to ensure that the fuel design can meet the acceptance criteria during steady-state, AOOs, and accident conditions.

The licensee stated that a new fuel design is not needed for operation at the 1.5% power uprate to ensure safety. However, revised loading patterns, slightly larger batch sizes, and new fuel designs may be used to provide additional operating flexibility and maintain fuel cycle length. For the operating cycle (OC) in which the 1.5% power uprate will be implemented, the PNPS core will consist of General Electric (GE) 11 and GE 14 fuel designs, which were introduced for OC 14. The new fuel introduction analyses and the standard reload analyses establish the bases for ensuring that the fuel and the core will be designed, loaded and operated in accordance with NRC-approved fuel design acceptance criteria and meet all of the safety, licensing and regulatory requirements. Any new fuel designs that do not meet the NRC-approved acceptance criteria will require NRC review and approval. For PNPS, the fuel design acceptance criteria and the associated new fuel introduction analyses, methods and codes are based on the NRC-approved Amendment 22 to GESTAR II. The reload analyses performed to establish the core and fuel performance at the uprated conditions are based on an NRC-approved methodology, analytical methods, and codes specified in GESTAR II. The new fuel introduction analyses and the reload analysis will demonstrate that the fuel design for operation at the uprated power conditions meet NRC requirements.

The slightly higher operating power and the increased steam void content will affect the core and fuel performance and may necessitate an increase in the energy generated in the core for each cycle. The reactor core and fuel performance characteristics may also change, depending on the core design changes used to achieve the necessary additional energy. The slight increase in the necessary cycle energy can be met by increasing the bundle enrichment, increasing the reload batch fraction and/or by changes in the fuel loading pattern. Since the additional energy necessary for the 1.5% power uprate is small and will be obtained in accordance with NRC-approved methodologies, the power uprate core design changes will not significantly impact the reactor core and fuel performance.

Since the impact of the 1.5% power uprate on the fuel design and operation is not expected to be significant, and NRC-approved new fuel design and standard reload analyses will demonstrate the fuel design performance, the staff finds this approach will adequately ensure NRC requirements for fuel design are met for operations at the 1.5% power uprate.

3.1.2 Thermal Limits Assessment

As defined in Appendix A to 10 CFR Part 50, GDC 10 requires that the reactor core and the associated control and instrumentation systems must be designed with appropriate margin to ensure that the specified acceptable fuel design limits (SAFDLs) are not exceeded during normal operation, including AOOs. Operating limits are established to ensure that regulatory and/or safety limits are not exceeded for a range of postulated events (transients and accidents).

a. Minimum Critical Power Ratio (MCPR) Safety and Operating Limit

The safety limit minimum critical power ratio (SLMCPR) ensures that 99.9% of the fuel rods are protected from boiling transition during steady-state operation. The operating limit minimum critical power ratio (OLMCPR) ensures that the SLMCPR will not be exceeded as a result of an AOO.

Cycle-specific core configurations are evaluated before each reload to establish the cycle-specific operating limits. The transient analyses used to establish the OLMCPR are discussed in Section 3.8 of this SE. The licensee will establish the OLMCPR based on the reload transient analysis for the power uprate before implementation. Section 5.3.3 and Appendix E of the thermal power optimization licensing topical report, NEDC-32938 (TLTR), provides the bases of GE Nuclear Energy's (GENE's) transient analysis disposition, including the power level used in the transient analysis for different events.

Based on the predicted change in the OLMCPR for the power uprate, the use of NRC-approved analytical methods and codes, and the fact that the core-specific OLMCPR will be established during the first reload analyses for the 1.5% power uprate, the staff finds that determining the OLMCPR at the uprated power conditions during the reload analysis is acceptable.

The 1.5% power uprate will result in a slight decrease in the steady-state operating MCPR, with no changes in the rod pattern, fuel design, or core design. For the GENE methods, the plant-specific SLMCPR is confirmed during every reload using Monte Carlo analyses that assume the core is operating at the thermal limits. Such analyses assume a core power distribution that maximizes the number of assemblies operating near these limits. The SLMCPR will be calculated for the uprated power implementation fuel cycle and confirmed for each subsequent cycle. Any changes to the SLMCPR limits specified in the TS will also require staff review and approval. Since the SLMCPR will be calculated using appropriate cycle-specific inputs with NRC-approved analytical methods, codes, and any changes to the codes will be submitted to the NRC for review, the staff finds calculating the SLMCPR during the reload analysis for the power uprate is acceptable.

b. Maximum Average Planar Heat Generation Rate (MAPLHGR) and Maximum Linear Heat Generation Rate (MLHGR) Operating Limits

The MAPLHGR and MLHGR limits ensure that the plant does not exceed regulatory limits established in 10 CFR 50.46 or by the fuel design limit. The MAPLHGR limit is determined by analyzing the limiting loss-of-coolant accident (LOCA). The MLHGR limits are determined by the fuel rod thermal-mechanical design.

The licensing basis LOCA analyses have historically been performed at 102% of the current licensed thermal power (CLTP); therefore, the effect of the 1.5% power uprate on the MAPLHGR operating limit is expected to be comparable to the cycle-to-cycle variation. The licensee conducts reload analyses for each core reload to confirm that the MAPLHGR and the MLHGR operating limits for each fuel bundle design will remain within the MAPLHGR limit established by the ECCS-LOCA analysis and the fuel-specific MLHGR limit. In addition, the licensee will also confirm the adequacy of the applicable power- and flow-dependent MCPR and MLHGR limits for the uprated operating range during reload analyses following the 1.5% power uprate.

The licensee stated that the fuel and core performance and thermal limits assessments will be deferred to the reload analyses that will be performed before the implementation of the power uprate. The staff has reviewed the licensee’s stated approach and determined that with the licensee using appropriate cycle-specific inputs with approved codes as discussed above, the plant reload analysis will ensure that there are acceptable margins between the licensing limits and the corresponding operating limits, and that the core design changes for the uprate will not significantly impact reactor core and fuel performance. Therefore, the staff finds that the licensee’s proposal to perform the thermal limits assessment during the reload analysis is acceptable for ensuring MAPLHGR and MLHGR limits are met.

3.1.3 Reactivity Characteristics

The licensee is required to maintain a sufficient shutdown margin (SDM) to meet the PNPS TS requirements. The licensee calculates the required SDM on a cycle-specific basis. The licensee stated in their July 5, 2002, application that they will use NRC-approved analytical methods, codes and licensing methodology to demonstrate that cycle-specific core design meets the SDM conditions and to calculate the boron concentration required to achieve cold shutdown conditions. This TS requirement remains the same for the subsequent cycles and core designs. Therefore, the staff finds the licensee’s disposition of the reactivity characteristic assessment to the cycle-specific reload analysis acceptable.

Power/Flow Operating Map

The 1.5% power uprate will maintain the previously licensed rod line, and the power uprate will be achieved by increasing the core flow along the licensed Maximum Extended Load Line Limit Analysis (MELLLA) rod line. Since previous safety analyses support operation along the licensed operating domain up to 102% power, the licensee stated that the 1.5% power uprate is limited to the previously established and analyzed operating domain and rod line up to the uprated power level. Maintaining the previously licensed rod line also ensures that the reactor power is not increased at the low-flow conditions in order to avoid changes in the plant’s response to anticipated transient without scram (ATWS) and instability. Extending the licensed rod line up to the uprated power level will slightly reduce the core flow range in the full power portion of the operating window. Figure 1-1 of the PNPS report, NEDC-33050, Revision 1, “Safety Analysis Report for Pilgrim Nuclear Power Station Thermal Power Optimization” (TSAR), submitted as an attachment to the July 5, 2002, application provides a power/flow map that demonstrates the 1.5% uprate strategy and the associated instrumentation actuation. Both the power/flow map and the reactor heat balance in Table 1-2 of the TSAR substantiates that the previously-licensed upper rod line is maintained but extended up to the uprated power level. The TSAR power/flow map also identifies the pre- and post-uprate power. The staff reviewed the PNPS TSAR and determined that it clearly specifies the uprate approach on the power/flow map. The staff has confirmed that the uprate will be accomplished along the currently licensed rod line, and the staff finds the defined operating domain is acceptable.

3.1.4 Stability

Appendix A to 10 CFR Part 50, GDC 12, “Suppression of Reactor Power Oscillations,” states that “the reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding SAFDLs are not possible or can be reliably and readily detected and suppressed.”

The long-term stability solutions for boiling water reactors (BWRs) are discussed in NEDO-31960-A, “BWR Owners' Group Long-Term Stability Solutions Licensing Methodology,” dated April 1996, and in NEDO-32339(A), “Licensing Topical Report, Reactor Stability Long-Term Solution: Enhanced Option 1-A,” dated December 1996. In accordance with the licensing methodology specified in GESTAR II, the stability responses of new fuel designs are evaluated during the fuel introduction analyses. The licensee stated in the July 5, 2002, application that the PNPS evaluations are performed in accordance with this methodology for each fuel cycle to ensure that the applicable long-term solution stability criteria are met.

For the 1.5% power uprate, the licensee will maintain the currently-licensed highest flow control line. Therefore, the high-power/low-flow portion of the power flow map susceptible to instabilities will be unchanged. The licensee stated that PNPS utilizes stability Enhanced Option I-A (E1A), which is a prevention solution. The E1A solution relies on an Exclusion, Restricted, and Administratively Controlled Monitored Regions. This option uses an analytically-based flow-biased Average Power Range Monitor (APRM) flux scram and control rod block trip functions, in conjunction with administrative monitoring, to provide stability protection. The licensee stated that they will confirm or adjust stability regions during the reload analyses. The staff compared the licensee’s proposed operating strategy to the E1A solutions and determined that the strategy comports with the E1A solutions, and the methodology will ensure stability protection for the uprated conditions. For PNPS, the MELLLA upper rod line will not change in the high-power/ low-flow region of the operating domain for operation at the uprated power level, and the reload stability evaluation will ensure acceptable stability performance and protection for future cores operating at the uprated conditions. Therefore, the staff finds that the NRC-approved E1A will continue to provide stability protection at the uprated conditions.

3.1.5 Reactivity Control

Control Rod Drives (CRDs) and CRD Hydraulic System

The licensee evaluated the control rod drive mechanisms (CRDMs) by comparing the proposed parameters to those in the design basis analysis. The licensee indicated that the reactor vessel's operating and design pressure and temperature that are used in the existing design basis analysis remain bounding for the proposed 1.5% power uprate. The licensee concluded that the existing PNPS design basis for stresses and fatigue cumulative usage factors of the CRD mechanisms are not affected by the proposed 1.5% power uprate condition.

The staff reviewed the operating parameters for the CRDMs against the uprated conditions and determined that the CRDMs will operate within the existing parameters and the values of the design basis analysis for the proposed 1.5% power uprate condition.

3.1.6 Reactor Core and Fuel Performance - Conclusion

Based on the evaluation presented in Sections 3.1.1 through 3.1.5 of this SE, the staff concludes that the proposed 1.5% power uprate is acceptable with respect to its impact on the reactor core and fuel performance.

3.2 Reactor Coolant and Connected Systems

3.2.1 Nuclear System Pressure Relief/Overpressure Protection

The SRVs provide reactor overpressure protection for the nuclear steam supply system (NSSS) to prevent failure of the nuclear system pressure boundary and uncontrolled release of fission products. The SRV setpoints are established to provide the reactor overpressure protection function, while ensuring that there is adequate margin between the reactor operating pressure and the SRV actuation setpoints to prevent unnecessary SRV actuations during normal plant maneuvers.

For the 1.5% power uprate, the licensee will not increase the SRV setpoints or change the number of SRVs out-of-service (OOS) assumed in the American Society of Mechanical Engineers (ASME) overpressure analysis. However, the licensee will increase the total SRV capacity by modifying valve internals to increase the existing SRV throat diameters. This modification is expected to result in a 7.5% increase in the SRV capacity. The ASME Code allowable peak pressure for the reactor vessel is 1375 psig (110% of the design pressure of 1250 psig), which is the acceptance limit for pressurization events. The licensee analyzed AOOs that may result in the largest overpressure transient on a cycle-specific basis, taking into account the power uncertainty. The most limiting overpressure transient event for PNPS is the main steam isolation valve closure event with high neutron flux scram. The licensee stated that the calculated peak ASME overpressure is 1301 psig, based on the current SRV capacity and 102% of the CLTP. The staff has reviewed the results of the licensee's analysis and determined that this analysis bounds operation at the proposed 1.5% power uprate level. Therefore, the staff finds that, with an increased SRV flow, sufficient capacity is maintained to meet the ASME overpressure limit at the uprate condition.

Evaluation of SRV Discharge Line Seismic Analysis Methodology

In its application dated July 5, 2002, the licensee included a proposed new piping analysis methodology to be used for the analysis of the SRV discharge line (SRVDL) at Pilgrim in support of the 1.5% power uprate. This proposed new methodology was further discussed with the staff during a meeting on July 24, 2002.

The current licensing basis (CLB) for seismic analysis of Class 1 piping for Pilgrim is based on the application of the envelope response spectrum (ERS) method, taken as the envelope of the spectra about or near the center of gravity of the piping system (42 ft). As stated in the Updated Final Safety Analysis Report (UFSAR), paragraph 12.2.3.5.4, the method uses 0.5% damping for the operating basis earthquake (OBE) response spectra and 1% damping for the safe shutdown earthquake (SSE) response spectra. However, for the analysis of the effects on the SRVDL piping and supports due to increased SRV flow and seismic loading, the licensee proposed the application of the ISM response spectrum analysis method, described in the UFSAR Revision 5, July 1985, paragraph 12.2.3.5.5. This method was previously used at Pilgrim only for seismic analysis of replacement recirculation residual heat removal (RHR) and reactor water cleanup system (RWCU) piping. The ISM response spectra for RHR and RWCU piping were applied at the support attachment locations, and were based on 2% damping for OBE spectra and 3% damping for SSE spectra. For the power uprate application, the licensee combined modal responses in accordance with the provisions of Regulatory Guide (RG) 1.92. The licensee's analysis did not specify the method that was used for combining modal responses due to individual or group support motion.

The staff's position for the application of ISM methodology was stated in NUREG-1061, Volume 4, dated December 1984. Due to the approximate nature of the technique, the calculated results depend on the method by which the modal responses are combined; in particular, the modal responses for groups of supports. The staff discussed two approaches that may be considered for the combination of group modal responses: square-root-of-the-sum-of-the-squares (SRSS) or absolute sum (ABS). Although considered conservative, the staff's position as stated in NUREG-1061 is that the combination between groups should be by ABS. The staff based its position on the evaluation of various piping analyses using the ISM method and the SRSS or ABS combinations, and comparison with results using time-history methods. The calculated results of any static and dynamic analysis of a structure represent estimates of the true stresses and loads in the structure for the given loading conditions. The results of using the ABS combination were shown to always overestimate the results based on time-history calculation, while the results of using the SRSS combination occasionally underestimated the time-history results. The staff adopted the ABS combination on the basis of the criterion that design stresses and support loads calculated by using any analytical method should not underestimate the stresses and loads as they may exist in the as-built structure under the prescribed loading conditions, in particular since there are many uncertainties regarding the construction of the structure, the soil-structure interaction, and the representation of the actual loading conditions. The licensee contends that the ABS combination method generates overly conservative estimates of stresses and loads. The licensee, therefore, proposed to use the ISM method, with the SRSS combination, between group modal responses and RG 1.61 damping, using the computer program ADLPIPE. The licensee stated that this program has the capability for performing ISM response spectrum analysis.

By letter dated November 6, 2002, the licensee provided responses to a request for additional information (RAI). In its response, the licensee provided the user's manual and benchmarking calculations for the ADLPIPE program. The staff has reviewed this information and finds that, based on experience with prior industry programs and staff engineering judgment, the ADLPIPE program is acceptable for application to ISM analysis.

Regarding the proposed use of the ISM method with response spectra based on RG 1.61 piping damping instead of the damping stated in the CLB, the licensee stated that RG 1.61 damping was shown in industry studies to be overly conservative as compared to experimentally determined damping in plant piping systems. However, experimental/analytical seismic studies and tests performed under the Electric Power Research Institute (EPRI) auspices on a typical nuclear piping system geometry at the University of California at Berkeley Earthquake Engineering Research Center shake table, and documented in EPRI Report NP-6153 "Seismic Analysis of Multiply Supported Piping Systems," dated March 1989, indicate that in the great majority of these tests, a number of support loads, as calculated by applying the ISM method with the SRSS group combination and 2% damping (equivalent to RG 1.61 damping for OBE), were significantly underestimated as compared to the recorded test loads. Based on these EPRI sponsored tests, the staff concluded that using the ISM method with the SRSS group combination would only be acceptable with lower damping, such as UFSAR damping. The staff therefore concluded that the licensee did not provide an acceptable basis for using response spectra based on RG 1.61 damping in combination with the ISM method with SRSS group combination of modal responses.

To address the staff's concerns, the licensee, in its November 6, 2002, response to the staff's RAI, provided a parametric study of a piping model consisting of Main Steam (MS) Line A and SRVDL A, subjected to the load combination of pressure, dead-weight, increased SRV loads and licensing basis SSE loads. This model is considered representative of the three MS lines and their associated SRVDLs. The study investigated the effects of various sets of inputs and analysis methods and parameters to determine the sensitivity and impact on the pipe stresses and support loads. The stresses and support loads due to these external loads were combined by the SRSS method, subject to meeting the ASME Code Service Level C pipe stress and support (snubber) allowables.

In the parametric study, the licensee analyzed the piping model with the CLB ERS approach, and with various combinations of ERS or ISM methods, ISM group combinations using the ABS or SRSS approaches, UFSAR or RG 1.61 damping spectra, and 2-directional (2D) or 3-directional earthquake combinations. The licensee reported the results for 11 cases, including the two largest pipe stresses for each piping model. The MS system model contained three snubbers and the SRV downcomer lines contained 8 snubbers.

The licensee's analyses determined that the CLB approach, consisting of the center-of-gravity ERS approach based on UFSAR SSE damping and a 2-D earthquake combination, provided the smallest increase in pipe stresses and support loads. This resulted in four supports requiring modifications for the MS Line A and SRVDL A. However, performing analysis using the same ERS approach, which included the response spectra based on UFSAR SSE damping at the upper level of the reactor vessel (86.9 ft), resulted in increased pipe stresses and support loads and resulted in eight supports requiring modification. All other cases in the study also resulted in higher pipe stresses and support loads as compared to the CLB approach, except one.

The staff reviewed the licensee's evaluation and finds that, in general, the staff would accept the application of the ISM/SRSS group combination methodology when used with RG 1.60 spectrum and RG 1.61 damping. Although the design spectrum with FSAR damping is not as conservative as a spectrum developed based on RG 1.60, the staff finds the licensee demonstrated, in its parametric study, that the application of the ISM/SRSS group combination methodology with RG 1.61 damping result in modifications that exceed those necessary if the analysis is based on the CLB piping design methodology. Therefore, the staff finds that proposed methodology using the ISM/SRSS group combination methodology and RG 1.61 damping is more conservative than the CLB methodology. The staff concludes that the licensee's application of the ISM method and modifications are acceptable for this plant-specific case since the licensees's proposed modifications are more conservative compared to the modifications that would be necessary based on the CLB methodology.

The staff notes that its acceptance of the licensee's plant-specific application of the ISM method and associated modifications to the SRVDL piping for this power uprate in a manner inconsistent with the staff's position as stated in NUREG 1061, Volume 4, applies only to Pilgrim specifically for the SRVDL modifications, and does not constitute an endorsement for its generic use at other facilities.

3.2.2 RPV and Internals

Fracture Toughness

Appendix G to 10 CFR Part 50 contains the fracture toughness requirements for ferritic materials of pressure-retaining components of the reactor coolant pressure boundary (RCPB). Appendix G requires, with certain exceptions not relevant here, that beltline materials maintain Charpy upper-shelf energy of no less than 50 foot-pounds (ft-lb) throughout the life of the vessel. Table 1 of Appendix G contains the pressure-temperature (P-T) limits and minimum temperature requirements that are defined by the operating conditions of the reactor vessel.

Appendix H to 10 CFR Part 50 contains the requirements for the RPV material surveillance program. Appendix H requires that RPVs with peak neutron fluence exceeding 10^{17} n/cm^2 at the end of the design life have their beltline materials monitored by a surveillance program complying with American Society for Testing and Materials (ASTM) E 185, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," as modified by Appendix H. The design of the surveillance program and the capsule withdrawal schedule must meet the requirements of the edition of ASTM E 185 that is current on the date on which the RPV was purchased.

The staff reviewed the fracture toughness analyses of the PNPS RPV presented in the licensee's application. The licensee calculated the end-of-life (EOL) fluence for the 1.5% power uprate and the fluence for current conditions was used to evaluate the vessel against the requirements of 10 CFR Part 50, Appendices G and H. The results of the staff's review are provided below:

1. The staff performed confirmatory calculations, and agrees that the minimum EOL upper shelf energy (USE) for beltline materials is above 85 ft-lb through the end of the facility's

current operating license (32 effective full power years (EFPYs)). Therefore, the PNPS RPV beltline materials comply with the USE requirements in 10 CFR Part 50, Appendix G.

2. The staff determined that the licensee's calculated PNPS P-T curves for 32 EFPYs would remain valid until 27 EFPYs at the 1.5% power uprate level. The RPV surface fluence will increase as a result of an increase in the power level. However, the net effect in 1/4T fluence at 32 EFPYs is negligible for operation at the 1.5% power uprate conditions. In evaluating this 1/4T fluence contribution to the resulting adjusted reference temperature (ART), the staff determined that there is little change to the ART for EFPYs up to, and including, 27 EFPYs. PNPS's existing P-T curves account for a shift value of 105°F and will not change at 27 EFPYs for operation at the proposed uprate condition.

   Due to staff concerns that the existing PNPS plant-specific calculations for the original fluence value were outdated, the existing P-T curves were limited to use through OC 14 (~19 EFPYs) when the staff issued license amendment (LA) 190 on April 13, 2001. On March 28, 2003, the staff issued LA 197 that approved the use of the existing P-T curves with an assumed 48 EFPY fluence through OC 16 (approximately 23 EFPYs). The use of the existing P-T curves calculated with a 48 EFPY fluence provides additional conservatism to ensure RPV integrity will be maintained. The proposed 1.5% power uprate will be implemented during OC 15. Although operation at the proposed uprated power level will increase the RPV surface fluence, the net effect in 1/4T fluence over two OCs up to approximately 23 EFPYs is bounded by the 48 EFPY value. Also, there will be little change in the ART up to 23 EFPYs. The staff concludes that, with the valve of the fluence bounded, the P-T curves approved for use though OC 16 remain valid for operation at the uprate conditions until 23 EFPYs is reached. Therefore, the staff finds the PNPS RPV P-T limits will continue to comply with the requirements of Appendix G through 23 EFPYs. As discussed in the staff's safety evaluation for LA 197, the licensee will need to develop new P-T curves for operations at the uprated power level in accordance with NRC approved methodology prior to operation beyond OC 16 (23 EFPYs).

3. In its application, the licensee stated that the PNPS RPV material surveillance program consists of three capsules that have been in the reactor vessel since plant startup. Under this program, the licencee removed the first capsule after approximately 4.17 EFPYs of operation, the second capsule is scheduled to be removed at 21 EFPYs, and the third capsule is scheduled to be removed at 32 EFPYs (EOL). The licensee concluded that the 1.5% power uprate has no effect on the existing surveillance capsule withdrawal schedule. The licensee also stated that PNPS is currently part of the BWRVIP Integrated Surveillance Program/Supplemental Surveillance Capsule Program (ISP/SSP). The NRC has not received a request for the BWRVIP ISP to be approved for PNPS. Although a small increase in neutron fluence will result from the 1.5% power uprate, the small increase will not significantly impact the RPV integrity such that a change in the surveillance program schedule is necessary. Therefore, the staff agrees that there will not be an impact on the capsule withdrawal schedule of the PNPS RPV material surveillance program. The staff also believes that if Pilgrim is approved for participation in the BWRVIP ISP in the future, based on the discussion above, the small increase in neutron fluence due to the 1.5% power uprate would not result in the need to

modify the BWRVIP ISP surveillance capsule withdrawal schedule. The staff concludes that the RPV surveillance program will continue to meet the requirements of 10 CFR Part 50, Appendix H at the proposed increased power level.

RPV Structural Evaluation

The licensee evaluated the RPV considering the changes in design input parameters and loads due to the proposed 1.5% power uprate. The licensee indicated that they evaluated the effect of the proposed uprate on the reactor vessel components, other than the FW nozzles, in accordance with the ASME Code, 1965 Edition with addenda up to, and including, Summer 1966, which is the Code of Record. In its RAI response dated November 6, 2002, the licensee indicated that they evaluated the FW nozzle that previously had a modification to its safe end using the 1980 edition of the ASME Code, which was the applicable code at the time of its modification. The proposed power uprate does not change the operating reactor pressure and temperature from the current operating condition. There are no changes in fuel lift and seismic loads due to the uprate. The current design basis transients remain valid for the proposed power uprate. The licensee evaluated the RPV for the design, the normal, upset, emergency, and faulted conditions. The licensee provided the calculated stresses and fatigue usage factors (CUFs) for the affected limiting reactor vessel components in Table 3-2 of Attachment 2 of the original amendment request, dated July 5, 2002. The staff reviewed the licensee's code design limits and the calculated stresses and CUFs and determined that the calculated stresses and CUFs are less than the ASME Code limits and, therefore, acceptable for the proposed 1.5% power uprate.

3.2.3 Reactor Internals

The licensee evaluated the reactor's internal components, considering the changes in the values of design input parameters and loads due to the proposed 1.5% power uprate. The loads applicable to internal components include reactor internal pressure difference, LOCA, SRV, seismic, annulus pressurization, jet reaction, and fuel lift loads.

The licensee evaluated the reactor internals by reviewing design loads due to pressure, temperature, weight, seismic, and flow. As a result of this evaluation, the licensee confirmed that the design loads were either bounded by the current design basis loads or remain unchanged and, therefore, concluded that the design basis analysis for the reactor internal components will remain valid for the proposed 1.5% power uprate. Based on the staff's review of the licensee's evaluation, the staff agrees with the licensee's conclusion that the effects of the proposed 1.5% power uprate on reactor internals will either be bounded by the design basis or remain unchanged for the components. The staff concludes, therefore, that the reactor internals will remain acceptable under the uprated conditions.

3.2.4 Flow-Induced Vibration

The licensee assessed flow-induced vibration for the proposed power uprate for limiting reactor internal components. The licensee indicated that there is a slight increase in flow-induced vibration for the shroud, shroud head and separator, steam dryer, and FW sparger because of approximately a 2% increase in steam and FW flow due to the power uprate. Other internal components are not affected since the maximum core flow and the maximum recirculation drive flow remain unchanged following the proposed 1.5% power uprate. As a result of their evaluation, the licensee concluded that vibration of safety-related internal components due to flow-induced vibration loads will remain within the GE acceptable stress limits of 10 ksi. The staff reviewed the licensee's vibration analysis and determined that the results are conservative, considering the GE acceptable limits of 10 ksi, in comparison to the ASME allowable limit of 13.6 ksi for service cycles equal to 10^{11}. The staff concludes, therefore, that the reactor internals design remains acceptable for flow induced vibration at the uprated conditions.

3.2.5 Piping Evaluation

a. RCPB Piping

The licensee evaluated the effects of the proposed 1.5% power uprate condition on the reactor coolant piping, components and their supports with regard to changes in flow rate, temperature and pressure. The licensee's application summarized its evaluation of RCPB piping inside containment. The piping systems evaluated included: the recirculation, MS and attached piping systems inside the containment; SRVDL piping; reactor core isolation cooling (RCIC) piping; MS drain lines; RPV head vent line; FW piping (inside containment); RPV bottom head drain line; RHR piping; core spray (CS) piping; high pressure coolant injection (HPCI) piping; RWCU; low pressure coolant injection (LPCI) piping; RWCU piping; and standby liquid control (SLC) system piping.

The licensee indicated that there are no changes in the reactor dome operating and design pressures and temperatures, nor are there any changes in the MS operating and design pressures and temperatures. There is a slight increase in the MS and FW flow rate and in the FW system operating pressure and temperature. The licensee reviewed the design basis calculation of the RCPB piping, and its support components and found that the proposed 1.5% power uprate does not have an impact on any piping except for a portion of the FW and MS piping, branch piping connected to FW and MS lines, and the SRVDL piping. The licensee evaluated the MS line and its attached piping systems, and determined that there are sufficient margins between the calculated stresses and the allowable limits to accommodate the slight increase (about 2%) in steam flow for the proposed power uprate condition. The licensee also indicated that the increased pressure, temperature and flow rate in the FW line and its attached piping systems, due to the proposed power uprate, are bounded by CLB conditions. Based on the evaluation as discussed above, the licensee concluded that, except for the SRVDL piping, the existing RCPB piping and supports will continue to perform their intended function for operation at the 1.5% uprate conditions.

Based on a review of the licensee's evaluation against the CLB, the staff finds the licensee's evaluation of the RCPB piping and supports, with the exception of SRVDL piping, is adequate in that current operating conditions are bounding with respect to the proposed power uprate conditions, and the piping is acceptable for plant operation at the proposed 1.5% power uprate condition. The evaluation of the SRVDL piping modifications for the power uprate is addressed later in this section.

In their RAI response dated November 6, 2002, the licensee stated that the original piping Code used for the RCPB at PNPS is the USAS B31.1.0, 1967 Edition. The licensee used the USAS B31.7, 1969 Edition for the MS line fatigue evaluation. The licensee later reanalyzed the MS lines in accordance with USAS B31.1, 1973 Edition, which does not specify a fatigue evaluation. The staff notes that when the recirculation piping was replaced, the licensee analyzed the piping using Section III of the ASME Code, 1980 Edition, through the winter 1981 Addenda. However, the licensee used the ASME Code, Section III, 1977 Edition, through Summer 1977 Addenda for the current SRVDL piping analysis.

The licensee indicated that the current SRV capacity is inadequate with the 2% MS flow increase necessary for the proposed power uprate. Therefore, SRVs will be modified to allow additional steam flow (about 7.5%) for the power uprate. As a result, the original SRVDL analytical margins would be exceeded by the increased loads due to the higher SRV flow. In response to the staff's RAI, the licensee indicated that they reanalyzed the modifications of the SRV piping system, the MS system, and SRVDL piping and supports for the power uprate to address the increased loads due to higher flow. The licensee performed the analysis according to the ASME Code, Section III, 1977 Edition, through Summer 1977 Addenda, Subsection NC-Class 2, which is the code of record, as stated above.

The licensee also indicated that the analysis used ISM/SRSS grouping methodology, in conjunction with the utilization of RG 1.61 damping in the piping/support seismic SSE analysis, combined with the increased SRV loads at the proposed power uprate condition. This design methodology is not within the CLB of the plant. In their response to the staff's RAI, the licensee provided a detailed discussion regarding the methodology used for the SRVDL seismic reanalysis. The staff's review of the ISM methodology including the damping used in the analysis, is provided in Section 3.2.1 of this SE.

Based on a review of the licensee's evaluation against the CLB, the staff determined that the licensee's evaluation of the SRV piping system, the MS, and SRVDL piping and supports is adequate for the proposed 1.5% power uprate. Therefore, the staff concludes that the RCPB piping is adequate to maintain integrity at the uprated conditions.

b. Non-RCPB Safety Class Piping Evaluation

The licensee evaluated the impact of the proposed 1.5% power uprate on large-bore and small-bore Safety Class, ISI Class 2 and 3 piping and supports. The licensee determined that the DBA LOCA loads are based on 102% of the CLTP, which bounds the proposed 1.5% power uprate. The licensee concluded that the proposed 1.5% power uprate has no impact on non-RCPB Safety Class piping.

Based on a review of the licensee's evaluation against the CLB, the staff determined that the licensee's evaluation of the piping is bounding for the proposed 1.5% power uprate. Therefore, the staff concludes that the piping is adequate for ensuring against piping failure at the uprated conditions.

3.2.5.1 Balance-of-Plant (BOP) Piping Evaluation

The licensee evaluated the BOP piping systems by comparing the original design basis conditions with those for the proposed power uprate. The licensee determined the BOP piping systems that are affected by reactor and BOP heat balances for the uprated conditions. The systems affected by the proposed power uprate are: MS line (outside containment); MS turbine bypass piping; MSIV drain piping; extraction steam piping; FW line outside containment to the inboard FW check valve, condensate and FW lines, condensate demineralizer (CD) (less than 1°F change), and FW heater drain.

The licensee reviewed the piping stress analyses-of-record. The input parameters (temperature and pressure) used in the design basis piping stress analyses remain bounding for the proposed power uprate. The licensee did not identify any new postulated pipe break locations in the systems evaluated. The licensee concluded that the BOP piping and related support systems remain within allowable stress limits in accordance with "Power Piping ASME B31.1-1989 Edition," which is the code of record specified in the Pilgrim UFSAR.

The staff reviewed the licensee's evaluation and determined that the CLB analysis is bounding for operation at the proposed 1.5% power uprate. Therefore, the staff concludes that the BOP systems will operate without adverse effects on the piping system and its supports at the proposed 1.5% power uprate conditions.

Flow Accelerated Corrosion (FAC) in Piping

In their application, the licensee stated that carbon steel MS piping, carbon steel FW piping, and carbon steel high-energy piping systems can be affected by FAC. FAC is influenced by changes in fluid velocity, temperature, and moisture content which will result from the proposed power uprate condition. PNPS has established a program for monitoring pipe wall thinning in single and two-phase high-energy carbon steel piping. The licensee evaluated changes resulting from the power uprate condition resulting in minor changes to the parameters affecting these systems. The licensee does not plan to change the scope and frequency of inspection of MS and FW piping, since their continuing inspection program takes into consideration adjustments to predict material loss rates used to project the need for maintenance/replacement prior to reaching minimum wall thickness requirements.

The staff determined that the licensee appropriately addressed the evaluation and inspection of FAC in the BOP systems through actions taken in response to Generic Letter (GL) 89-08, "Erosion/Corrosion in Piping." The PNPS FAC program currently monitors affected systems and ensures the integrity of susceptible high-energy piping systems. This program will include appropriate changes to the piping inspection frequency to ensure adequate margin for those systems with changing process conditions.

In their RAI response dated November 6, 2002, the licensee stated that they used a predictive code as part of the PNPS FAC program, "Flow Accelerated Corrosion (FAC) Version 1.0F (Build 52)." In addition, the licensee's RAI response provided a summary of the predicted change of wear-rates calculated by the revised predictive code for the power uprate condition.

The staff reviewed the licensee's application and supplemental information and concluded that the percent change in predicted wear-rate and the change in predicted wear-rates (mils/year) were negligible for the power uprate condition. The staff, therefore, concludes that the licensee's FAC program is acceptable for the uprated conditions.

3.2.6 Reactor Recirculation System

The reactor recirculation system evaluation described in TLTR Section 5.6.2 applies to the PNPS plant. PNPS is currently licensed to operate at up to a maximum core flow of 107.5% (74.2 Mlbm/hr), and the 1.5% power uprate will not increase the currently licensed maximum core flow. The licensee predicts that the power uprate may result in a small increase in the core flow pressure drop of less than 1 psi. This slight change in the core pressure drop is within the recirculation system capability.

The cavitation protection interlock will remain the same in relation to absolute power, since the interlock is based on the FW flow. The licensee identified that the ratio of core thermal power level to FW flow remains unchanged during operation at the rated thermal power (RTP); therefore, the cavitation interlock remains unchanged.

The staff reviewed the licensee's evaluation against the CLB and agrees that the changes associated with the 1.5% power uprate will not have a significant impact on the function of the PNPS recirculation system. The staff concludes, therefore, that reactor recirculation system operation is acceptable for power uprate conditions.

3.2.7 MS Line Flow Restrictors and MSIVs

The MSIVs are part of the RCPB and must be able to close within specific limits at all design and operating conditions upon receipt of a closure signal. The licensee stated that the closure times used in the design analyses for the MSIVs remain unchanged for the 1.5% power uprate, and that all safety and operational aspects of the MSIVs are within previous evaluations. Regarding the MS line flow restrictors, the licensee stated that the flow inputs used in the design analyses remain unchanged for the power uprate, because no change in steam break flow occurs (since the operating pressure is unchanged), and the safety and operational aspects of the flow restrictors are within previous evaluations.

Based on the NRC staff's review and the experience gained from the review of power uprate applications for similar BWR plants, the staff finds that plant's operation at the 1.5% power uprate condition will not have a significant impact on the ability of the MSIVs and MS line flow restrictors to meet their design objectives. The analysis of these components are, therefore, acceptable for power uprate conditions.

3.2.8 RCIC

The RCIC system provides core cooling when the RPV is isolated from the main condenser, and the RPV pressure is greater than the maximum allowable for starting a low-pressure core cooling system. The RCIC system is designed to provide rated flow over a range of reactor pressures from 150 psig to the maximum pressure corresponding to the lowest opening setpoint for the SRVs. In particular, the loss-of-feedwater (LOFW) flow transient assumes that the RCIC will maintain sufficient water level inside the core shroud to ensure that the top of the active fuel (TAF) will be covered throughout the event. The transient analysis also assumes that the low-setpoint SRVs would remove the stored and decay heat since MSIV closure on low water level isolates the reactor from the main condenser. The transient is a power-dependent transient and is more severe at a higher initial power since there is more stored energy and decayed heat to be dissipated and the water level drops faster.

Section 5.6.7 and Appendix E of the TLTR provide an assessment of the RCIC system capability for the uprated power operation. The RCIC capacity and the decay heat calculations are based on 102% of the CRTP, and remain unchanged. Therefore, the capability to maintain the water level above the TAF will also remain unchanged for the proposed 1.5% power uprate.

In their November 21, 2002, RAI response, the licensee stated that the original PNPS LOFW analysis was performed at 100% power and the generic evaluation provided in the TLTR was not applicable to PNPS. Based on generic LOFW analyses of BWR/3s, the licensee estimated the range of core power over which the RCIC system can fullfil its design basis function. Using the results of these generic analyses, the licensee estimated the effect of a 2% increase in thermal power will be small. The licensee stated that there is 4 feet of margin to the TAF for the PNPS LOFW analysis-of-record and a 2% thermal power increase will result in an approximately 4-inch decrease in the minimum level. The licensee concluded that, since there is sufficient margin to the TAF, the PNPS RCIC will continue to meet its design basis function at the proposed uprated conditions.

Based on its review of the licensee's evaluation, and considering the planned increase in the SRV capacity at PNPS against the CLB, the staff determined that the SRVs are capable of removing stored and decay heat and that there is sufficient margin to the TAF to maintain core cooling capability at the increased power level. Accordingly, the staff concludes that the RCIC system performance will be acceptable at the proposed 1.5% power uprate conditions.

3.2.9 RHR System

The generic discussions provided in TLTR Sections 5.6.4 and J.2.3.13, are applicable to the PNPS plant. The RHR is designed to restore and maintain the coolant inventory in the reactor vessel, and to provide primary system decay heat removal after reactor shutdown, for both normal and post-accident conditions. The RHR system is designed to operate in the LPCI mode, the shutdown cooling mode, the suppression pool cooling mode, and the containment spray cooling mode.

Shutdown Cooling (SDC) Mode

The objective of normal shutdown, as stated in the PNPS UFSAR, is to reduce the bulk reactor temperature after scram to 125°F in approximately 20 hours, using two SDC heat-exchanger loops. RG 1.139, "Guidance for Residual Heat Removal," provides an alternative approach to demonstrate SDC capability: The RHR system can reduce the reactor coolant temperature to 200°F within 36 hours.

The licensee stated that the evaluation in TLTR Sections 5.6.4 and J.2.3.13 of Appendix J is applicable to PNPS, and the slightly higher decay heat resulting from operation at the proposed 1.5% uprate power level has a negligible effect on the SDC mode of the RHR.

The staff reviewed the licensee's evaluation and determined that the bases for the generic evaluations in the TLTR are applicable to PNPS. The staff, therefore, concludes that the performance of the SDC mode of the RHR system will be acceptable at the proposed uprated operating conditions.

3.2.10 RWCU

The primary parameters that affect the RWCU system are power transients, RWCU operating temperature and pressure, and recirculation flow temperature and system impurities, such as fission and corrosion products. Power transients are the primary challenge to the RWCU system and are independent of the power uprate. The licensee stated that there is no significant effect on operating temperature and pressure conditions in the high-pressure portion of the system.

The staff reviewed the licensee's evaluation and determined that the proposed power uprate is acceptable because the RWCU system performance is not significantly affected at the uprate conditions.

3.2.11 Reactor Coolant System and Connected Systems - Conclusion

Based on the evaluations in Sections 3.2.1 through 3.2.10 of this SE, the staff concludes that the proposed power uprate is acceptable with respect to its impact on the reactor coolant system and connected systems.

3.3 Engineered Safety Features

3.3.1 Containment Systems Performance

The licensee stated that, with the exception of the subcompartment pressurization analysis, the current containment evaluations are bounding for the 1.5% power uprate, because they were performed at 102% of the CLTP level. Although the nominal operating conditions increase slightly at the uprated power level, the initial conditions for containment analysis inputs remain the same. The licensee evaluated the containment short-term pressure and temperature response, the long-term temperature response of the suppression pool, the containment dynamic loads, and containment isolation. Additionally, the licensee reviewed the annulus subcompartment pressurization. The licensee's evaluation found that the increase in annulus subcompartment pressurization, due to the 1.5% power uprate, was calculated to be less than 0.1%, which is considered negligible.

Based on the staff's review of the licensee's application and experience gained from the review of power uprate applications for similar BWR plants, the staff agrees with the licensees's assessment that the proposed 1.5% power uprate will have negligible effects on containment system performance. The staff concludes, therefore, that containment performance will remain acceptable under the uprated conditions.

Motor Operated Valves

The licensee assessed its GL 89-10, "Consideration of the Results of NRC-Sponsored Tests of Motor-Operated Valves," program related to motor-operated-valves (MOVs). The licensee's evaluation indicated that they had performed the existing MOV evaluation at PNPS based on the reactor SRV pressure settings that are bounding for the power uprate condition. The licensee evaluated its commitments relating to GL 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves," associated with the pressure locking and thermal binding of safety-related power operated gate valves that are required to perform an intended safety function. The licensee found that the existing analysis conditions remain bounding for the 1.5% power uprate. The licensee also evaluated its response relating to the GL 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions," program regarding the over-pressurization of isolated piping segments.

The licensee concluded that the existing evaluation for GL 96-06 was performed at the containment design temperature and pressure and is, therefore, bounding for the proposed power uprate.

The staff reviewed the licensee's use of approved NRC programs and determined that the existing evaluation of safety-related valves remain bounding for the 1.5% power uprate. Therefore, the power uprate will have no adverse effects on safety-related valves. The staff concludes, therefore, that safety-related valve operation will remain acceptable under the uprated conditions.

3.3.2 ECCS

The ECCS is designed to provide protection in the event of a LOCA resulting from a rupture of the primary system piping. Although such DBAs are not expected to occur during the lifetime of a plant, plants are designed and analyzed to ensure that the radiological dose from a DBA will not exceed the 10 CFR Part 100 limits. For a LOCA, 10 CFR 50.46 specifies design acceptance criteria based on: (1) the peak cladding temperature; (2) local cladding oxidation; (3) total hydrogen generation; (4) coolable core geometry; and (5) long-term cooling. The LOCA analysis considers a spectrum of break sizes and locations, including a rapid circumferential rupture of the largest recirculation system pipe. Assuming a single failure of the safety equipment needed to mitigate a LOCA, the LOCA analyses identify the break sizes that most severely challenge the ECCS systems and the primary containment. The MAPLHGR operating limit is based on the most limiting LOCA analysis, and the licensee performs LOCA analyses for each new fuel type to demonstrate that the 10 CFR 50.46 and Appendix K acceptance criteria can be met.

The ECCS for PNPS consist of the HPCI system, the LPCI mode of the RHR system, the CS system, and the ADS.

HPCI

The HPCI system is a turbine-driven system designed to pump water into the reactor vessel through FW system piping over a wide range of operating pressures. The HPCI system, with other ECCS systems as backup, is designed to maintain reactor water level inventory during small- and intermediate-break LOCAs and isolation transients. The primary purpose of the HPCI system is to maintain reactor vessel coolant inventory in the event of a small break LOCA that does not immediately depressurize the reactor vessel. The HPCI system also serves as a backup to the RCIC system to provide makeup water in the event of a LOFW transient.

The licensee stated that the generic evaluation in Section 5.6.7 of the TLTR is applicable to PNPS. The licensee confirmed that the capability of the PNPS HPCI system to perform its safety function at the 1.5% power uprate is demonstrated based on previous ECCS-LOCA analysis performed at 102% of the CLTP.

Based on a review of the licensee's application against the CLB, the staff has determined that the operating conditions for the HPCI system at the uprated power level are bounded by those used in the current DBA. Accordingly, the staff concludes that the HPCI system will remain capable of performing its intended function at the 1.5% power uprate condition.

CS

The primary purpose of the CS system is to provide reactor vessel coolant inventory makeup during a large break LOCA and for any small break LOCA after the reactor vessel has depressurized. The CS system also provides spray cooling for long-term core cooling in the event of a LOCA.

The licensee states the generic evaluation in Section 5.6.10 of the TLTR is applicable to PNPS. In addition, the adequacy of the CS system is demonstrated by ECCS-LOCA analyses performed at 102% of the CLTP.

Based on a review of the licensee's application against the CLB, the staff has determined that the operating conditions for the CS system at the uprated power level are bounded by those used in the current DBA. Accordingly, the staff concludes that the CS system will remain capable of performing its intended function at the 1.5% power uprate condition.

LPCI

The LPCI mode of the RHR system is automatically initiated in the event of a LOCA. The primary purpose of the LPCI mode is to provide reactor vessel coolant inventory makeup during a large break LOCA or any small break LOCA after the reactor vessel has depressurized. In conjunction with other ECCS systems, the LPCI mode is used to provide adequate core cooling for all LOCA events. The ECCS-LOCA analysis, performed at 102% of the CLTP, demonstrates the adequacy of the system for providing core cooling.

Based on a review of the licensee's application against the CLB, the staff has determined that the operating conditions for the LPCI system at the uprated power level are bounded by those used in the current DBA. Accordingly, the staff concludes that the LPCI system will remain capable of performing its intended function at the 1.5% power uprate condition.

ADS

The ADS uses the relief valves or SRVs to reduce reactor pressure after a small-break LOCA with HPCI failure, allowing the LPCI and the CS to provide cooling flow to the vessel. The licensee states that the generic evaluations in Section 5.6.8 of the TLTR are applicable to PNPS. The ability of the ADS to perform its required safety function is demonstrated by the ECCS-LOCA analysis performed at 102% of the CLTP.

However, for the proposed power uprate, the existing SRVs do not have sufficient capacity to meet the additional overpressure protection requirements. Therefore, the licensee stated they will modify the SRVs by increasing the capacity by 7.5%, and the additional SRV capacity provides adequate overpressure protection for the 1.5% power uprate. The licensee concluded that all of the safety aspects of the ADS are within the previous evaluation (with the modification to the SRVs), and the pressure limits are unchanged for the uprated power conditions. The overpressure protection capability of the SRVs during an ATWS is discussed in Section 3.8.3 of this SE.

Based on a review of the licensee's application against the CLB with the modification to the SRVs, the staff has determined that the operating conditions for the ADS at the uprated power level are bounded by those in the current DBA. Accordingly, the staff concludes that the ADS will remain capable of performing its intended function at the 1.5% power uprate condition.

ECCS Net Positive Suction Head (NPSH)

The licensee stated that the most limiting case for NPSH typically occurs at the peak long-term suppression pool temperature. In addition, the licensee states that the generic evaluation of the containment provided in Appendix G to the TLTR is applicable to PNPS. Since the containment analysis was performed at 102% of the CLTP, the licensee concludes that there is no change in the available NPSH for systems using suppression pool water and the power uprate has no effects on the NPSH requirements.

Based on a review of the licensee's application against the CLB, the staff has determined that the NPSH for systems using suppression pool water remains unchanged at the 1.5% power uprate condition. Accordingly, the staff concludes that the current NPSH analyses remain valid for the proposed power uprate conditions.

3.3.3 ECCS Performance

The ECCS is designed to provide protection against postulated LOCAs caused by ruptures in the primary system piping. ECCS performance under all LOCA conditions and the analysis models must satisfy the requirements of 10 CFR 50.46 and Appendix K to 10 CFR Part 50.

ECCS-LOCA Codes and Methodology

Section 5.3.1 and Appendix D of the TLTR address the ECCS-LOCA performance evaluation for operation at the proposed 1.5% power uprate level. For the GE Nuclear Energy (GENE) analytical methods, the Appendix K ECCS-LOCA analyses are performed at 102% of the CRTP. Therefore, the licensing basis ECCS-LOCA calculations, derived from the Appendix K ECCS-LOCA methodology, bound operation at the power uprate condition.

However, GENE's nominal/upper-bound SAFER/GESTR-LOCA analysis is performed at the RTP and core flow conditions, which would not bound the uprated power conditions. The TLTR established a margin criteria for upper-bound PCT. Since the PNPS upper-bound PCT did not meet the TLTR ECCS-LOCA margin criteria, the licensee performed a plant-specific analysis at the proposed uprate power level. The result of the ECCS-LOCA analysis at the 1.5% power uprate condition indicated that the upper-bound PCT would increase by less than 1°F.

In their RAI response, dated November 21, 2002, the licensee stated that in order to meet the original standards of the SAFER/GESTR-LOCA analysis method and maintain the upper-bound PCT less than or equal to 1600°F, the permissible MAPLHGR for both GE 14 and GE 11 is restricted to a level that is below the design power limit of the fuel. The MAPLHGR is limited for both fuel types in order to meet the upper-bound PCT limit at the CLTP and at the 1.5% power uprate level.

The 1600°F upper-bound PCT limit is specific to the SAFER/GESTR-LOCA method. This method[1] has been revised, removing the necessity to demonstrate that plants meet the upper-bound PCT limit. Without the 1600°F PCT limit requirements, the SAFER-GESTR/LOCA method acceptance criteria is based solely on 10 CFR 50.46 requirements. The licensee stated that in the next OC, the PNPS MAPLHGR limits will be established based on the 10 CFR Part 50.46 requirements, with no consideration of the previous upper-bound PCT.

Based on the above discussion, and the licensee's plant-specific evaluations performed in accordance with NRC-approved methodology, the staff concludes that the PNPS LOCA analysis will meet the requirements of 10 CFR 50.46 and Appendix K at the uprated conditions.

3.3.4 Main Control Room (MCR) Environmental Control System

The licensee evaluated the current design basis analyses for accident dose accumulation to the MCR operators. An increase in rated reactor power of 1.5% would increase the estimated dose to the MCR occupants by approximately 1.5%. The licensee's evaluation determined that the increased calculated dose would remain below the MCR habitability limits of 10 CFR Part 50, Appendix A, GDC 19.

Based on the staff's review of the licensee's evaluation and the experience gained from the review of power uprate applications for similar BWR plants, the staff finds that the licensee's existing analysis for the MCR Environmental Control System meets applicable regulatory guidelines, and the impact on MCR operators is negligible at the proposed uprated condition. The staff concludes, therefore, that the MCR environmental control system remains acceptable for the uprated conditions.

3.3.5 Standby Gas Treatment System (SGTS)

The SGTS minimizes the offsite and control room doses during venting and purging of the containment atmosphere under abnormal conditions. The licensee designed this system to maintain the secondary containment at a slightly negative pressure under such conditions. The licensee stated that charcoal beds in the SGTS can accommodate DBA conditions at 102% of the CLTP, which bounds the 1.5% power uprate.

Based on the staff's review of the licensee's application, the staff confirmed that the current analysis remains bounding and concludes, therefore, that the SGTS operation remains acceptable for the uprated conditions.

3.3.6 Post LOCA Combustible Gas Control System (CGCS)

The CGCS maintains the post-LOCA concentration of oxygen or hydrogen in the containment atmosphere below the flammability limit. The licensee stated that the existing evaluation of the CGCS was performed for accident conditions at 102% of the CLTP level. The metal available

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[1]GESTR-LOCA and SAFER Models for Evaluation of Loss-of-Coolant Accident, Volume III, Supplement 1-Additional Information for the Upper-bound PCT Calculation, NEDC-23786P-A, Supplement 1, Revision 1, March 2002

for reaction is unchanged by the 1.5% power uprate and hydrogen production due to radiolytic decomposition is bounded by the previous evaluation.

Based on the staff's review of the licensee's application, the staff confirmed that the current analysis remains bounding and concludes, therefore, that the CGCS operation remains acceptable for the uprated conditions.

3.3.7 Engineered Safety Features - Conclusions

Based on the evaluation in Sections 3.3.1 through 3.3.7, the staff concludes that the proposed power uprate is acceptable with respect to its impact on the PNPS Engineered Safety Features.

3.4 Instrumentation and Control

3.4.1 Instrumentation and Control - Evaluation

The licensee will install a Crossflow UFM system for FW flow measurement at the PNPS to support the power uprate. The Crossflow UFM flow measurement system consists of four transducers, a signal conditioning unit, and a data processing computer. The transducers are mounted on the exterior of the FW piping. There is one upstream and one downstream transducer station, each having one transmitter and one receiver. Fluid flow affects the ultrasonic beam transmitted across the pipe in a manner dependent upon details of the turbulence patterns within the beam. The turbulence pattern across any diameter, at any time, is essentially random, but it persists in a recognizable form for a short time as the fluid moves downstream. Therefore, it is possible to "sample" the turbulence pattern at one point in the pipe, and then monitor for it to appear at some fixed point downstream. The time required for the pattern to move that fixed distance gives the flow velocity, and that, along with flow profile and pipe interior geometrical considerations, yields an accurate indication of the volumetric flow rate. This increased flow measurement accuracy (as compared to a venturi-based flowmeter) improves the accuracy of calculated core thermal power.

The ABB Combustion Engineering Nuclear Power, Inc., (ABB-CE) topical report, CENP-397-P, Revision-01-P, "Improved Flow Measurement Accuracy Using Crossflow Ultrasonic Flow Measurement Technology," documents the theory, design, and operating features of the Crossflow UFM and its ability to achieve increased accuracy of flow measurements compared to the existing venturi-based flow measurements. The staff reviewed the ABB-CE topical report and issued a safety evaluation report (SER) dated March 20, 2000, which approved use of the report for demonstrating the accuracy of the Crossflow UFM.

Plant operation at the uprated power level will alter many operating parameters, and, therefore, could affect the severity of DBAs and transients. The licensee has evaluated the impact of the proposed power uprate on the plant systems and components. The evaluations were performed on a case-by-case basis for all applicable instrumentation and control systems and components of: (1) the NSSS, (2) the BOP systems, and (3) the instrument systems credited in the accident analyses. The evaluations and their results are summarized in the licensee's submittal. The licensee stated that results of evaluations demonstrate that, with very few minor design modifications, all regulatory acceptance criteria will continue to be met at the proposed uprated power. The instrumentation and control parameters that could be changed as a result of the proposed 1.5% power uprate are: neutron flux; turbine inlet pressure; steam flow; FW flow; and temperature. The licensee stated that they have evaluated the suitability of the instrumentation and controls systems, components, instrument signal ranges, and setpoints to assess the impact of the proposed power uprate. The evaluation indicated that re-calibration of the APRMs will be required to indicate 100% of the proposed uprated power level. However, the APRM high flux reactor scram, and the upper limit of the rod block setpoints expressed in units of percent of licensed power will not be changed. No adjustment will be needed to ensure that the Intermediate Range Monitors (IRMs) have adequate overlap with the Source Range Monitors and APRMs. At the uprated power level, the flux at some Local Power Range monitors will increase. The rod block monitor instrumentation is referenced to an APRM channel, but because the APRMs will be rescaled for the proposed power uprate, the impact of the higher average local flux on APRM performance will not be significant.

Based on its review of the licensee's evaluation of instrumentation and controls the staff determined that the appropriate instrumentation and control parameters that could be impacted by the proposed power uprate have been identified and that only minor modifications are necessary. Accordingly, the staff concludes that plant instrumentation and control systems will continue to operate acceptably during the uprated conditions.

3.4.2 TS Instrument Setpoints

The licensee stated that the proposed power uprate will increase steam flow by approximately 2% and FW temperature by $<2^0$F. The licensee has evaluated the associated setpoints with respect to higher steam flow and increased FW temperature. The licensee concluded that the proposed power uprate has no significant effect on current settings of temperature-based and non-temperature-based leak detection functions. The setpoint evaluation indicated that current analytical limits, allowable values (AVs) and nominal setpoints for SRV settings, MS line high flow isolation, fixed APRM scram, flow-biased APRM scram, rod worth minimizer low power, rod block monitor, low steam line pressure MSIV closure (during reactor "Run" mode), reactor water level monitoring, and MS line tunnel high temperature isolation functions will be acceptable for the proposed power uprate.

The licensee stated that, as a part of the turbine replacement project, they will perform a setpoint evaluation for the turbine first-stage pressure function that provides input to the turbine generator trip reactor scram function. The licensee stated that they used standard GE setpoint methodology for instrument scaling calculations, AVs, and nominal trip setpoint calculations. The licensee is establishing each trip-setting to preclude inadvertent initiation of the protective action(s) while assuring adequate allowances have been provided for instrument accuracy, calibration, drift, and effects of environmental conditions applicable to normal and accident design basis events. The licensee found changes in the setpoint margins due to changes in instrument accuracy, and calibration errors caused by the changes in environmental conditions around the instrument due to the proposed power uprate, to be negligible.

Based on the staff's review of the licensee's evaluation, the staff determined that the instrument setpoints for temperature-based and non-temperature-based leak detection functions will not be significantly impacted by plant operation at the proposed power uprate. Additionally, the staff determined that the changes in the environmental conditions around instrumentation as a result of operation at the increased power level will not significantly impact instrumentation setpoint margins. Accordingly, the staff concludes that the setpoints remain acceptable at the uprated conditions.

3.4.3 Crossflow UFM Evaluation

The licensee stated that the proposed operation at a power increase of 1.5% is based on the fact that the sum of increased core power level (1.5%) and the improved power measurement uncertainty ($\leq$ 0.5%) is within the previously analyzed conditions (where the plant analysis used a 2% margin above the CLTP to account for power measurement uncertainty).

The licensee stated that, although use of the Crossflow UFM system for monitoring FW flow is not a safety-related function, the system is being designed and manufactured under the vendor's standard quality control program. This program addresses configuration control, maintenance, deficiency reporting, and deficiency correction procedures. The licensee further stated that the Crossflow UFM system software will be controlled in accordance with the Entergy Software Quality Assurance (QA) program for Level B software, and that the plant procedures necessary for system maintenance, instrument calibration, control of software and hardware system configuration, identifying, reporting, and correcting system deficiencies, and for addressing manufacturer-identified deficiencies will be in place prior to operation at the uprated power level. The licensee will conduct training for System Engineering, Operations, and Maintenance personnel as part of implementation of the Crossflow UFM system.

In their supplement dated April 14, 2003, the licensee provided a pre-installation thermal power measurement uncertainty calculation for PNPS that demonstrates that the power measurement uncertainty has a 95% probability of not exceeding 0.42%. This calculation is based on achieving an installed UFM uncertainty of 0.35%, or less. An additional unassigned safety margin of 0.08% is included in the total power measurement uncertainty. Therefore, the existing 2% margin can be allocated such that 1.5% is applied to obtain the uprate to 2028 MWt, and the remaining 0.5% is retained to account for the total power measurement uncertainty.

Since the UFM uncertainty is affected by the details of the installation, the actual UFM uncertainty cannot be predicted accurately until the instrumentation is installed and commissioned. Based on the discussion of the thermal power measurement uncertainty calculation presented above, the 1.5% power uprate is supported by the ability of the installed Crossflow UFM system to achieve an uncertainty of 0.35%, or less. In addition, the calculation also includes assumptions that could significantly impact the basis for determining the total power measurement uncertainty in support of the proposed 1.5% power uprate.

In their application, the licensee committed to install the Crossflow UFM system in accordance with the vendor's recommendations and to finalize the uncertainty calculation, based on installed operational data, in support of the 1.5% power uprate, prior to operating above the CLTP level of 1998 MWt.

Regarding the use of the Crossflow UFM to support the proposed power uprate, the staff reviewed the licensee's submittal to verify that:

1. the proposed power uprate is based on guidance provided in the ABB-CE topical report, the staff's SER approving the topical report, and the guidance provided in

   Regulatory Information Summary (RIS) 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications" (RIS-2002-03);

2. the licensee addressed the four criteria described in the staff's SE for the topical report; and

3. the licensee evaluated the impact of increased power operation on the applicable instrumentation and control systems and parameters.

The staff's SER for the ABB-CE topical report provided additional criteria that licensees needed to address when referencing the topical report in support of their requests for license amendment. In their submittal, the licensee addressed each of those criteria as follows:

(1) *The licensee should discuss the development of maintenance and calibration procedures that will be implemented with the Crossflow UFM installation. These procedures should include process and contingencies for an inoperable Crossflow UFM and the effect on thermal power measurement and plant operation.*

In their submittal the licensee stated that before they exceed the CLTP, the required plant procedures for the Crossflow UFM instruments and all other instruments that affect the power calorimetric will be in place for: (a) maintaining calibration; (b) controlling software and hardware configuration; (c) performing corrective actions; (d) reporting deficiencies to the manufacturer; and (e) receiving and addressing manufacturer deficiency reports. In addition, the procedures will be in place: (1) to verify instrument operability; (2) to address inoperable or OOS instrumentation conditions; and (3) to ensure that periodic in-service inspections will be performed to verify that the instrument uncertainty is never greater than the uncertainty value(s) used in the analyses to justify the proposed power uprate.

The licensee stated that they will maintain system accuracy and reliability through regular signal accuracy verification and diagnostic testing. The licensee will perform regular calibration and maintenance of the Crossflow UFM system using site procedures that will be developed using the vendor's instructions and in-house technical manuals. The licensee stated that the computer input modules include sufficient self-checking to obviate the need for periodic calibration, and that the on-board references used in the self-checking are configured and applied in such a manner as not to require verification. Therefore, the accuracy of the calculation provides assurance that the computer input module uncertainty will not exceed the value assumed in the development of the parameter uncertainty data.

The licensee stated that they will perform all work for installation and maintenance in accordance with site work-control procedures. Verification of system operation will be provided by the system software, and the licensee will control software and its configuration in accordance with the Entergy Software QA program for Level B software. The licensee stated that no failure of the Crossflow UFM equipment will adversely impact the FW system, because the Crossflow UFM system will be externally mounted on the FW piping.

The licensee is installing redundant Crossflow UFM systems to increase monitoring reliability. There will be online detection of non-conservative readings which could be a result of either rapid signal de-fouling and/or component failure. Alarms are provided for signal abnormalities and/or for the loss-of-signal conditions. In the event one of the two redundant units fails, there is an alarm function that will prompt the operator to switch control to the other operating UFM. This switchover will be controlled by procedure, which will also allow the operator to take one UFM OOS for maintenance. With one unit OOS, and the other in operation, there will be no change in plant operation.

The licensee will periodically compare the FW flow rate measured by the existing venturi-based channels with the UFM-based readings. A "Correction Factor" (CF) will be computed and updated at each comparison to bring the venturi-based readings into conformance with the UFM-based readings. All control and analysis functions presently based upon the venturi-based flowmeters, will continue to be based upon the current instruments, but will benefit from the added accuracy of the UFMs. In the unlikely event that both redundant Crossflow UFMs are OOS at the same time, the plant would continue to operate with the last good CF until one UFM is restored to operation, or until the end of the allowed OOS (AOT) of 24 hours, whichever occurs first. Once the AOT is ended, with both UFMs still unavailable, the power level will be limited to the pre-uprate value. If, during the AOT, some event results in the CF being deemed unreliable, then the licensee stated they will promptly reduce the power level to the pre-uprate limit.

The staff reviewed the licensee's response and concluded that it is acceptable because the licensee adequately addressed the criteria in the topical report and the staff's SER related to the maintenance and calibration procedures. The licensee indicated that new procedures will be developed and existing procedures revised, as appropriate, for the Crossflow UFM. In addition, the staff finds the licensee identified appropriate procedures that will address contingencies for an inoperable Crossflow UFM and the effect on thermal power measurement and plant operation.

(2) *For plants that currently have the Crossflow UFM installed, the licensee should provide an evaluation of the operational and maintenance history of the installed UFM and confirm that the instrumentation is representative of the Crossflow UFM, and is bounded by the requirements set forth in Topical Report CENPD-397-P.*

In their submittal, the licensee stated that Pilgrim does not have an existing Crossflow UFM installed. Therefore, this criterion is not applicable to Pilgrim.

(3) *The licensee should confirm that the methodology used to calculate the uncertainty of the Crossflow UFM in comparison to the current FW flow instrumentation is based on accepted plant setpoint methodology (with regard to the development of instrument uncertainty). If an alternative methodology is used, the application should be justified and applied to both the venturi and the Crossflow UFM for comparison.*

The licensee stated that they will perform calculations for FW flow measurement uncertainty after installation data is available. The licensee prepared a preliminary thermal power uncertainty calculation using an assumed value for FW flow uncertainty that is expected to envelope the actual value to be computed after installation. The licensee committed in their July 5, 2002, application, to determine the actual FW flow uncertainty after installation of the Crossflow UFM and to perform the total power uncertainty calculation to verify assumed parameters before increasing power above the CLTP limit. The licensee also stated that they will have operating procedures for the Crossflow UFM in place to verify and ensure that all parameters in the uncertainty calculation remain valid.

In addition, the licensee performed evaluations to assess the suitability of existing instruments, instrumentation setpoints, AVs, and signal ranges for the affected parameters, including: neutron flux; turbine inlet-pressure; steam flow; steam-dome pressure; FW flow and temperature; CRD flow and temperature; RWCU system flow and temperature; recirculation pump power and efficiency; system thermal losses; correction factor tolerance; and operation variances. The licensee stated that they performed these evaluations using the staff-approved GE setpoint calculation methodology with the uncertainty at a 95% probability level.

The staff has reviewed the thermal power uncertainty calculation, and finds the calculation methodology is consistent with the usual mathematical methods used in the industry for addressing uncertainty and are reasonable. In addition, the licensee performed its setpoint evaluation in accordance with a staff-approved setpoint methodology and is, therefore, acceptable.

(4) *The licensee of a plant at which the installed Crossflow UFM was not calibrated to a site-specific piping configuration (flow profile and meter factors not representative of the plant-specific installation) should submit additional justification. This justification should show that the meter installation is either independent of the plant-specific flow profile for the stated accuracy, or that the installation can be shown to be equivalent to known calibration and plant configurations for the specific installation, including the propagation of flow profile effects at higher Reynolds numbers. Additionally, for previously installed and calibrated Crossflow UFM, the licensee should confirm that the plant-specific installation follows the guidelines in the Crossflow UFM topical report.*

In their submittal, the licensee stated that the UFMs will be calibrated in accordance with the plant-specific piping configuration. The vendor will supervise the installation of the equipment to ensure the installation guidelines of the ABB-CE topical report are followed. The staff finds the licensee’s response consistent with the implementation procedures provided in the topical report and approved by the staff’s SER and, therefore, acceptable.

The staff finds that the licensee's response to these criteria appropriately addresses the plant-specific implementation of Crossflow UFM maintenance and calibration, hydraulic configuration, processes, and contingencies for an inoperable Crossflow UFM.

3.4.4 Instrumentation and Controls - Conclusion

Based on the above evaluation, the staff concludes that the licensee will implement the proposed changes in a manner that is consistent with the guidelines provided in the ABB-CE topical report, the criteria provided in the staff's SER for the topical report, and the guidance provided in RIS 2002- 03.

The licensee has addressed the criteria of the staff's SER for the ABB-CE topical report regarding the installation and implementation of the Crossflow UFM. In addition, the licensee has provided the necessary information in accordance with the guidance in RIS 2002-03. Based on the evaluations presented in Sections 3.4.1 through 3.4.3, the staff finds the proposed change acceptable in regard to the impact of the proposed power uprate on instrumentation and controls.

3.5 Electrical Power and Auxiliary Systems

3.5.1 Alternating Current (AC)/Off-site Power

In their application dated July 5, 2002, as supplemented September 27 and December 30, 2002, the licensee provided an evaluation of the 1.5% power uprate on off-site power sources. The licensee stated that the main generator is rated at 780 MVA (699 MWe at a 0.896 power factor). The main generator provides power through the isolated phase bus at 24 kV to both the main transformer and the unit auxiliary transformer. The generator voltage is stepped-up through the main transformer to a 345 kV ring bus. The preferred AC power source provides offsite AC power to the auxiliary power distribution system for the startup, operation, or shutdown of the station. The preferred AC power also provides a source of off-site AC power to all emergency loads necessary for the safe shutdown of the reactor. The electrical distribution system has been previously evaluated to conform to GDC 17. Also, the plant has been previously evaluated for environmental qualification for electrical equipment and station blackout considerations (10 CFR 50.49, and 10 CFR 50.63).

Grid Stability

The licensee completed a grid stability study in November 2002, that was approved by the Independent System Operator New England on December 10, 2002. The current operating point of the main generator is 699 MWe at a 0.896 power factor. At the 1.5% power uprate, the main generator will operate at 734 MWe at a 0.941 power factor. The generator capability curve contained in the study shows that operation at the power uprate condition is possible at a power factor of 0.941. The main generator can operate within the range of 280 megavolt-ampere reactive (MVAR) lagging and 100 MVAR leading. The study identified a minor grid stability issue necessitating the replacement of a lockout relay for the West Walpole substation 345 kV circuit breaker. The relay replacement is scheduled prior to operation at the uprated power level. The study identified no other grid stability issues for the 1.5% measurement uncertainty uprate for the unit.

The staff reviewed the licensee's submittal and determined that the impact of the power uprate on the grid stability is insignificant. The staff concludes, therefore, that grid stability remains acceptable at the uprated power level.

Main Generator

The licensee stated that the current operating point of the main generator is 699 MWe at a 0.896 power factor. At the proposed 1.5% power uprate, the main generator will operate at 734 MWe at a 0.941 power factor. A generator capability curve shows that operation at the uprated power condition is possible at a power factor of 0.941. The proposed power uprate of 1.5% does not affect the generator auxiliaries since the generator will continue to operate within its design rating.

Therefore, main generator performance is bounded by the existing design limits and is not impacted by the proposed power uprate.

The staff reviewed the main generator capability curve and determined that the main generator will continue to perform its intended function at the proposed power uprate and, therefore, is acceptable for operation at the uprated power level.

Main Power Transformer

The licensee stated that the main power transformer rating is 880 MVA. The maximum load on the main power transformer does not change and remains at 780 MVA, if operating at 100% power with the station electrical loads being supplied from the startup transformer.

The staff reviewed the licensee's submittal and determined that the load at the proposed power uprate of 1.5% is below the main transformer's maximum design rating of 880 MVA and, therefore, operation at the uprated condition is acceptable with regard to the main power transformer.

Isophase Bus

The licensee stated that the isophase bus rating is 20 kA for the main section and 850 A for the branch section. The proposed power uprate does not change the current in the main section as it remains at 18.764 kA for 780 MVA at a rated 24,000 volts. The main generator vendor manual states that the generator may operate at ± 5% voltage. Therefore, the maximum isophase bus current due to low voltage is 19.752 kA for 780 MVA at 22,800 volts. The branch section current would increase from 686 A to 698 A at 24,000 volts (721 A to 734 A at 22,800 volts) due to an increase in non-Class 1E loads on the unit auxiliary transformer.

The staff reviewed the licensee's submittal and determined that the impact of the proposed 1.5% power uprate on the isophase bus is below the design rating and, therefore, operation at the proposed uprated condition is acceptable with regard to the isophase bus.

Startup Transformer

The licensee stated that the startup transformer rating is 37.3 MVA. The load on the startup transformer increases from 28.5 MVA to 29 MVA at 100% power. The increase in load due to power uprate is bounded by its design rating of 37.3 MVA.

The staff reviewed the licensee's submittal and determined that the startup transformer loading resulting from the 1.5% power uprate is below its maximum design rating and, therefore, operation at the proposed uprated condition is acceptable with regard to the startup transformer.

Unit Auxiliary Transformer

The licensee stated that the maximum rating of the unit auxiliary transformer is 37.3 MVA with an increased load from 28.5 MVA to 29 MVA at 100% power. The unit auxiliary transformer loading at the proposed power uprate is bounded by the design load and is not impacted by the power uprate.

The staff reviewed the licensee's submittal and determined that the unit auxiliary transformer loading, resulting from the 1.5% power uprate, is below its maximum design rating and, therefore, operation at the proposed uprated condition is acceptable with regard to the unit auxiliary transformer.

The staff has reviewed the licensee's submittal for the effect of the proposed power uprate on the off-site power system and, for the reasons set forth above, concludes that the off-site power system will continue to meet the requirements of GDC 17 following implementation of the proposed power uprate. In short, adequate physical and electrical separation exists, and the off-site power system has the capacity and capability to supply power to all safety loads and other required equipment. The staff further concludes that the impact of the proposed power uprate on grid stability is insignificant. Therefore, the staff finds the proposed power uprate acceptable with respect to the off-site power system.

3.5.2   AC/On-site Power

The emergency diesel generators (EDGs) supply a source of AC power following a loss of off-site power or under off-site power degraded voltage conditions. The EDGs automatically supply AC power to the Class 1E buses in order to provide power to equipment required for a safe shutdown of the plant. The licensee stated that station loads under accident conditions are based on equipment nameplate data, except for the ECCS pumps where a conservatively high flow brake horse power (BHP) is used. Operation under accident conditions at the proposed uprated power level is achieved by utilizing existing equipment operating at, or below, the nameplate rating and within the calculated BHP for the ECCS pumps. Therefore, under accident conditions, the AC on-site electrical power system is adequate.

The staff has reviewed the licensee's submittal for the effect of the proposed power uprate on the AC on-site power system, and determined that the licensee has adequately accounted for the effects of the proposed power uprate on the system's functional design. The staff further determined that the AC on-site power system will continue to meet the requirements of GDC 17, following implementation of the proposed power uprate. Therefore, the staff finds the proposed power uprate acceptable with regard to the on-site AC power system.

3.5.3 Direct Current (DC) Power

The licensee assessed the DC loading requirements in the UFSAR and did not identify any reactor power-dependent loads. The licensee concluded that operation at the proposed uprated power level does not increase any loads or revise any control logic.

The staff has reviewed the licensee's submittal for the effect of the proposed power uprate on the DC onsite power system and, based on experience gained from previous power uprate reviews, has determined that the licensee has adequately accounted for the effects of the proposed power uprate on the system's functional design. In short, adequate physical and electrical separation exists, and the system has the capacity and capability to supply power to all safety loads and other required equipment. In addition, the staff concludes that the DC onsite power system will continue to meet the requirements of GDC 17, following implementation of the proposed power uprate. Therefore, the staff finds the proposed power uprate acceptable with regard to the DC onsite power system.

3.5.4 Fuel Pool

The fuel pool cooling and cleanup system (FPCCS) removes heat from the spent fuel assemblies stored in the spent fuel pool (SFP) in order to maintain the pool temperature at, or below, its design temperature during normal plant operations. In addition, the FPCCS reduces activity, maintains water clarity, and maintains the cooling function during and after a seismic event.

The licensee stated that the fuel pool heat load increases slightly as a result of the power uprate. However, the new heat load is within the design basis heat load for the FPCCS, and it will not result in a delay in removing the RHR system from service (i.e., the duration of supplemental cooling will not be increased). The licensee determined that the SFP cooling is adequate by calculating the heat load generated by a full-core discharge plus remaining space filled with spent fuel discharged at regular intervals.

Regarding other fuel pool design considerations, the crud activity and corrosion products in the SFP can increase slightly; however, the licensee determined that this increase is insignificant and the water quality will be maintained by the FPCCS. In addition, the licensee determined that the normal radiation levels around the SFP may increase slightly; however, the licensee determined that the increase will not significantly increase the operational doses to personnel or equipment. Also, there is no effect on the design of the spent fuel racks because the original SFP design temperature is not exceeded.

Based on the staff's review of the licensee's evaluation against the CLB and experience gained from the review of power uprate applications for similar BWR plants, and in view of the foregoing, the staff finds that the FPCCS, in combination with the RHR system, can maintain the SFP temperature at, or below, design limits for all offload conditions at the proposed 1.5% uprated power level.

3.5.5 Water Systems

The licensee stated that the safety-related salt service water (SSW) system serves as the heat sink for all systems cooled by either the reactor building closed cooling water system (RBCCW) or turbine building closed cooling water (TBCCW) system during all planned operations in all operating states. The safety-related performance of the SSW system during and following the DBA is unchanged since it was previously evaluated at 102% of CLTP. The licensee further stated that there is no change in the safety-related heat loads and the resultant increase in heat loads are within the capacity of the RHR and SSW systems.

Regarding nonsafety-related heat loads, the licensee stated that the SSW system functions during emergency conditions to shift flow automatically from the TBCCW heat exchanger to the RBCCW heat exchanger. Sufficient margin remains in the SSW system to ensure that normal operation at the uprated power level does not adversely affect the operation of the SSW system.

The licensee stated that the main condenser, circulating water, and normal heat sink systems are designed to remove the heat rejected to the condenser and maintain a low condenser pressure. The 1.5% power uprate increases the heat rejected to the condenser and may reduce the difference between the operating pressure and minimum condenser vacuum; however, the licensee’s evaluation of the design duty over the actual yearly range of circulating water inlet temperatures determined that the condenser, circulating water system, and the heat sink are adequate for the power uprate.

The licensee stated that the heat loads on the RBCCW system do not increase significantly due to the 1.5% power uprate because they depend on either reactor vessel water temperature or flow rates in the systems cooled by the RBCCW. The change in reactor vessel water temperature is minimal and there is no change in nominal reactor operating pressure. Upon implementation of the power uprate, the fuel pool heat exchangers will reject a slightly greater heat load to the RBCCW; however, the licensee determined that the RBCCW system has adequate design margin to remove the additional heat load.

The licensee stated that the power-dependent heat loads on the TBCCW system, which increase due to the 1.5% power uprate, are rejected to the coolers for the isophase bus, turbine, and generator. The remaining heat loads are not strongly dependent on reactor power and do not increase significantly. The licensee has determined that the TBCCW system has sufficient capacity to remove the additional increase in heat load.

The ultimate heat sink (UHS) is provided by the Atlantic Ocean. The licensee has determined that although the amount of heat discharged to the UHS increases by a small amount, it will have no effect on plant operation. The licensee has administrative limits for the use of the UHS, which limit the allowable inlet and discharge temperatures, as well as the temperature rise between them. The proposed 1.5% power uprate will not change these limits.

Based on the staff’s review of the licensee’s evaluations and the experience gained from the review of power uprate applications for similar BWR plants, the staff has determined that operation at the proposed 1.5% uprated power level does not change the design aspects and operation of the SSW systems, and that operation of the plant's water system will remain bounded by current analyses. Therefore, the staff finds that the impact of plant operations at the proposed uprated power level of these systems is acceptable.

3.5.6 SLC System

The SLC system provides an alternate means of attaining and maintaining cold shutdown conditions, assuming no control rod movement, as required by 10 CFR 50.62.

The license evaluates shutdown capability of the SLC system and the boron solution necessary for each reload cycle in accordance with NRC-approved methodology. The licensee determined that the capability of the SLC system to provide its backup shutdown function is unchanged and will continue to meet the requirements of 10 CFR 50.62. The staff reviewed the licensee's evaluation and concludes that, because the proposed power uprate will not change the operating parameters of the SLC system and the shutdown capability of the system is determined for each reload cycle, the SLC will continue to perform its intended function at the uprated power level.

3.5.7 Power Dependent Heating, Ventilation and Air Conditioning (HVAC)

The function of the HVAC systems is to prevent extreme thermal environmental conditions from impacting personnel and equipment by ensuring that design temperatures are not exceeded. The licensee stated that the HVAC systems that could potentially be affected by the proposed 1.5% power uprate include heating, cooling supply, exhaust, and recirculation units in the turbine building, reactor building, and the drywell.

The licensee stated that the 1.5% power uprate results in a minor increase in heat load caused by the slightly higher FW process temperature ($< 2^\circ$F). The licensee states that the increased heat load is within the margin of the steam tunnel area coolers and is within the drywell cooler capacity. The maximum temperature increases in the FW heater bay and the condenser areas within the turbine building, as a result of the increase in FW process temperatures, are less than 2°F. The increase in heat load, due to a slight SFP cooling process temperature increase within the reactor building, is within the margin of the area coolers. Other areas are unaffected by the 1.5% power uprate because the process temperatures and electrical heat loads are not impacted.

Based on the staff's review of the licensee's evaluation and the experience gained from the review of power uprate applications for similar BWR plants, the staff has determined that operation of the HVAC systems remains acceptable at the uprated conditions.

3.5.8 Fire Protection

The licensee stated that fire detection and suppression systems are not expected to be impacted by plant operation at the proposed 1.5% uprated power level, since there are no physical plant configuration or combustible load changes resulting from the uprated power operations. In addition, the safe shutdown systems and equipment used to achieve and maintain cold shutdown conditions do not change, and the operator actions necessary to mitigate the consequences of a fire are not affected by the uprated conditions.

Based on the staff's review of the licensee's evaluation and the experience gained from the review of power uprate applications and fire protection programs for similar BWR plants, the staff finds that the safe shutdown systems and procedures used to mitigate the consequences of a fire will continue to meet 10 CFR 50.48 and 10 CFR Part 50, Appendix R, and will not be affected by plant operation at the proposed 1.5% power uprate level.

3.5.9 Electrical Power and Auxiliary Systems - Conclusions

Based on the evaluation in Sections 3.5.1 through 3.5.8, the staff concludes that the proposed power uprate is acceptable with respect to its impact on the PNPS electrical power and auxiliary systems.

3.6 Power Conversion Systems

The licensee stated that the PNPS power conversion systems and their support systems are designed for operation at the CLTP level with some margin. The licensee evaluated each system separately for the power uprate conditions, and as described below, where design margin is limited, modifications to systems and components are being implemented such that all systems are able to support operation at the proposed uprated power level in the valves-wide-open (VWO) condition.

3.6.1 Turbine-Generator

The PNPS turbine-generator is designed with a maximum flow-passing and generator capability at rated conditions to ensure that the design rated output is achieved. Because the turbine-generator operates at near VWO conditions, there is currently insufficient margin to support the proposed power uprate. In the application, the licensee stated that the high pressure (HP) turbine steam path will be replaced with a new design that incorporates a 5% flow margin for manufacturing tolerances and reactor pressure control margin. The increased throttle flow is approximately 101.7% of current rated flow. These modifications will increase the electrical power output of the turbine-generator from 697 kW to 709 kW. The increased electrical output remains within the current capacity limits of the generator. In addition, the licensee determined that the rotor missile and turbine overspeed analyses have adequate margin to bound the 1.5% power uprate condition. Finally, the licensee stated that the turbine overspeed trip settings would be changed from 110% and 111%, to 110.6% and 111.6% respectively. The licensee further evaluated the need for the change to the trip settings and stated in its RAI response, dated November 21, 2002, that it has elected not to change the setpoints based on further analysis that demonstrated the modification was unnecessary, since the original settings bounded all analyses for which they are credited.

The staff reviewed the licensee's evaluation and agrees with the licensee, based on the information provided, that the new turbine will operate within the current capacity limits for the electrical output at the proposed uprated power level. Therefore, the existing rotor missile and turbine overspeed analyses remain acceptable for operation at the uprated conditions.

3.6.2 Condenser and Steam Jet Ejectors

The licensee evaluated the impact of the power uprate on condenser performance based on current circulating water system flow. The licensee's evaluation determined that the design margin in the condenser heat removal capability can accommodate the additional heat rejected for operation at the proposed power uprate. Additionally, the licensee determined that air leakage into the condenser does not increase as a result of the proposed 1.5% power uprate, and the small increase in hydrogen and oxygen flows from the reactor do not effect the steam jet air ejector performance because the design was based on operation at flows significantly greater than those needed at the proposed uprated power.

The staff reviewed the licensee's evaluation and agrees with the licensee, based on the information provided, that operation of the condenser and steam jet ejectors at the proposed uprated power level is bounded by the current design capabilities of the condenser and steam jet ejectors. Therefore, the existing condenser and steam jet ejectors remain adequate.

3.6.3 Turbine Steam Bypass

The steam bypass pressure control system was originally designed for a steam flow capacity of a nominal 25% of the 100% rated flow at CLTP. The steam bypass capacity at the proposed power uprate is a nominal 25% of the 100% power uprate RTP steam flow rate. The transient analyses that credit the turbine bypass system availability use the actual capacity. In its RAI response dated November 21, 2002, the licensee stated that the transient is based on the generic evaluations of Appendix E of the TLTR and that the evaluations and conclusions of Appendix E are applicable to PNPS. Appendix E demonstrated that the effect of the TPO uprate is small. Additionally, the licensee will confirm the results of the limiting analysis as part of the normal reload analysis.

Based on the staff's review of the licensee's evaluation against the criteria and analyses in the TLTR, and experience gained from the review of power uprate applications for similar BWR plants, the staff concludes that operation of the steam bypass pressure control system remains acceptable at the uprated conditions.

3.6.4 FW and Condensate Systems

The FW and condensate systems are not safety-related; however, their performance may have an effect on plant availability and the capability to operate reliably at the uprated power conditions. The licensee reviewed the PNPS FW heaters, heater drains, CDs, and FW and condensate pumps to demonstrate that the components are capable of performing in the proper design range and, therefore, provide the slightly higher flow rate for the uprated conditions at the desired temperature and pressure. Additionally, the licensee determined that the FW control valves are capable of maintaining water level control at the uprated power level. The licensee evaluated the operation of the FW and condensate systems and found that sufficient design margin exists in both systems for operation during normal and transient conditions at the proposed power level.

The staff reviewed the licensee's evaluation against the CLB and determined that the FW and condensate system design bounds the operating conditions at the proposed power level. Accordingly, the staff concludes that operation of the FW and condensate systems remains acceptable at the proposed uprated power level.

3.6.5 Condensate Demineralizers

The licensee evaluated the impact of the power uprate on the CDs and determined that no measurable effect results from the power uprate. The licensee determined that the CDs will experience slightly higher loadings and a slight pressure drop at the power uprate conditions. Since the current cleaning frequency is based on an 80-to-100-day cycle rather than on pressure drop, the cleaning frequency is not expected to be affected.

The staff reviewed the licensee's submittal and, based on its knowledge of CD design and operation, determined that the expected cleaning frequency is reasonable. Accordingly, the staff concludes that operation of the CDs remain acceptable at the proposed uprated power level.

3.6.6 Power Conversion Systems - Conclusion

Based on the staff's review, as discussed in Sections 3.6.1 through 3.6.5 above, the staff finds that the power conversion systems, with the identified planned modification of the HP steam path, can accommodate plant operations at the proposed 1.5% uprated power level. Therefore, the staff finds that operation of the power conversion systems at the proposed power level remains acceptable.

3.7 Radwaste and Radiation Sources

3.7.1 Liquid and Solid Waste Management

The liquid radwaste system collects, monitors, processes, stores, and returns processed radioactive waste to the plant for reuse, discharge, or shipment. In their application, the licensee stated that CD resins are cleaned on an 80-to-100-day cycle and are replaced on a schedule of 18 months, based on ion depletion. The licensee stated that operation at the proposed 1.5% power uprate will result in approximately 2% increased flow through the CDs, but is not expected to result in changes to the resin cleaning and replacement schedules. The licensee also stated that the RWCU filter demineralizer may require more frequent replacements, due to slightly higher levels of activation and fission products at the increased power level. The floor drain collector and waste collector subsystems will not experience a significant increase in volume due to operation at the proposed 1.5% power uprate level.

The licensee concluded that the activated corrosion products in the liquid wastes are expected to increase proportionally to the power uprate, and the total volume of processed waste is not expected to increase significantly as a result of the proposed power uprate. The licensee also concluded that the requirements of 10 CFR Part 20 and 10 CFR Part 50, Appendix I, will continue to be met based on a review of the plant operating effluent reports and the slight increases expected from the 1.5% power uprate.

Based on the staff's review of the licensee's evaluation against the CLB, the staff concludes that the requirements of 10 CFR Part 20 and 10 CFR Part 50, Appendix I, applicable to the liquid radwaste management system, will continue to be satisfied at the proposed 1.5% increase in power level since there will only be a slight increase in the volume processed by the system, and the activation and fission products in the liquid wastes.

3.7.2 Gaseous Waste Management

The gaseous waste systems collect, control, process, and dispose of gaseous radioactive waste generated during normal operation and abnormal operational occurrences. The gaseous waste management systems include the offgas system and various building ventilation systems that function to reduce radioactive gaseous releases from the plant.

The licensee stated that the amount of fission products released into the coolant is dependent on the number and nature of fuel rod defects and is not dependent on reactor power. Therefore, the licensee concluded that the activity of airborne effluents released through building vents is not expected to increase significantly due to the proposed 1.5% power uprate. The licensee administratively controls radioactive releases which, therefore, are not a function of reactor core power. The impact of the 1.5% power uprate on the reactor fuel is addressed in Section 3.1 of this safety evaluation.

The licensee evaluated the impact of the 1.5% power uprate on the offgas system, including the effects of hydrogen water chemistry and noble metal injection. The offgas system is designed for 90 standard cubic feet per minute (scfm) of hydrogen and 45 scfm of oxygen from radiolytic decomposition of water. The licensee stated that the current flows at the CLTP level are 58.7 scfm and 34.6 scfm, respectively. The increases in H_2 and $O_{2,}$ due to the 1.5% power uprate, remain well within the capacity of the system and the system radiological release rate is administratively controlled. Therefore, the licensee concluded that gaseous effluents are expected to remain well within the release limits following implementation of the 1.5% power uprate.

Based on a review of the licensee's evaluation against the CLB, the staff concludes that the gaseous radwaste management system will continue to operate acceptably at the proposed uprated conditions, since the activity of the airborne effluents is not expected to increase significantly and releases are administratively controlled.

3.7.3 Radwaste and Radiation Sources - Conclusion

Based on the evaluation in sections 3.7.1 and 3.7.2 of this safety evaluation, the staff concludes that the proposed power uprate is acceptable with respect to its impact on radwaste and radiation sources.

3.8 Reactor Safety Performance Evaluation

3.8.1 AOOs

AOOs are conditions that are expected to occur one or more times in the life of a plant and are initiated by a malfunction, a single failure of equipment, or a personnel error. The applicable acceptance criteria for AOOs are based on GDC 10, 15, and 20, as defined in Appendix A to 10 CFR Part 50.

GDC 10 requires that the reactor core and associated control and protection systems be designed with sufficient margin to ensure that the SAFDLs are not exceeded during normal operation, including AOOs. GDC 15 requires that sufficient margin must be included to ensure that the design conditions of the RCPB are not exceeded during normal operating conditions, including AOOs. GDC 20 requires that the protection system must automatically initiate appropriate systems to ensure that the specified fuel design limits are not exceeded during AOOs.

The PNPS UFSAR documents analyses of a range of potential transients. Chapter 14 of the UFSAR contains the safety analyses of the effects of AOOs resulting from changes in system parameters, such as: (1) a decrease in core coolant temperature; (2) an increase in reactor pressure; (3) a decrease in reactor core coolant flow rate; (4) reactivity and power distribution anomalies; (5) an increase in reactor coolant inventory; and (6) a decrease in reactor coolant inventory.

The licensee analyzes plant responses to the most limiting transients for each reload cycle and uses the results to establish the thermal limits. A potentially limiting event is an event that has the potential to affect the core operating and safety limits. In their application, the licensee deferred performing the limiting transient analyses for the proposed power uprate to the reload analyses for the implementation cycle, which is consistent with the approach specified in the TLTR. Appendix E to the TLTR includes a discussion of the GENE analytical methods and codes used to perform the transient analyses and presents the changes in the OLMCPR for previous uprate transient analyses. Experience gained from previous power uprates indicate that changes attributable to the 1.5% power uprate would be insignificant.

Based on experience gained from the review of previous power uprate submittals, and the fact that the licensee will perform reload analyses using NRC-approved GENE methodology with appropriate cycle-specific inputs, the staff finds that deferring the limiting transient analyses to the reload analysis for the cycle implementing the power uprate is acceptable with regard to the impact of AOOs at the proposed uprated power level.

3.8.2 DBAs

The staff reviewed the impact of the proposed changes on DBA dose consequences analyses, as documented in Chapter 14 of the Pilgrim UFSAR. In their submittal, the licensee stated that the current DBA dose consequences analyses of record for Pilgrim were evaluated at a power level at 2% above the CLTP level of 1998 MWt. The staff reviewed the information in the Pilgrim UFSAR, and was not able to confirm that the DBA dose consequences analyses were performed at a power greater than the CLTP level of 1998 MWt. In an RAI dated August 30, 2002, the staff requested that the licensee confirm that the dose consequences analyses for all DBAs were performed at a power level that bounds the requested uprate.

By letter dated September 27, 2002, the licensee stated that the DBA dose consequence analyses were performed at 102% of the CLTP, with the exception of the LOCA and the MS line break (MSLB). The licensee performed updated calculations for the LOCA and MSLB that assumed a power level equal to 102% of the CLTP and included assumptions and modeling consistent with current regulatory guidance. The licensee provided results of these calculations and a description of the analysis changes in RAI responses dated September 27, 2002, and February 4, 2003.

The staff reviewed the licensee’s application and RAI responses and determined that Pilgrim’s CLB DBA dose consequences analyses include assumptions and inputs that are based on 102% of the CLTP level. The staff determined that these analyses bound the conditions expected for the proposed 1.5% power uprate. Therefore, the staff concluded that the CLB dose analyses is bounding for the proposed 1.5% power uprate.

Based upon the above discussion, the staff has determined that the dose consequences of DBAs for a 1.5% increase to the RTP level would be bounded by the doses estimated in previous analyses. Accordingly, the staff finds that there is reasonable assurance that dose consequences of DBAs meet the dose limits given in 10 CFR Part 100, 10 CFR Part 50, and GDC-19, as well as applicable dose acceptance criteria given in NUREG-0800, Standard Review Plan, Chapter 15. Therefore, the staff finds that the dose consequences of DBAs are acceptable for the proposed 1.5% power uprate.

3.8.3 Special Events

3.8.3.1 ATWS

ATWS is defined as an AOO with failure of the reactor protection system to initiate a reactor scram to terminate the event. The requirements for ATWS are specified in 10 CFR 50.62. For BWR facilities, 10 CFR 50.62 requires the following mitigating features for an ATWS event:

(1) a standby liquid control (SLC) system with the capability of injecting a borated water solution, with reactivity control equivalent to the control obtained by injecting 86 gpm of a 13 weight percent sodium pentaborate decahydrate solution at the natural boron-10 isotope abundance into a 251-inch inside-diameter reactor vessel;

(2) an alternate rod injection (ARI) system that is designed to perform its function in a reliable manner, and that is independent from the reactor trip system from sensor output to the final actuation device; and

(3) equipment to trip the reactor coolant recirculation pumps automatically under conditions indicative of an ATWS

The PNPS design meets the ATWS mitigation requirements defined in 10 CFR 50.62 by providing: (1) boron injection capability with appropriate reactivity control; (2) an alternate rod insertion system; and (3) an automatic recirculation pump trip logic for ATWS conditions.

BWR performance during an ATWS is also compared to the criteria used in the development of the ATWS safety analyses described in NEDO-24222, “Assessment of BWR Mitigation of ATWS,” Volume II, dated December 1979. The PNPS-specific ATWS analysis at the CLTP demonstrates that the following ATWS acceptance criteria are met: (1) the peak vessel bottom pressure is less than the ASME Service Level C limit of 1500 psig; (2) the PCT remains below the 10 CFR 50.46 limit of 2200°F; (3) the cladding oxidation remains below the limit specified in 10 CFR 50.46; (4) peak suppression pool temperature is less than 185°F; and (5) the peak containment pressure does not exceed the maximum containment design pressure of 62 psig.

Section 5.3.5 and Appendix L of the TLTR, provide the generic criteria for ATWS analysis related to power uprates. The licensee evaluated an ATWS event at the proposed 1.5% power uprate based on these criteria, and determined that no PNPS-specific ATWS suppression pool analysis is needed for operation at the proposed power uprate. However, the licensee determined that PNPS did not have sufficient margin to the ATWS ASME Service Level C peak vessel bottom pressure limit of 1500 psig at the CRTP to apply the TLTR criteria. Therefore, the licensee performed a plant-specific ATWS analysis for the uprated power level.

The plant-specific ATWS analysis is based on an increased SRV capacity and a drift uncertainty, about the mean, of 22 psid. In its RAI response dated November 21, 2002, the licensee provided additional discussion on the basis for the SRV setpoint drift value used in the analysis. The licensee stated that the “as-found” setpoint deviation from 1987 to 2001 is +0.92% (10 psi), which is within the ±11 psi setpoint tolerance specified in the TSs. The licensee stated that the average setpoint value used in the ATWS analysis is 1128.5 psig, which bounds the average as-found setpoint of value 1125 psig. In performing its review of the RAI response, the staff noted that the 22 psi drift uncertainty is not added to the nominal value for all of the valves (1115 psi + 22 psi) for the ATWS analysis. Instead, the ATWS analysis assumes one SRV actuates at 1136 psig and three SRVs actuate at 1126 psig (1% upper setpoint tolerance). The licensee is not changing the PNPS SRV setpoints to support operation at the uprated conditions and the ATWS analysis conservatively assumed one valve lifts at 1136 psig.

The results of the plant-specific ATWS analysis determined that the most limiting event is an event in which a pressure regulator fails open at the beginning of the cycle. The peak vessel bottom pressure calculated for the 1.5% power uprate condition is 1495 psig. This value is below the 1500 psig limit for ATWS acceptance criteria. Therefore, the licensee concludes that ATWS acceptance criteria are met for the proposed 1.5% power uprate.

The staff reviewed the licensee’s evaluation of ATWS. The staff determined that, based on the margin criteria provided in the TLTR, and the results of the licensee’s ATWS pressurization analyses, PNPS will continue to meet the requirements specified in 10 CFR 50.62 for operations at the uprated power level.

3.8.3.2 Station Blackout

In accordance with the requirements of 10 CFR 50.63, the reactor core and the associated coolant, control, and protection systems must have sufficient capacity to cool the core and maintain containment integrity in the event of a station blackout (SBO) event for a specified duration. RG 1.155 and NUMARC 87-00 describes methods acceptable to the staff for meeting the requirements of 10 CFR 50.63.

The licensee evaluated the affect of the proposed 1.5% power uprate on the plant’s coping capability in the event of an SBO. PNPS has margins of 15,000 gallons for the condensate storage inventory volume and 50°F for the containment peak temperature limit. These margins are well in excess of the margin criteria specified in the TLTR. Therefore, based on the generic evaluations performed in the TLTR for power uprates, the licensee determined that no PNPS-specific SBO analysis is necessary for the proposed 1.5% power uprate.

Based on the staff’s review of the licensee’s evaluation, the staff has determined that the existing SBO analyses will remain bounding and will be acceptable for operation at the proposed 1.5% power uprate.

3.8.4 Reactor Safety Performance Evaluations - Conclusions

Based on the evaluations in Sections 3.8.1 through 3.8.3 of this SE, the staff concludes that the proposed power uprate is acceptable with respect to its impact on reactor safety performance.

3.9 Other Evaluations

3.9.1 High-Energy Line Break Analysis (HELB)

The licensee stated that since operating temperatures and pressures change only slightly at the uprated power conditions, the HELB mass and energy releases do not change significantly. There is no change in the nominal vessel dome pressure at the proposed power uprate level. The proposed power uprate does not change the postulated break locations and the piping configuration will not change. The existing HELB analyses were performed assuming a value of 102% of the CLTP level which bounds the proposed 1.5% power uprate condition. Therefore, the licensee concluded that the existing HELB analysis, break locations, pipe whip and jet impingement analyses remain unchanged. The existing pipe whip restraints, jet impingement shields, and their supporting structures are also adequate for the proposed 1.5% power uprate condition.

Based on the staff’s review of the licensee’s evaluation, the staff has determined that the existing analyses for HELB will remain bounding and is, therefore, acceptable for operations at the proposed uprated power level.

3.9.2 Environmental Qualification (EQ)

In their application, as supplemented by letters dated December 30, 2002, February 10, 2003, and March 17, 2003, the licensee addressed power uprate issues related to EQ of electrical equipment. The licensee stated that they performed the current MSLB, DBA-LOCA, and containment analyses at 102% of CLTP. Therefore, the equipment qualification envelope inside containment continues to be applicable for operation at the uprated power conditions. The licensee also stated that the normal radiation profiles, both inside and outside containment, and the accident temperature, pressure, and humidity environments outside containment were based on the CLTP. The licensee performed an analysis and determined that all of the equipment, both inside and outside of the containment, is qualified and the installed equipment remains within its qualification envelope at the uprated conditions.

The staff has reviewed the licensee’s evaluation of the effects of the proposed power uprate on the EQ of the electrical equipment and has determined that the electrical equipment remains within its qualification envelope at the proposed uprated power conditions. Accordingly, the staff concludes that the equipment continues to meet the relevant requirements of 10 CFR 50.49. Therefore, the staff finds the proposed power uprate acceptable with respect to EQ of electrical equipment.

3.9.3 Operating Training and Human Factors

3.9.3.1 Operating Procedures

In the November 21, 2002, supplement to its application, the licensee stated that they have reviewed all systems and conducted a review of industry experience to identify all necessary modifications associated with the power uprate. The licensee further stated that existing procedures (normal, emergency, and abnormal) are being revised and new procedures will be developed, where appropriate. The licensee stated that all required procedure modifications will be completed prior to operation above the CLTP level.

The staff finds that the licensee has appropriately identified plant procedures impacted by the 1.5% power uprate and indicated that new procedures will be developed and existing procedures revised, as appropriate. The staff concludes, therefore, that the PNPS operating procedures will appropriately address plant operation at the uprated conditions.

3.9.3.2 Risk-Important Operator Actions Sensitive to Power Uprate

The licensee stated in the July 5, 2002, application that "...for [thermal power optimization] TPO conditions, operator response to transient, accident and special events are not affected. Operator actions for maintaining safe shutdown, core cooling, and containment cooling, etc., do not change for the TPO uprate."

The staff concludes that the licensee has adequately addressed the question of operator actions sensitive to the power uprate by describing the lack of affect on operator response and actions.

3.9.3.3 Control Room and Simulator Controls, Displays, and Alarms

In the November 21, 2002, supplement, the licensee stated that the control room and training simulator alarms and displays will be modified to alert operators to off-normal conditions associated with the power uprate. In addition, the licensee stated that the alarms and displays will assist the operators in resolving any problems with the new equipment. A new annunciator marked "FW FLOW CORRECTION FACTOR TROUBLE ALARM," will alert the operators to trouble with the FW Flow Correction Factor. The licensee will install new Emergency and Plant Information Computer displays to indicate when "Trouble" conditions exists and provide the operator with details of any problems in the system. Additionally, the licensee will develop new procedures for the operation of the equipment, and train the operators in the proper implementation of the procedures. Simulator changes and validation for the power uprate will be performed in accordance with PNPS procedures.

The staff finds the licensee has adequately identified the changes that will occur to controls, displays, and alarms as a result of the power uprate and appropriately described how these changes will be implemented. The staff concludes, therefore, that the control room and simulator controls, displays, and alarms will be appropriately modified to support operations at the uprated conditions.

3.9.3.4 Safety Parameter Display System

In the November 21, 2002, supplement, the licensee stated that changes will be made to the Safety Parameter Display System (SPDS) screens to reflect the new blowdown rates from the enlarged SRVs. The changes will involve displays 37 and 38, “Heat Capacity Temperature Limits,” and 39, “Boron Injection Initiation Temperature.”

The staff finds the licensee has adequately identified the changes that will occur to the SPDS as a result of the power uprate and described how the changes will be implemented. The staff concludes, therefore, that the SPDS will be appropriately modified to support operations at the uprated conditions.

3.9.3.5 Operator Training Program

In the July 5, 2002, application the licensee stated that no additional training, other than normal training for plant changes, is required to operate the plant at the power uprate level. Minor changes to the power/flow map, flow-referenced setpoint, and changes to the TSs will be communicated through routine operator training prior to operation at the power uprate level. In the November 21, 2002, supplement, the licensee stated that plant operators will be trained in the use of the new system prior to implementation of the power uprate.

The staff finds the licensee has adequately addressed the changes to the operator training program to ensure operators have the necessary skills and knowledge to operate the plant at the increased power level. The licensee has also identified that operators will be trained prior to implementation of the power uprate. The staff concludes, therefore, that the operator training program will be appropriately modified to address changes associated with operations at the uprated conditions.

3.9.3.6 Operating Training and Human Factors - Conclusion

Based on the evaluation in Sections 3.9.4.1 through 3.9.5.5 of this SE, the staff concludes that the proposed power uprate is acceptable with respect to its impact on operator training and human factors considerations.

3.10 Facility Operating License and Technical Specifications Changes

The licensee proposed to revise the FOL and TSs as follows to reflect the increase in licensed power level from 1998 MWt to 2028 MWt:

A. Paragraph 3.A in FOL DPR-35, “Maximum Power Level,” would be revised to authorize operation at a steady state reactor core power level not in excess of 2028 MWt thermal (100% power).

B. The definition of DESIGN POWER in TS 1.0 would be revised to reflect the increase from 1998 MWt to 2028 MWt.

C. Note 1D for Table 3.1.1, “Reactor Protection System (SCRAM) Instrumentation Requirement,” in the TS would be revised to “Reduce power to less than 32.5% of design” instead of 45% of design.

The staff has modified the License with the following conditions:

A. Install the Westinghouse/AMAG Crossflow ultrasonic flow measurement (UFM) instrumentation system under the supervision of the vendor and in accordance with the vendor's recommendations as described in the July 5, 2002, application and evaluated in the NRC staff's associated Safety Evaluation dated May 9, 2003. The installation will be in accordance with the ABB Combustion Engineering Nuclear Power Topical Report CENP-397-P, Revision 1. The Crossflow UFM will be calibrated in accordance with the plant-specific piping configuration. The Updated Final Safety Analysis Report (UFSAR) shall be revised to reflect this in the next update of the UFSAR submitted to the NRC pursuant to 10 CFR 50.71(e).

B. The Crossflow UFM system software will be controlled in accordance with the Entergy Software Quality Assurance program for Level B software. Plant procedures will be developed or revised, where appropriate, for the Crossflow UFM operation; maintenance; calibration; control of software and hardware system configuration; identifying, reporting, and correcting system deficiencies; and for addressing manufacturer-identified deficiencies as described in the July 5, 2002, application and evaluated in the NRC staff's associated Safety Evaluation dated May 9, 2003.

C. Finalize the feedwater flow measurement uncertainty calculation based on installed operational data as described in the July 5, 2002, application and evaluated in the NRC staff's associated Safety Evaluation dated May 9, 2003. The UFSAR shall be revised to reflect this in the next update of the UFSAR submitted to the NRC pursuant to 10 CFR 50.71(e).

D. Modify the Safety Parameter Display System as described in the licensee's November 21, 2002, letter, and evaluated in the NRC staff's associated Safety Evaluation dated May 9, 2003. The UFSAR shall be revised to reflect this in the next update of the UFSAR submitted to the NRC pursuant to 10 CFR 50.71(e).

The FOL and TS changes reflect the proposed increase in licensed power level based on installation of the CROSSFLOW UFM system for FW flow and temperature measurements. Based on the evaluations discussed in Section 3.1 through 3.9 of this SE, the staff concludes that the above-described changes to the FOL and TSs are acceptable.

Additionally, the following TS bases changes would be made to support the propose change in steady state power level:

1. Bases 3.1, REACTOR PROTECTION SYSTEM, page B3/4.1-2: Revise "45% of rated core thermal power," to "32.5% of rated core thermal power."

2. Bases 3.1, REACTOR PROTECTION SYSTEM, page B3/4.1-3: Revised design power from "1998 MWt" to "2028 MWt."

The proposed TS bases changes are consistent with the changes to the license and TSs. The staff has no objections to the TS bases changes.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. 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 (67 FR 56322). 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 amendments.

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.

Principal Contributors: C. Long, C. Lauron, C. Wu, M. Hartzman, Z. Abdullahi, M. Hart,
N. Trehan, P. Rebstock, T. Tate

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