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

August 18, 2010

Mr. Mark A. Schimmel
Site Vice President
Prairie Island Nuclear Generating Plant
Northern States Power – Minnesota
1717 Wakonade Drive East
Welch, MN 55089-9642

SUBJECT: PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2 – AMENDMENT RE: MEASUREMENT UNCERTAINTY RECAPTURE POWER UPRATE (TAC NOS. ME3015 AND ME3016)

Dear Mr. Schimmel:

The U.S Nuclear Regulatory Commission has issued the enclosed Amendment No. 197 to Facility Operating License No. DPR-42 and Amendment No. 186 to Facility Operating License No. DPR-60 for the Prairie Island Nuclear Generating Plant, Units 1 and 2, respectively. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated December 28, 2009, as supplemented by letters dated April 19, April 23, and June 17, 2010.

The amendments revise the license and TSs to reflect a 1.64 percent increase in the licensed rated thermal power (RTP) from 1650 megawatts thermal (MWt) to 1677 MWt. The RTP increase is based on reduced uncertainty in the RTP measurement achieved by installation of a Caldon Leading Edge Flow Meter CheckPlus™ System used to measure FW flow and temperature.

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

Sincerely,

Thomas J. Wengert, Senior Project Manager
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-282 and 50-306

Enclosures:
1. Amendment No. 197 to DPR-42
2. Amendment No. 186 to DPR-60
3. Safety Evaluation

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

NORTHERN STATES POWER COMPANY - MINNESOTA

DOCKET NO. 50-282

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No.197
License No. DPR-42

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

   A. The application for amendment by Northern States Power Company, a Minnesota Corporation (NSPM, the licensee), dated December 28, 2009, as supplemented by letters dated April 19, April 23, and June 17, 2010, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(1) and 2.(C).(2) of Facility Operating License No. DPR-42 is hereby amended to read as follows:

   (1) Maximum Power Level

   NSPM is authorized to operate the facility at steady state reactor core power levels not in excess of 1677 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendices A, as revised through Amendment No. 197 , are hereby incorporated into the license. NSPM shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 180 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Joseph G. Giitter

Joseph G. Giitter, Director
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Facility Operating License
and Technical Specifications

Date of Issuance: August 18, 2010

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

NORTHERN STATES POWER COMPANY - MINNESOTA

DOCKET NO. 50-306

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 186
License No. DPR-60

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

   A. The application for amendment by Northern States Power Company, a Minnesota Corporation (NSPM, the licensee), dated December 28, 2009, as supplemented by letters dated April 19, April 23, and June 17, 2010, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(1) and 2.C.(2) of Facility Operating License No. DPR-60 is hereby amended to read as follows:

   (1) Maximum Power Level

   NSPM is authorized to operate the facility at steady state reactor core power levels not in excess of 1677 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendices A, as revised through Amendment No. 186, are hereby incorporated in the license. NSPM shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented by within 180 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Joseph G. Giitter, Director
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Facility Operating License
and Technical Specifications

Date of Issuance: August 18, 2010

ATTACHMENT TO LICENSE AMENDMENT NOS. 197 AND 186

FACILITY OPERATING LICENSE NOS. DPR-42 AND DPR-60

DOCKET NOS. 50-282 AND 50-306

Replace the following pages of Facility Operating License Nos. DPR-42 and DPR-60 with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
DPR-42, License Page 3	DPR-42, License Page 3
DPR-60, License Page 3	DPR-60, License Page 3

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
1.1-5	1.1-5
5.0-36	5.0-36
5.0-37	5.0-37

(4) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, NSPM to receive, possess and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument and equipment calibration or associated with radioactive apparatus or components;

(5) Pursuant to the Act and 10 CFR Parts 30 and 70, NSPM to possess but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility;

(6) Pursuant to the Act and 10 CFR Parts 30 and 70, NSPM to transfer byproduct materials from other job sites owned by NSPM for the purpose of volume reduction and decontamination.

C. This amended license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

NSPM is authorized to operate the facility at steady state reactor core power levels not in excess of 1677 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 197 , are hereby incorporated in the license. NSPM shall operate the facility in accordance with the Technical Specifications.

(3) Physical Protection

NSPM shall fully implement and maintain in effect all provisions of the Commission-approved physical security, guard training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans, which contains Safeguards Information protected under 10 CFR 73.21, is entitled: "Prairie Island Nuclear Generating Plant Security Plan, Training and Qualification Plan, Safeguards Contingency Plan, and Independent Spent Fuel Storage Installation Security Program," Revision 1, submitted by letters dated October 18, 2006, and January 10, 2007.

(5) Pursuant to the Act and 10 CFR Parts 30 and 70, NSPM to possess but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility;

(6) Pursuant to the Act and 10 CFR Parts 30 and 70, NSPM to transfer byproduct materials from other job sites owned by NSPM for the purposes of volume reduction and decontamination.

C. This amended license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) <u>Maximum Power Level</u>

NSPM is authorized to operate the facility at steady state reactor core power levels not in excess of 1677 megawatts thermal.

(2) <u>Technical Specifications</u>

The Technical Specifications contained in Appendix A, as revised through Amendment No. 186 , are hereby incorporated in the license. NSPM shall operate the facility in accordance with the Technical Specifications.

(3) <u>Physical Protection</u>

NSPM shall fully implement and maintain in effect all provisions of the Commission-approved physical security, guard training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans, which contains Safeguards Information protected under 10 CFR 73.21, is entitled: "Prairie Island Nuclear Generating Plant Security Plan, Training and Qualification Plan, Safeguards Contingency Plan, and Independent Spent Fuel Storage Installation Security Program," Revision 1, submitted by letters dated October 18, 2006, and January 10, 2007.

1.1 Definitions (continued)

PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)	The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, and the OPPS arming temperature for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.6. Plant operation within these operating limits is addressed in LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) –Reactor Coolant System Cold Leg Temperature (RCSCLT) > Safety Injection (SI) Pump Disable Temperature," and LCO 3.4.13, "Low Temperature Overpressure Protection (LTOP) - Reactor Coolant System Cold Leg Temperature (RCSCLT) ≤ Safety Injection (SI) Pump Disable Temperature."
QUADRANT POWER TILT RATIO (QPTR)	QPTR shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 1677 MWt.
REACTOR TRIP SYSTEM (RTS) RESPONSE TIME	The RTS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RTS trip setpoint at the channel sensor output until opening of a reactor trip breaker. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

18. WCAP-7908-A, "FACTRAN – A FORTRAN IV Code for Thermal Transients in a UO_2 Fuel Rod";

19. WCAP-7907-P-A, "LOFTRAN Code Description";

20. WCAP-7979-P-A, "TWINKLE – A Multidimensional Neutron Kinetics Computer Code";

21. WCAP-10965-P-A, "ANC: A Westinghouse Advanced Nodal Computer Code";

22. WCAP-11394-P-A, "Methodology for the Analysis of the Dropped Rod Event";

23. WCAP-11596-P-A, "Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores";

24. WCAP-12910 Rev. 1-A, "Pressurizer Safety Valve Set Pressure Shift";

25. WCAP-14565-P-A, "VIPRE-01 Modeling and Qualification for pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis";

26. WCAP-14882-P-A, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses";

27. WCAP-16009-P-A, "Realistic Large Break LOCA Evaluation Methodology Using Automated Statistical Treatment of Uncertainty Method (ASTRUM)";

28. Caldon Engineering Report ER-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM System"; and

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

29. Caldon Engineering Report ER-157P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM or LEFM CheckPlus System".

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. RCS pressure and temperature limits for heat-up, cooldown, low temperature operation, criticality, and hydrostatic testing, OPPS arming, PORV lift settings and Safety Injection Pump Disable Temperature as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits";
LCO 3.4.6, "RCS Loops - MODE 4";
LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled";
LCO 3.4.10, "Pressurizer Safety Valves";
LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) – Reactor Coolant System Cold Leg Temperature (RCSCLT) > Safety Injection (SI) Pump Disable Temperature";
LCO 3.4.13, "Low Temperature Overpressure Protection (LTOP) – Reactor Coolant System Cold Leg Temperature (RCSCLT) ≤ Safety Injection (SI) Pump Disable Temperature"; and
LCO 3.5.3, "ECCS - Shutdown".

UNITED STATES
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SAFETY EVALUATION BY THE OFFICE NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 197 AND 186 TO

FACILITY OPERATING LICENSE NOS. DPR-42 AND DPR-60

NORTHERN STATES POWER COMPANY - MINNESOTA

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2

DOCKET NOS. 50-282 AND 50-306

1.0 INTRODUCTION

By letter dated December 28, 2009 (Reference 1), as supplemented by letters dated April 19, 2010 (Reference 2), April 23, 2010 (Reference 3), and June 17, 2010 (Reference 4), Northern States Power Company – Minnesota (NSPM, the licensee) requested an amendment to the operating license and plant technical specifications (TSs) to increase the licensed rated thermal power (RTP) as a result of a measurement uncertainty recapture (MUR) power uprate (PU) for the Prairie Island Nuclear Generating Plant (PINGP), Units 1 and 2.

The proposed change revises the license and TSs to reflect an increase in the licensed RTP power from 1650 to 1677 megawatts thermal (MWt) (a 1.64 percent increase). The increase in RTP is based on the use of Caldon CheckPlus™ Leading Edge Flow Meter (LEFM) ultrasonic flow meter (UFM) instrumentation. This instrumentation would decrease the uncertainty in the measurement of feedwater (FW) flow, thereby decreasing the power level measurement uncertainty from 2.0 percent to 0.36 percent. This type of application is commonly referred to as an MUR-PU.

Specifically, the licensee proposes the following changes:

- Paragraphs 2.C.(1) in Facility Operating License Nos. DPR-42 and DPR-60 will be revised to authorize a steady state reactor core thermal power level not in excess of 1677 MWt.
- The definition of RATED THERMAL POWER (RTP) in TS 1.1, page 1.1-5, will be revised to reflect an increase from 1650 to 1677 MWt.
- TS 5.6.5, Core Operating Limits Report, will be revised to add Caldon Engineering Reports 80P and 157P.

The supplemental letters dated April 19, April 23, and June 17, 2010, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on May 11, 2010 (75 FR 26291). The April 23, 2010, letter reduced the scope of the original application by withdrawing a requested change to the "Pressure Temperature Limits Report (PTLR)." The updated PTLR analysis methodology is not required for the MUR-PU and the existing pressure-temperature limit curves remain valid.

The licensee developed the application using the guidance of NRC Regulatory Issue Summary (RIS) 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications," issued January 31, 2002 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML013530183). To improve the efficiency of the NRC staff review, guidance is provided in RIS 2002-03 for licensees requesting an MUR-PU. When not accompanied by other requests or changes, applications for power uprates that are based on improved FW measurement techniques should have a limited effect on plant analyses and equipment. When licensees submit applications for this type of power uprate, the staff will focus its review on "affected areas," or those areas for which existing analyses of record do not bound plant operation at the proposed power level and, as a result, new analyses or evaluations should be performed to provide a basis for operation at the proposed power level. Similarly, "affected equipment" includes equipment for which existing analyses of record for capacities and/or design, actual design, and/or operational or licensing functional requirements should be modified to support operation at the proposed power level.

Neutron flux instrumentation is calibrated to the core thermal power, which is determined by an automatic or manual calculation of the energy balance around the plant nuclear steam supply system. This calculation is called "secondary calorimetric" for a pressurized-water reactor (PWR). The accuracy of this calculation depends primarily upon the accuracy of FW flow and FW net enthalpy measurements. FW flow is the most significant contributor to the core thermal power uncertainty. An accurate measurement of this parameter will result in an accurate determination of core thermal power.

FW flow rate is typically measured using a venturi. This device generates a differential pressure proportional to the square of the FW velocity in the pipe. Because of the high cost of calibrating the venturi and the need to improve flow instrumentation measurement uncertainty, the industry evaluated other flow measurement techniques and found the Caldon LEFM Check and LEFM CheckPlus™ UFMs to be a viable alternative.

This power uprate is based on a reduced measurement uncertainty of core thermal power resulting from the installation of a Cameron (formerly Caldon) LEFM CheckPlus™ system to measure FW flow and temperature at PINGP. The licensee's submittal referenced Cameron Topical Report ER-80P, Revision 0, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the System," issued in March 1997 (ADAMS Accession No. 9807210146), and its supplement, Topical Report ER-157P, Revision 5, "Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM √™ or LEFM CheckPlus™ System," issued in October 2001 (ADAMS Accession No. ML013440078).

Topical Report ER-80P describes the LEFM technology, includes calculations of power measurement uncertainty using a Caldon LEFM Check system in a typical two-loop PWR or two FW line boiling-water reactor (BWR), and provides guidelines and equations for determining the plant-specific power calorimetric uncertainties. ER-80P was approved by the NRC staff on March 8, 1999 (ADAMS Accession No. 9903190053), for use in justification of MUR-PUs up to 1 percent. Its supplement, Topical Report ER-157P, describes the Caldon LEFM CheckPlus™

system and lists nonproprietary results of a typical PWR or BWR thermal power measurement uncertainty calculation using either the Caldon LEFM Check or LEFM CheckPlus™ system. ER-157P was approved by the NRC staff on December 20, 2001 (ADAMS Accession No. ML013540256), for use in justifying MUR-PUs up to 1.7 percent. Together, these two reports provide a generic basis and guidelines for power uprate.

The NRC has recently issued similar MUR-PU license amendments for Vogtle Electric Generating Plant, Units 1 & 2 on February 27, 2008 (ADAMS Accession No. ML080350347), Cooper Nuclear Station on June 30, 2008 (ADAMS Accession No. ML081540280), Davis-Besse Nuclear Power Station, Unit 1 on June 30, 2008 (ADAMS Accession No. ML081410652), Calvert Cliffs Nuclear Power Plant on July 22, 2009 (ADAMS Accession No. ML091820366), and North Anna Power Station on October 22, 2009 (ADAMS Accession No. ML092260616).

2.0 BACKGROUND

Nuclear power plants are licensed to operate at a specified maximum core thermal power, often called rated thermal power (RTP). Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Appendix K, formerly required licensees to assume that the reactor has been operating continuously at a power level at least 1.02 times the licensed power level when performing loss-of-coolant accident (LOCA) and emergency core cooling system (ECCS) analyses. This requirement was included to ensure that instrumentation uncertainties were adequately accounted for in the analyses. In practice, many of the design bases analyses assume a 2 percent power uncertainty, consistent with 10 CFR Part 50, Appendix K.

A revision to 10 CFR Part 50, Appendix K, effective on July 31, 2000, allows licensees to use a power level less than 1.02 times the RTP, but not less than the licensed power level, based on the use of state-of-the art FW flow measurement devices that provide a more accurate calculation of power. Licensees can use a lower uncertainty in the LOCA and ECCS analyses provided the licensee has demonstrated that the proposed value adequately accounts for instrumentation uncertainties. Because there continues to be substantial conservatism in other Appendix K requirements, sufficient margin to ECCS performance in the event of a LOCA is preserved.

However, the final rule by itself did not allow increases in licensed power levels. Because the licensed power level for a plant is a TS limit, proposals to raise the licensed power level must be reviewed and approved under the license amendment process. The PINGP is currently licensed to operate at a maximum power level of 1650 MWt, which includes a 2 percent margin in the ECCS evaluation model to allow for uncertainties in core thermal power measurement as was previously required by 10 CFR Part 50, Appendix K. Currently, with the RTP of 1650 MWt, an analytical power level of 1683 MWt (102 percent of 1650 MWt) is used in the safety analysis. With a requested revised RTP of 1677 MWt and a measurement uncertainty of 0.36 percent, the analytical power level is unchanged at 1683 MWt.

The desired MUR-PU will be accomplished by increasing the electrical demand on the turbine-generator. As a result of this demand increase, steam flow will increase and the resultant steam pressure will decrease. At full power, the reactor coolant system (RCS) nominal cold-leg temperature will decrease slightly while the hot-leg temperature will increase slightly in response to the increased steam flow demand. As a result, the RCS average temperature will increase slightly.

The NRC staff finds that the LEFM-assisted core thermal power measurement uncertainty is limited to 0.36 percent of actual reactor thermal power and, therefore, can support the proposed 1.64 percent power uprate. This result is a proposed increase of 1.64 percent in the PINGP licensed power level using current NRC-approved methodologies.

3.0 TECHNICAL EVALUATION

3.1 Human Factors

3.1.1 Regulatory Evaluation

The human factors review addresses programs, procedures, training, and plant design features related to human performance during normal and accident conditions. The NRC staff's human factors evaluation was conducted to confirm that human performance would not be adversely affected as a result of system and procedure changes made to implement the proposed MUR-PU. The scope of the review included changes to operator actions, human-system interfaces, and procedures and training needed for the proposed MUR-PU.

3.1.2 Technical Evaluation

The NRC staff has developed a standard set of questions for human factors reviews of proposed MUR-PU license amendment requests (LARs) in RIS 2002-03, Attachment 1, Section VII, items 1 through 4. The following sections evaluate the licensee's response to these questions in the LAR.

3.1.2.1 Operator Actions

The licensee stated in its submittal that it performed a review of operator actions and time available for operator actions. The licensee stated that the result of the review determined that existing required operator actions are not affected by the MUR-PU. The review determined engineered safety features (ESF) system design and setpoints are based on 1683 MWt, which is higher than the 1677 MWt proposed for the MUR-PU. The licensee also stated that there is no reduction in the time required for the necessary operator actions. The licensee concluded that operator actions and the time available for operator actions to be completed will be unaffected by the MUR-PU. The licensee also determined that there were no new manual actions or automation of existing operator actions required as a result of the MUR-PU.

The NRC staff reviewed the licensee's submittal statements relating to any impacts of the MUR-PU to existing or new operator actions credited in the safety analyses. The NRC staff concludes that the proposed MUR-PU will not adversely impact operator actions and their response times because there were no changes identified. The NRC staff finds that the statements provided by the licensee are in conformance with Section VII.1 of Attachment 1 to RIS 2002-03.

3.1.2.2 Emergency and Abnormal Operating Procedures

The licensee stated in its submittal that the current emergency operating procedures (EOPs) and abnormal operating procedures (AOPs) requirements are based on 1683 MWt, which, as stated previously, bounds the proposed MUR-PU. The licensee concluded that no changes to operator actions are required due to the MUR-PU. The licensee identified a Control Room Alarm Response Procedure that will be revised to incorporate the administrative restrictions for allowable plant operating power level based on actual Caldon LEFM system status.

The NRC staff concludes that the proposed MUR-PU does not present any adverse impacts on the EOPs and AOPs. This conclusion is based upon the current EOPs and AOPs being associated with a thermal power greater than the 1677 MWt and the identified changes necessary to facilitate the MUR-PU to be reflected in the operator training program prior to MUR-PU implementation. The NRC staff finds that the statements provided by the licensee are in conformance with Sections VII.2.A, VII.3, and VII.4 of Attachment 1 to RIS 2002-03.

3.1.2.3 Control Room Controls, Displays, and Alarms

The licensee described in its submittal the effects of the MUR-PU on the control room controls and displays, including the Safety Parameter Display System (SPDS) and alarms. The Emergency Response Computer System (ERCS) serves as the plant computer system, as well as the SPDS. The LEFM is not relied upon for any emergency procedure actions; therefore, there are no changes to the SPDS displays.

The licensee states that the following changes will occur prior to implementation of the proposed MUR-PU:

- Re-scaling of the plant control and protection instrumentation consistent with the increase in 100% nominal core power from 1650 MWt to 1677 MWt.

- Revision of the Control Room Alarm Response Procedure to incorporate the administrative restrictions for allowable plant operating power level based on the actual LEFM status.

- Revision of the ERCS Thermal Power Monitor and the (secondary calorimetric) CALM Programs to adjust the allowable licensed thermal power values used in these programs.

- Revision to plant procedures and alarm responses for inclusion in the Operator Training Program

As part of the MUR-PU implementation modifications, current alarms will be evaluated and recalibrated as necessary to reflect small process condition changes in some balance of plant (BOP) systems. The licensee states in its submittal that no changes to or deletions of existing alarms or current setpoints are anticipated.

The NRC staff reviewed the licensee's evaluation and proposed changes to the control room. The staff concludes that the proposed changes do not present any adverse effects to the operators' functions in the control room. The licensee committed to making all necessary modifications to the control room and providing training on these changes prior to MUR-PU implementation. The NRC staff finds that the statements provided by the licensee are in conformance with Sections VII.2.B and VII.3 of Attachment 1 to RIS 2002-03.

3.1.2.4 Control Room Plant Reference Simulator and Operator Training Program

The licensee stated in its submittal that there will be changes to the simulator calorimetric and thermal power monitor programs with the new administrative power limits based on LEFM status. In addition, other core power dependent simulator programs such as Xenon, NIS Power, and Boron Concentration will be revised to reflect the revised core power of 1677 MWt.

The licensee stated in its submittal that as part of the MUR-PU implementation modification, required changes to plant procedures and alarm responses will be included in the Operator Training Program. Training will be provided on the allowable at-power administrative limits and new Technical Requirements Manual (TRM) governing LEFM out-of-service time prior to MUR-PU implementation.

The NRC staff has reviewed the licensee's proposed changes to operator training and the plant simulator related to the MUR-PU. The NRC staff concludes that the changes do not present any adverse effects on the plant simulator or the operator training program. The licensee will make modifications to the plant simulator and incorporate changes into the operator training program prior to MUR-PU implementation. The NRC staff finds that the statements provided by the licensee are in conformance with Sections VII.2.C, VII.2.D, and VII.3 of Attachment 1 to RIS 2002-03.

3.1.3 Conclusion

The NRC staff reviewed the licensee's proposed human factors changes and concludes that the licensee has adequately considered the impact of the proposed MUR-PU on operator actions, procedures, EOPs and AOPs, control room components, the plant simulator and operator training programs for the PINGP.

3.2 Dose Consequences Analysis

3.2.1 Regulatory Evaluation

RIS 2002-03 recommends that to improve efficiency of the NRC staff's review, licensees requesting an MUR-PU should identify existing design-basis accident (DBA) analyses of record that bound plant operation at the proposed uprated power level. For any existing DBA analyses of record that do not bound the proposed uprated power level, the licensee should provide a detailed discussion of the reanalysis.

This safety evaluation (SE) documents the NRC staff's review of the impact of the proposed changes on analyzed DBA radiological consequences. In PINGP Amendments 166 and 156 to Units 1 and 2, respectively, which were issued on September 10, 2004 (ADAMS Accession No. ML0424305040), the NRC approved implementation of the alternative source term for fuel handling accidents (FHA) in accordance with 10 CFR 50.67, and following the guidance provided in applicable sections of Regulatory Guide (RG) 1.183, "Alternative Radiological

Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors." Therefore, the staff conducted this evaluation to verify that the results of the licensee's FHA DBA radiological dose consequence analyses continue to meet the dose acceptance criteria given in 10 CFR 50.67 for offsite doses and 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 19 (or equivalent for plants licensed before the GDC were in existence) with respect to control room habitability.

Following the guidance provided in applicable sections of RG 1.195, "Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors," the staff conducted an evaluation to verify that the results of the licensee's non-FHA DBA radiological dose consequence analyses continue to meet the dose acceptance criteria given in 10 CFR 100.11 for offsite doses and 10 CFR Part 50, Appendix A, GDC 19 (or equivalent for plants licensed before the GDC were in existence) with respect to control room habitability. The staff utilized the regulatory guidance provided in applicable sections of RG 1.183, RG 1.195, NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (SRP), Section 6.4, for control room habitability, and PINGP Updated Safety Analysis Report (USAR) Chapter 14, for DBAs, in performing this review.

3.2.2 Technical Evaluation

The NRC staff reviewed the regulatory and technical analyses performed by the licensee in support of its proposed MUR-PU license amendment, as they relate to the radiological consequences of DBA analyses. Information regarding these analyses was provided by the licensee in Enclosure 2 to the December 28, 2009 application. The findings of this SE are based on the descriptions and results of the licensee's analyses and other supporting information docketed by the licensee.

The NRC staff reviewed the impact of the proposed 1.64 percent MUR-PU on DBA radiological consequence analyses, as documented in Chapter 14 of the USAR. The NRC staff reviewed all the accidents that have the potential for a significant dose consequence. The specific DBA analyses that were reviewed were as follows:

- Loss-of-Coolant Accident (LOCA)
- Locked Reactor Coolant Pump (RCP)/Locked Pump Rotor
- Fuel-Handling Accident (FHA)
- Main Steam Line Break (MSLB)
- Steam Generator Tube Rupture (SGTR)
- Control Rod Ejection Accident (CREA)/Uncontrolled Rod Cluster Control Assembly (RCCA) Withdrawal at Power

In the LAR submittal, the licensee stated that the current DBA dose analyses of record for the PINGP associated with LOCA, RCP, FHA, MSLB, and CREA, which depend on core power level, were performed at 1683 MWt, or 102 percent of the currently licensed thermal power of 1650 MWt. The current DBA dose analysis of record for SGTR was performed at 1721.4 MWt, or 104 percent of the currently licensed thermal power of 1650 MWt. Therefore, the current licensing basis (CLB) dose consequence analyses remain bounding at the proposed MUR uprated power level of 1677 MWt with a margin that is within the assumed uncertainty associated with advanced flow measurement techniques, including use of the Caldon LEFM CheckPlus™ system credited by the licensee.

The licensee has accounted for the potential for an increase in measurement uncertainty should the LEFM system experience operational limitations. In the event the LEFM internal checking system detects an alarm condition, the LEFM status will change to "ALERT" or "Fail" and the licensee will reduce power to ensure that the CLB dose consequence analyses remain bounding.

Using the licensing basis documentation as contained in the current PINGP USAR, in addition to information in the December 28, 2009, LAR, the NRC staff verified that the existing PINGP USAR Chapter 14 radiological analyses source term and release assumptions bound the conditions for the proposed 1.64 percent power uprate to 1677 MWt, considering the higher accuracy of the proposed FW flow measurement instrumentation.

3.2.3 Conclusion

As described above, the NRC staff reviewed the assumptions, inputs, and methods used by the licensee to assess the radiological consequences of the postulated DBA dose consequence analyses at the proposed uprated power level. The staff finds that the licensee will continue to meet the applicable dose limits following implementation of the proposed 1.64 percent MUR-PU. The staff further finds reasonable assurance that PINGP, as modified by this approved license amendment, will continue to provide sufficient safety margins, with adequate defense-in-depth, to address unanticipated events and to compensate for uncertainties in accident progression, analysis assumptions, and input parameters. Therefore, the NRC staff concludes that the proposed license amendment is acceptable with respect to the radiological dose consequences of the DBAs.

3.3 Fire Protection

3.3.1 Regulatory Evaluation

The purpose of the fire protection program is to provide assurance, through a defense-in-depth design, that a fire will not prevent the performance of necessary plant safe-shutdown functions nor will it significantly increase the risk of radioactive releases to the environment. The NRC staff's review focused on the effects of the increased decay heat due to the MUR-PU on the plant's safe-shutdown analysis to ensure that structures, systems, and components (SSCs) required for the safe-shutdown of the plant are protected from the effects of the fire and will continue to be able to achieve and maintain safe-shutdown following a fire. The NRC's acceptance criteria for the fire protection program are based on: (1) 10 CFR 50.48, "Fire protection," insofar as it requires the development of a fire protection program to ensure, among other things, the capability to safely shutdown the plant; (2) GDC 3 of Appendix A to 10 CFR Part 50, insofar as it requires that [a] SSCs important to safety be designed and located to minimize the probability and effect of fires, [b] noncombustible and heat resistant materials be used, and [c] fire detection and suppression systems be provided and designed to minimize the adverse effects of fires on SSCs important to safety; and (3) GDC 5 of Appendix A to 10 CFR Part 50, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions.

A revision to 10 CFR Part 50, Appendix K, effective July 31, 2000, allowed licensees to use a power uncertainty of less than 2 percent in design basis LOCA (DBLOCA) analyses, based on the use of state-of-the-art FW flow measurement devices that provide for a more accurate calculation of reactor power. Appendix K to 10 CFR Part 50 did not originally require that the reactor power measurement uncertainty be determined, but instead required a 2 percent margin. The revision allows licensees to justify a smaller margin for power measurement uncertainty based on power level instrumentation error. This type of change is also commonly referred to as an MUR-PU.

RIS 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications," January 31, 2002, Attachment 1, Sections II and III (ADAMS Accession No. ML013530183) recommends that, to improve the efficiency of the staff's review, licensees requesting an MUR-PU should identify current accident and transient analyses of record that bound plant operation at the proposed uprated power level. For any design basis accident for which the existing analyses of record do not bound the proposed uprated power level, the licensee should provide a detailed discussion of the reanalysis.

3.3.2 Technical Evaluation

The licensee developed the LAR consistent with the guidelines in RIS 2002-03. In the LAR, the licensee re-evaluated the applicable SSCs and safety analyses at the proposed MUR core power level of 1677 MWt against the previously analyzed core power level of 1650 MWt.

The NRC staff reviewed LAR Enclosure 2, Section II.2.26, and "10.3.1 – Plant Fire Protection Program (Appendix R). The staff also reviewed the licensee's commitment to 10 CFR 50.48, "Fire protection" (i.e., approved fire protection program). The review covered the impact of the proposed MUR-PU on the results of the safe-shutdown fire analysis as noted in RIS 2002-03, Attachment 1, Sections II and III. The review focused on the effects of the MUR-PU on the post-fire safe-shutdown capability and the increase in decay heat generation following plant trips.

In a letter dated March 17, 2010, the staff issued a request for additional information (RAI). The staff noted that Enclosure 2 to the LAR, Section II.2.26, "10.3.1 – Plant Fire Protection Program (Appendix R)," states that *"...Evaluations conclude that the current safe shutdown analyses use an analytical core power of 1683 MWt (or higher) and, as such, the MUR Power Uprate has no effect on the plant equipment and systems credited with achieving safe shutdown. Likewise, evaluations also conclude that MUR PU has no impact on Appendix R manual action constraints..."* The staff requested the licensee to verify that (1) the MUR-PU will not require any new operator actions, and (2) any effects from additional heat in the plant environment from the increased power will not interfere with existing operator manual actions being performed at their designated time and place.

In a letter to the NRC dated April 19, 2010, the licensee provided additional information outlined in this paragraph in response to the above RAI. In its response, the licensee stated that the current safe-shutdown analyses use a core power of 1683 MWt or higher, and the MUR-PU has no effect on plant systems and equipment relied upon to achieve safe-shutdown conditions. This evaluation considered the impacts of the MUR-PU on system temperatures, operating environments, and operator actions. Further, the licensee stated that the PINGP Appendix R Fire Protection Program analyses and procedures are based on achieving safe-shutdown from analyzed core power levels of 1683 MWt, and bound the MUR-PU conditions. The current operator actions in existing plant procedures associated with an Appendix R condition will remain applicable for operation up to an uprated core power of 1677 MWt. Operator actions included in the current program are not changed and no new operator actions are required.

The licensee's response satisfactorily addresses the NRC staff's concerns, and this RAI issue is considered resolved based on the following: For the MUR-PU condition, the licensee reviewed its systems to achieve and maintain post-fire safe-shutdown condition at core power level of 1683 MWt. The results demonstrate that existing plant procedures associated with an Appendix R condition do not impact safe-shutdown for the proposed MUR-PU to 1677 MWt, which is still 0.36 percent below the analyzed power level. Further, the proposed MUR-PU does not impact the previous operator manual actions in the fire safe-shutdown analysis and (1) no new operator manual actions have been identified, (2) the proposed MUR-PU does not impact current operator manual actions, and (3) the FW temperature increases only slightly from 434.9 degrees Fahrenheit (°F) to 437.5 °F for the proposed MUR-PU, which does not create any adverse environmental condition that would impact performance of existing operator manual actions at their designated time and place.

In a letter dated March 17, 2010, the NRC staff requested the licensee to verify whether the PINGP Units 1 and 2 credit aspects of their fire protection systems for other than fire protection activities (e.g., utilizing the fire water pumps and water supply as backup cooling or inventory for nonprimary reactor systems). If the PINGP Units 1 and 2 credit their fire protection systems for other than fire protection activities, the MUR-PU LAR should identify the specific situations and discuss to what extent, if any, the MUR-PU affects these "nonfire-protection" aspects of the plant fire protection system. If the PINGP Units 1 and 2 do not take such credit, the staff requested that the licensee verify this as well.

In a letter to the NRC dated April 19, 2010, the licensee provided additional information outlined in this paragraph in response to the staff's RAI. In its response the licensee stated that the PINGP USAR accident analyses in Chapter 14 do not take credit for the fire protection system for functions other than fire protection. The PINGP USAR identifies that the fire protection system can supply makeup water to the spent fuel pool, in addition to fire protection activities. In emergency conditions that are beyond the design basis of the plant, PINGP damage mitigation procedure describes provisions to use the fire protection system for the following functions: (1) Provide makeup to the spent fuel pool, as stated above, (2) Inject water into the steam generators (SGs) and (3) Inject water into the reactor coolant system (RCS), using a specially constructed connecting piece and a specified flush connection. The capabilities to inject fire protection water into the SGs and RCS are not considered design basis functions and are not affected by the MUR-PU.

The licensee's response satisfactorily addresses the staff's concerns, and this RAI issue is considered resolved based on the following: The fire protection water is credited for three activities other than fire protection. In the first scenario, the fire protection water is utilized to add inventory to the spent fuel pool in the event spent fuel cooling is lost and boiling occurs. In the second scenario, the fire protection water injects into the SGs. In the third scenario, the fire protection water injects into the RCS. The licensee analyzed spent fuel cooling requirements for removal of decay heat associated with 102 percent of the current operating power and concluded this first scenario not to be affected by the MUR-PU conditions. For the other two scenarios, the licensee does not consider injection of fire protection water into SG or RCS as design basis functions and concluded that these two scenarios are not affected by the MUR-PU conditions. Therefore, the staff finds the response to the RAI acceptable because the licensee's analysis concluded that all three functions of nonfire suppression uses of fire protection water are not affected by the proposed MUR-PU.

Based on the licensee's fire-related safe-shutdown assessment and responses to the RAIs, the staff concludes that the licensee has adequately accounted for the effects of the 1.64 percent increase in decay heat on the ability of the required systems to achieve and maintain safe-shutdown conditions. The staff finds this aspect of the capability of the associated SSCs to perform their design basis functions at an increased core power level of 1677 MWt acceptable with respect to fire protection.

3.3.3 Conclusion

The NRC staff has reviewed the licensee's fire-related safe-shutdown assessment and concludes that the proposed MUR-PU will not have a significant impact on the fire protection program or post-fire safe shutdown capability. The staff finds this aspect of the capability of associated SSCs to continue to perform their design basis functions at an increased core power level of 1677 MWt to be acceptable with respect to fire protection.

3.4 Chemical Engineering

3.4.1 Flow Accelerated Corrosion

3.4.1.1 Regulatory Evaluation

Flow-accelerated corrosion (FAC) is a corrosion mechanism occurring in carbon steel components exposed to flowing single- or two-phase water. Components made from stainless steel are immune to FAC, and FAC is significantly reduced in components containing small amounts of chromium or molybdenum. Material loss rates due to FAC depend on flow velocity, fluid temperature, steam quality, oxygen content, and pH. During plant operation, flexibility to control these parameters to minimize FAC is limited. Therefore, loss of material by FAC is likely to occur. The NRC staff reviewed the effects of the proposed MUR-PU on FAC and the adequacy of the licensee's FAC program. The intent of the FAC program is to predict the rate of loss so that repair or replacement of damaged components can be made before they reach critical thickness. The licensee's FAC program is based on NRC Generic Letter (GL) 89-08, "Erosion/Corrosion - Induced Pipe Wall Thinning," May 1989, NRC GL 89-13, "Service Water System Problems Affecting Safety-Related Equipment," July 1989, and the guidelines in Electric Power Research Institute (EPRI) Report NSAC-202L-R3, "Recommendations for an Effective Flow-Accelerated Corrosion Program," April 1999. The program consists of predicting loss of material using the EPRI CHECWORKS FAC monitoring computer code, visual inspection, and volumetric examination of the affected components. The NRC's acceptance criteria are based on the structural evaluation of the minimum acceptable wall thickness for the components undergoing degradation by FAC.

3.4.1.2 Technical Evaluation

The licensee utilizes the EPRI CHECWORKS software to model susceptible safety-related and non-safety related systems and components at PINGP. The FAC in carbon steel piping is programmatically controlled by using plant procedures which do not require modification as a result of the MUR-PU. The licensee stated that a recent sensitivity study was conducted and indicated that the CHECWORKS model provides a conservative prediction of actual wear rates using best-estimate plant parameters. The licensee also stated that the methods, guidelines, acceptable criteria, and procedures for the FAC program will continue to adequately ensure plant and personnel safety following the MUR-PU.

In support of the MUR-PU, the licensee stated that the CHECWORKS model will be updated to incorporate the changes associated with the power uprate. The associated changes include: increased moisture carryover, operating temperature, pressure and flowrate. The licensee stated that a sensitivity study using the current PINGP CHECWORKS model with corresponding best-estimate heat balance information was performed on a representative sample of susceptible components. The licensee provided a table that compared FAC information for current power conditions to MUR-PU conditions. The results indicated that the impact by the MUR-PU on wear-rate predictions is minor. The NRC staff concurs with this assessment.

The licensee indicated that the CHECWORKS models will be revised to include the changes in moisture carryover, temperature, pressure and flowrate, as a result of the MUR-PU implementation. In addition, it was stated that the results of the upgrade models will be factored into future surveillance and piping repair plans, as applicable. The NRC staff finds this acceptable. The staff also reviewed the licensee's evaluation and confirmed that the applicable regulatory guidance was followed.

The NRC staff has also verified the calculations in the application. The licensee has demonstrated that the FAC program is adequate for managing the potential effects on the piping components susceptible to FAC. The staff's acceptance of the licensee's use of CHECWORKS can be found in NUREG-1766, "Safety Evaluation Report Related to the License Renewal of North Anna Power Station, Units 1 and 2, and Surry Power Station, Units 1 and 2," December 2002. The NRC staff concludes that the FAC program is adequate in predicting the rate of material loss.

3.4.1.3 Conclusion

The staff has reviewed the licensee's evaluation of the effect of the proposed MUR-PU on the FAC analysis for the plant and concludes that the licensee has adequately addressed the impact of changes in the plant operating conditions on the FAC analysis. The licensee has demonstrated that the updated analyses will predict the loss of material by FAC, and allow for timely repair or replacement of degraded components following implementation of the proposed MUR-PU. Therefore, the NRC staff finds the proposed MUR-PU is acceptable with respect to FAC.

3.4.2 Protective Coating Systems (Paints) – Organic Materials

3.4.2.1 Regulatory Evaluation

Protective coating systems (paints) protect the surfaces of facilities and equipment from corrosion and radionuclide contamination. Protective coating systems also provide wear protection during plant operation and maintenance activities. The NRC acceptance criteria for protective coating systems are based on (1) 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," and (2) RG 1.54, Revision 1, "Service Level I, II, and III Protective Coatings Applied to Nuclear Power Plants," issued in July 2000. Specific review criteria are contained in SRP Section 6.1.2, "Protective Coating Systems (Paints) – Organic Materials Review Responsibilities." The NRC staff review focused on protective coating systems used inside containment, including the coating's suitability for, and stability under, DBLOCA conditions, considering radiation and chemical effects.

3.4.2.2 Technical Evaluation

The staff requested that the licensee provide additional details on protective coating systems used in the containment and whether they remain qualified under MUR-PU conditions. The licensee responded to the request in the April 19, 2010, letter. In its response, the licensee stated that PINGP maintains protective coatings for interior surfaces and permanently installed equipment in the reactor building using the Safety Related Coatings program. The Safety Related Coatings program requires that new coating systems for areas inside containment be qualified and tested in accordance with American National Standard Institute (ANSI) N101.2, "Protective Coatings (Paints) for Light Water Nuclear Reactor Containment Facilities." It was further reported that those coatings that are determined not to be qualified are periodically inspected. The NRC staff finds this acceptable since ANSI N101.2 is consistent with NRC RG 1.54, Revision 1.

The new coating systems described in the Safety Related Coatings program specify qualification testing to a prescribed temperature, pressure and radiation level. The licensee stated that the temperature, pressure, and radiation exposure values in accordance with the post-LOCA temperature, pressure, and radiation profile contained in PINGP calculations are 274.4 °F, 45.5 pounds per square inch gauge (psig), and $2x10^8$ Rads, respectively. The temperature and pressure qualification requirements are not changed by the MUR-PU. However, the radiation doses (normal + accident) will increase by approximately 1.64 percent after implementation of the proposed MUR-PU. The licensee stated that the increased radiation level is reflected in the Environmental Specification H8-H analysis.

In addition, the licensee stated that the test data for coatings used at PINGP indicate qualification testing that includes a peak temperature of 323 °F, a peak pressure of 48.4 psig, and a radiation exposure of $3x10^8$ Rads. These values bound the anticipated MUR-PU post-LOCA conditions and no changes are required for the temperature, pressure, and radiation qualification requirements. The NRC staff finds this assessment acceptable.

The NRC staff has reviewed the licensee's evaluation and confirmed that the applicable regulatory guidance was followed. The NRC staff concludes that the coatings will not be adversely impacted by the MUR-PU and that temperature, pressure, and radiation limits under power uprate conditions continue to be bounded by the conditions to which the coatings were qualified.

3.4.2.3 Conclusion

The staff concludes that the impact of changes in conditions following a DBLOCA and the effects on the protective coatings had been appropriately addressed. The staff further concludes that protective coatings will continue to be acceptable following implementation of the proposed MUR-PU. Specifically, the protective coating will continue to meet the requirements of 10 CFR Part 50, Appendix B. Therefore, the NRC staff finds the proposed MUR-PU acceptable with respect to protective coating systems.

3.4.3 Steam Generator Program

3.4.3.1 Regulatory Evaluation

SG tubes constitute a significant part of the reactor coolant pressure boundary (RCPB). SG tube inservice inspections provide a means for assessing the structural and leak-tight integrity of SG tubes through periodic inspection and testing of critical areas and features of the tubes. The NRC staff reviewed the effects of changes in operating parameters (e.g., pressure, temperature, and flow velocities) resulting from the proposed power uprate on the design and operation of the SGs. Specifically, the staff evaluated whether changes to these parameters continue to be bounded by those considered in the plant design and licensing basis (i.e., the TS plugging limits).

3.4.3.2 Technical Evaluation

Unit 1 of the PINGP has two Framatome/AREVA model 56/19 replacement recirculating SGs. Each SG has 4868 thermally-treated Alloy 690 tubes. Each tube has an outside diameter of 0.750 inches and a nominal wall thickness of 0.043 inches. The tubes were hydraulically expanded at each end for the full depth of the tubesheet. The tubes are supported by support plates and anti-vibration bars (AVBs). All tube support plates are constructed from Type 410 stainless steel.

Unit 2 of the PINGP has two Westinghouse model 51 SGs. Each SG contains 3388 mill-annealed Alloy 600 tubes. Each tube has an outside diameter of 0.875 inches and a nominal wall thickness of 0.050 inches. The tubes were roll-expanded into the tubesheet at both ends for approximately 2.75 inches (i.e., they are expanded for only a fraction of the tubesheet thickness and are considered partial depth hard-rolled tubes). The tubes are supported by a number of carbon steel tube support plates. The original AVBs were removed and replaced.

The licensee stated that the PINGP USAR describes the analysis methodology for SGTR, which is derived from the original plant Final Safety Analysis Report. In addition, the licensee stated that a core power of 1721.4 MWt was used in the USAR, which is a value that is conservatively larger than the analytical core power of 1683 MWt associated with the MUR-PU. As such, the current analysis-of-record found in the USAR bounds the SGTR accident. The NRC staff finds the licensee's evaluation acceptable.

The licensee stated that the increased flow conditions within the SG steam drum under MUR-PU conditions are expected to initiate or accelerate material loss in the carbon steel steam drum components. The periodic SG inspections will continue to be utilized to detect degradation that may occur. The NRC staff finds this acceptable.

The licensee performed an analysis on the Unit 2 SGs and reported that the fluid-elastic stability ratios will remain below the allowable limit of 1.0 for most of the tube bundle assembly. The licensee indicated that the fluid-elastic stability ratio for the peripheral tubes in rows 6 through 8 and the interior U-bend region tubes in rows 10 and 11 did not remain below the allowable limit of 1.0. Although the tubes had ratios above 1, the licensee stated that the turbulence-induced displacements and tube wear predictions are below the respective allowable values. The licensee determined that under MUR-PU conditions, the predicted tube wear for the entirety of plant operation is below the allowable limit of 40 percent of the tube wall thickness.

The licensee calculated the stress analysis due to flow-induced vibration (FIV) on the Unit 2 SGs and determined that the maximum peak bending stress is below the 1.5 mean stress (Sm) limits of the Alloy 600 tube material of 34.96 ksi. Fatigue degradation from FIV is not anticipated and the fatigue usage factor for the tubes was stated to be less than 1.0. The staff finds this acceptable.

The licensee also stated that the analyses that support the Unit 1 SGs were reviewed to determine the impact of the MUR-PU. It was stated that the MUR-PU conditions have no impact on the Unit 1 SGs tube vibration because the current analyses bound the MUR-PU conditions. The staff finds the licensee's evaluation acceptable.

The NRC staff has reviewed the licensee's evaluation and calculation results found in the amendment request and have confirmed that the applicable regulatory guidance was followed. Based on the licensee's evaluation, the staff concludes that the proposed MUR-PU will introduce only insignificant changes as it relates to tube stresses resulting from tube vibration, cumulative fatigue usage factors, and potential tube wear. Therefore, the MUR-PU will not affect the satisfactory performance in maintaining SG tube integrity.

3.4.3.3 Conclusion

The NRC staff reviewed the licensee's evaluation of the effect of the proposed MUR-PU on SG tube integrity and concludes that the licensee has adequately assessed the continued acceptability of the plant's TSs. Specifically, the licensee has an ongoing periodic inspection program, which will continue to be utilized to detect degradation that may occur. Therefore, the staff finds the proposed MUR-PU acceptable with respect to the SG program.

3.4.4 Chemical and Volume Control System

3.4.4.1 Regulatory Evaluation

The chemical and volume control system (CVCS) provides a means for (1) maintaining water inventory and quality in the RCS, (2) supplying seal-water flow to the reactor coolant pumps and pressurizer auxiliary spray, (3) controlling the boron neutron absorber concentration in the reactor coolant, (4) controlling the primary-water chemistry and reducing coolant radioactivity level, and (5) supplying recycled coolant for demineralized water makeup for normal operation and high-pressure injection flow to the ECCS in the event of postulated accidents. The staff reviewed the safety-related functional performance characteristics of CVCS components. The NRC's acceptance criteria are based on (1) GDC-14, "Reactor Coolant Pressure Boundary (RCPB)," as it requires that the RCPB be designed to have an extremely low probability of abnormal leakage, of rapidly propagating fracture, and of gross rupture; and (2) GDC-29, "Protection Against Anticipated Operational Occurrences," as it requires that the reactivity control systems be designed to assure an extremely high probability of accomplishing their functions in the event of condenser in-leakage or primary-to-secondary leakage. Specific review criteria are contained in SRP Section 9.3.4, "Chemical and Volume Control System (PWR)."

3.4.4.2 Technical Evaluation

The licensee performed a transient analysis using a Westinghouse methodology for USAR Chapter 14, which included information on CVCS malfunction (i.e., boron dilution). The analysis and methodology were performed for a 14 x 14 optimized fuel assembly (OFA) configuration in the PINGP reactor core and were approved by the NRC staff in License Amendments 162 and 153 for Units 1 and 2, respectively.

The licensee stated that a revised analysis was conducted under 10 CFR 50.59, "Changes, Tests, and Experiments," using staff-approved methods (i.e., License Amendments 162 and 153) demonstrated acceptable for the OFA-only cores. The licensee stated that the transient analysis is bounded by the MUR-PU nominal core power of 1677 MWt. Specifically, the evaluation included revised core power and power measurement uncertainty for transient, safety, and accident analysis to 1683 MWt, including required uncertainties. The NRC staff finds this acceptable.

The NRC staff has reviewed the licensee's evaluation and has confirmed that the applicable regulatory guidance was followed. The licensee has demonstrated that the CVCS will continue to maintain RCS inventory and water chemistry. The staff confirms that the CVCS will continue to meet system design requirements and that no new design transients will be created at MUR-PU conditions.

3.4.4.3 Conclusion

The NRC staff reviewed the licensee's evaluation of the effects of the proposed MUR-PU on the CVCS and concludes that the licensee has adequately addressed changes in the temperature of the reactor coolant and its effects on the CVCS. The staff further concludes that the licensee has demonstrated that the CVCS will continue to be acceptable and will continue to meet the requirements of GDC-14 and GDC-29 following implementation of the proposed MUR-PU.

3.4.5 Steam Generator Blowdown System (SGBS)

3.4.5.1 Regulatory Evaluation

Control of secondary-side water chemistry is important for preventing degradation of SG tubes. The SGBS provides a means for removing SG secondary-side impurities, and thus assists in maintaining acceptable secondary-side water chemistry in the SGs. The design basis of the SGBS includes consideration of expected design flows for all modes of operation. The NRC staff reviewed the ability of the SGBS to remove particulate and dissolved impurities from the SG secondary-side during normal operation, including condenser in-leakage and primary-to-secondary leakage. The NRC's acceptance criteria for the SGBS are based on GDC 14, which requires the RCPB to be designed to have an extremely low probability of abnormal leakage, of rapidly propagating fracture, and of gross rupture. Specific review criteria are contained in SRP Section 10.4.8, "Steam Generator Blowdown System (PWR)."

3.4.5.2 Technical Evaluation

The licensee stated that the SGBS flow rates required during plant operation are based on chemistry control and tubesheet sweep requirements to control the buildup of solids. The SGBS flow rate used to control chemistry and solid buildup in the SGs is tied to allowable condenser in-leakage, total dissolved solids in the plant circulating water system, and allowable primary-to-secondary leakage. The licensee stated that the MUR-PU conditions are bounded by the CLB. The licensee further stated that, based on no-load steam pressure and the minimum full-load steam pressure, the MUR-PU will not impact SGBS flow control. The NRC staff finds this acceptable.

The NRC staff has reviewed the licensee's evaluation and confirmed that the applicable regulatory guidance was followed. The licensee has demonstrated that the SGBS is adequate for maintaining secondary-side water chemistry within industry guidelines for maintenance of controlled corrosion rates in secondary system components. The staff concludes that the SGBS will continue to meet system design requirements at MUR-PU conditions.

3.4.5.3 Conclusion

The NRC staff reviewed the licensee's evaluation of the effects of the proposed MUR-PU on the SGBS and concludes that the licensee has adequately addressed changes in system flow and impurity levels and the effect on the SGBS. The staff concludes that the licensee has demonstrated that the SGBS will continue to be acceptable and will continue to meet the requirements of GDC 14 following implementation of the proposed MUR-PU. Therefore, the staff finds the proposed MUR-PU acceptable with respect to the SGBS.

3.4.6 Overall Chemical Engineering Conclusion

In the areas of SGs and chemical engineering, the NRC staff concludes that the licensee has adequately addressed (1) the changes in the plant operating conditions for the FAC program, (2) the effects on protective coatings, (3) the changes in the SG operating parameters, the effects on the SGs and the determination that the SG tube integrity will continue to be maintained, (4) the changes of the reactor coolant and their effect on the CVCS, and (5) the changes in the system flow and impurity levels, and their effects on the SGBS.

3.5 Mechanical and Civil Engineering

3.5.1 Regulatory Evaluation

The NRC staff's review in the areas of mechanical and civil engineering covers the structural and pressure boundary integrity of the nuclear steam supply system (NSSS) and the BOP systems and components. Specifically, this review focuses on the impact of the proposed MUR-PU on the structural integrity of the (1) NSSS piping, components, and supports; (2) BOP piping, components, and supports; (3) the reactor pressure vessel (RPV) and its supports; (4) the pressure retaining portions of the control rod drive mechanisms (CRDMs); (5) the pressure retaining portions of the original SGs and the replacement SGs (RSGs) for Unit 1 and their supports; (6) the pressure retaining portions of the RCPs; (7) the pressurizer and its supports, and (8) the reactor vessel internals (RVIs) and core support structures. Technical areas covered by this review include stresses, fatigue and corresponding cumulative usage factors (CUFs), FIV, high-energy line break (HELB) locations and any corresponding jet impingement and thrust forces. The NRC staff's review focused on verifying that the licensee has provided reasonable assurance of the structural and functional integrity of the aforementioned piping systems, components, component internals and their supports under normal and transient loadings, including those due to postulated accidents and natural phenomena, such as earthquakes.

The PINGP received construction permits prior to May 21, 1971, which is the date the GDC in Appendix A of 10 CFR Part 50 became effective. Section 1.5, "General Design Criteria," of the PINGP USAR states that PINGP was designed and constructed to comply with the licensee's understanding of the intent of the Atomic Energy Commission General Design Criteria (AEC GDC or proposed GDC) for Nuclear Power Plant Construction Permits, as proposed on July 10, 1967. As such, the NRC staff's evaluation considered 10 CFR 50.55a and AEC GDC 1, 2, 6, 9, 33, and 34, which are described in Section 1.5 of the PINGP USAR. However, as noted in the PINGP USAR, the AEC staff's Safety Evaluation Report (SER) acknowledges that the AEC staff assessed the PINGP Final Safety Analysis Report (FSAR) against the Appendix A design criteria and "... [were] satisfied that the plant design generally conforms to the intent of these criteria."

As indicated above, the acceptance criteria are based on continued conformance with the requirements of the following regulations, which are applicable to the civil and mechanical engineering review areas: (1) 10 CFR 50.55a, and AEC GDC 1 as they relate to structures and components being designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed; (2) AEC GDC 2 as it relates to structures and components important to safety being designed to withstand the effects of earthquakes combined with the effects of normal or accident conditions; (3) AEC GDC 6 as it relates to the reactor core being designed with appropriate margin to assure that acceptable fuel damage limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences; (4) AEC GDC 9 as it relates to the reactor coolant system (RCS) being designed with sufficient margin to ensure that the design conditions are not exceeded; (5) AEC GDC 33 as it relates to structures and components important to safety being designed to accommodate the effects of, and to be compatible with, the environmental conditions of normal and accident conditions and these structures and components being appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids and (6) AEC GDC 34 as it relates to the reactor coolant pressure boundary being designed, fabricated, erected, and tested to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

The primary guidance used by NSPM and other licensees for LARs involving MUR-PUs is found in RIS 2002-03, which provides licensees with a guideline for organizing LAR submittals for MUR-PUs. Section IV of RIS 2002-03, "Mechanical/Structural/Material Component Integrity and Design," provides information to licensees on the scope and detail of the information that should be submitted to the NRC staff regarding the impact a MUR-PU has on the aforementioned components and technical areas.

3.5.2 Technical Evaluation

The NRC staff's technical review focused primarily on the effects of the power uprate on the structural and pressure boundary integrity of piping systems and components, their supports, the reactor vessel and associated internal components, the pressure retaining portions of the CRDMs and the BOP and NSSS interface piping systems. The proposed 1.64 percent power uprate will increase the rated thermal power level from 1650 MWt to 1677 MWt at PINGP. The licensee notes in Enclosure 1 of Reference 1 that the analyses comprising the CLB are based on a core power level of 1683 MWt or higher.

Enclosure 2, Table IV.1 B.4, of Reference 1 shows the pertinent temperatures, pressures, and flow rates for the current and uprated conditions. For the purposes of the structural and mechanical evaluations performed in support of the proposed power uprate, NSPM evaluated the effects of the proposed MUR-PU on various SSCs at a bounding power level of 1690 MWt. This power level corresponds to the proposed level following the uprate (1677 MWt) plus the revised uncertainty of 0.36 percent (1683 MWt), with an additional 7 MWt to account for RCP heat input. As shown in the table, there is no change in the RCS operating pressure (2250 psia). At full power, the hot-leg temperature increases from 592.1 °F to 592.6 °F while the cold-leg temperature decreases from 527.9 °F to 527.4 °F. The SG pressures decrease from 772 psia to 765 psia for Unit 1 and from 719 psia to 712 psia for Unit 2. The steam flow increases from 7.2 million pounds per hour (Mlbm/hr) and 7.19 Mlbm/hr to 7.36 Mlbm/hr for Units 1 and 2, respectively. The FW temperature increases from 434.9 °F to 437.5 °F. The design parameters for the primary system at PINGP are found in Chapter 4 of the PINGP USAR. The RCS components are designed to 650 °F (except the pressurizer and pressurizer surge line, which are designed to 680 °F) and 2485 psig. Chapter 11 of the PINGP USAR provides the design basis information for the secondary side systems, including the main steam (MS) system and the FW and condensate system.

3.5.2.1 Reactor Pressure Vessel

The licensee evaluated the effects of the proposed power uprate on the structural integrity of the RPV, including its nozzles, in Section IV.1.A.i of Enclosure 2 to Reference 1. These evaluations included a review of any possible load increases on these SSCs directly associated with the effects of the proposed MUR-PU, including revised transient loads due to the power uprate. Table IV.1.D.1 of Enclosure 2 provides the design codes of record for the SSCs evaluated in support of the MUR-PU. The design code of record for the RPV is the American Society of

Mechanical Engineers (ASME) Boiler & Pressure Vessel (B&PV) Code, Section III, 1968 Edition with Winter 1968 Addenda. The code of record for the replacement reactor vessel closure heads (RVCHs) at PINGP is the ASME B&PV Code, Section III, 1998 Edition with 2000 Addenda. The RPV supports were designed in accordance with the Sixth Edition of the American Institute for Steel Construction (AISC) Manual for Steel Construction. The licensee compared the expected temperatures and pressures for the proposed power uprate condition against the analyses of record.

Based on its evaluations, the licensee confirmed that the RPV and its associated components described above will continue to meet the stress and fatigue design requirements of the design codes of record for these components following the implementation of the MUR-PU. In response to an NRC staff RAI regarding the fatigue evaluations performed for SSCs affected by the proposed MUR-PU, the licensee confirmed that all fatigue evaluations considered a 60-year plant life. The NRC staff considers this time period acceptable based on the licensee's request for a renewal of the PINGP operating licenses for an additional 20 years beyond the original licenses (Reference 5). The licensee also confirmed in its submittal that there is no change in RCS design or operating pressure, and the effects of operating temperature changes for the cold and hot legs are within the design limits. Also, no additional transients have been proposed as a result of the MUR-PU at the PINGP.

The NRC staff has reviewed the licensee's evaluations related to the structural integrity of the RPV and its associated components, including the nozzles. For the reasons set forth above, which demonstrate that the RPV will continue to meet its design basis acceptance criteria under the conditions of the proposed MUR power level, the NRC staff concludes that the licensee has adequately addressed the effects of the proposed MUR on these components. Based on the above, the NRC staff further concludes that the licensee has demonstrated that the RPV and its associated components will continue to meet the applicable regulatory requirements, described above, following implementation of the proposed MUR-PU. Therefore, the NRC staff finds the proposed power uprate acceptable with respect to the structural integrity of the RPV and its associated components.

3.5.2.2 Control Rod Drive Mechanisms

The licensee evaluated the effects of the proposed power uprate on the structural integrity of the CRDMs in Section IV.1.A.iii of Enclosure 2 to Reference 1. These evaluations included a review of any possible load increases on the pressure retaining portions of the CRDMs directly associated with the effects of the proposed MUR-PU, including revised transient loads due to the power uprate. Table IV.1.D.1 of Enclosure 2 provides the design codes of record for the SSCs evaluated in support of the MUR-PU. The design code of record for the pressure retaining portions of the CRDMs is the ASME B&PV Code, Section III, 1998 Edition with 2000 Addenda. Based on the location of the CRDMs, these components are affected primarily by the RCS pressure and the RCS hot leg temperature, of which only the hot leg temperature changes as a result of the power uprate. The licensee confirmed that the current analyses of record for the CRDMs remain bounding at the conditions following the MUR-PU implementation. Therefore, the stresses and fatigue usage factors at the proposed power level remain valid with respect to the design code of record acceptance criteria.

As previously indicated, in response to an NRC staff RAI regarding the fatigue evaluations performed for SSCs affected by the proposed MUR-PU, the licensee confirmed in Reference 2 that all fatigue evaluations considered a 60-year plant life. Also, no additional transients have been proposed.

The NRC staff has reviewed the licensee's evaluations related to the structural integrity of the pressure retaining portions of the CRDMs. For the reasons set forth above, which demonstrate that the CRDMs will continue to meet their design basis acceptance criteria under the conditions of the proposed MUR power level, the NRC staff concludes that the licensee has adequately addressed the effects of the proposed MUR-PU on these components. Based on the above, the NRC staff further concludes that the licensee has demonstrated that the CRDMs will continue to meet the applicable regulatory requirements following implementation of the proposed power uprate. Therefore, the NRC staff finds the proposed MUR-PU acceptable with respect to the structural integrity of the pressure retaining portions of the CRDMs.

3.5.2.3 Reactor Coolant Piping and Components

In support of the proposed MUR-PU at the PINGP, the licensee evaluated the various piping systems, components and supports that make up the RCS. This evaluation included the RCS main loop piping and corresponding branch piping, the original and replacement, primary and secondary side SG components, the pressure retaining portions of the RCPs, and the pressurizer and corresponding pressurizer surge line. The NRC staff's review of these SSCs is detailed in the following subsections.

3.5.2.3.1 Reactor Coolant System Piping and Supports

The licensee evaluated the effects of the proposed power uprate on the structural integrity of the RCS piping and supports in Section IV.1.A.iv of Enclosure 2 to Reference 1. These evaluations included a review of any possible load increases on the RCS piping and supports directly associated with the effects of the proposed MUR-PU, including revised transient loads due to the power uprate. Table IV.1.D.1 of Enclosure 2 provides the design codes of record for the SSCs evaluated in support of the MUR power uprate at PINGP. The RCS piping for Unit 1 was designed in accordance with the 1955 Edition of the American Standards Association (ASA) B31.1 Code for Pressure Piping. The RCS piping for Unit 2 was designed in accordance with the 1967 Edition of the USA Standard (USAS) B31.1 Code for Power Piping, with the exception of the pressurizer surge lines, which are discussed in Section 3.5.2.3.4 of this SE. The licensee reviewed the revised design conditions for impact on the existing design basis analyses for the reactor coolant piping and supports.

For the purposes of evaluating the effects of the MUR-PU on the RCS piping and supports, there is no change in RCS design or operating pressure, and the effects of the variance in operating temperature for the hot leg (increases at uprated conditions) and cold legs (decreases at uprated conditions) are well within the design limits. In Section IV.1.B.i of Enclosure 2, the licensee focused on the effects of the proposed power uprate on the RCS pipe stress analyses of record. It was noted in Section IV.1.B.i that the proposed power uprate does affect the pipe rupture and thermal loads imposed on the branch piping connected to the reactor coolant loop (RCL) piping. As such, the analyses performed in support of the proposed uprate resulted in revised faulted anchor movements for the branch piping connections. Subsequently, revised stresses were calculated for the affected branch piping and nozzles and then compared to the faulted condition code allowable values. The licensee confirmed that the MUR-PU conditions are bounded by the current licensing basis for the RCS piping and pipe supports and that all RCS piping and pipe support stresses remain within code allowable values under the MUR-PU conditions.

3.5.2.3.2 Steam Generator Components

The licensee evaluated the effects of the proposed power uprate on the structural integrity of critical primary and secondary side components of the SGs in Section IV.1.A.vi of Enclosure 2 to Reference 1. A discussion regarding the Unit 2 SG supports design is located in Section IV.1.A.iv of Enclosure 2 in the submittal. Table IV.1.D.1 of Enclosure 2 provides the design codes of record for the SSCs evaluated in support of the MUR-PU. The RSGs at PINGP Unit 1 were installed in 2004 and the SGs at PINGP Unit 2 are tentatively scheduled to be replaced in 2013. The design code of record for the shell side of the RSGs at PINGP Unit 1 is the ASME B&PV Code, Section III, 1995 Edition with 1996 Addenda. The design code of record for the shell side of the SGs at PINGP Unit 2 is the ASME B&PV Code, Section III, 1965 Edition with Winter 1966 Addenda. The design code of record for the tube side is the same code of record as the shell side for each unit.

The revised design conditions were reviewed by the licensee to determine the impact on the existing design basis analyses for the critical original and RSG primary and secondary side components affected by the proposed MUR-PU. As previously discussed, the design and operating pressures for the RCS remain unchanged due to the proposed power uprate and the slight operating temperature changes remain well within the allowable design limits. The licensee confirmed that the design transients affecting the Unit 1 and Unit 2 SGs remain unchanged due to the proposed power uprate and no additional transients have been proposed. The licensee discussed in its submittal that a review of the Unit 1 and Unit 2 SG support designs, performed by the licensee, noted an unexpected variance in the design of the two units. However, given that there is no effect on the supports due to the MUR-PU and the analyses of record for the supports remain bounding, the NRC staff considers the supports adequate for operation at the proposed power level.

The licensee confirmed that the existing analyses of record for the SGs in each unit will continue to be bounded at the proposed power level due to the MUR-PU. As such, all loads, stresses, and fatigue usage factors remain valid at the higher power level. As previously indicated in the licensee's response to an NRC staff RAI regarding the fatigue evaluations performed for SSCs affected by the proposed MUR-PU (Reference 2), all fatigue evaluations considered a 60-year plant life.

3.5.2.3.3 Reactor Coolant Pump Components

The licensee evaluated the effects of the proposed power uprate on the structural integrity of the pressure retaining portions of the RCPs, which includes the casing, main flanges and main flange bolts, in Section IV.A.1.vii of Enclosure 2 to Reference 1. Table IV.1.D.1 of Enclosure 2 provides the design codes of record for the SSCs evaluated in support of the MUR-PU at the PINGP. The pressure retaining portions of the RCPs at PINGP were designed in accordance with Article 4 of the ASME B&PV Code, Section III. The licensee reviewed the revised design conditions to determine the impact on the existing design basis analyses for the affected portions of the RCPs. The RCP structural evaluations are highly dependent on the vessel inlet temperature, which decreases by only 0.5 °F from the current conditions to the proposed conditions resulting from the MUR-PU. The licensee confirmed that all pressure-retaining components of the RCPs are bounded by the current analyses of record. Also, no additional transients have been proposed. As such, the structural stresses and fatigue usage factors resulting from the MUR power uprate will remain within the acceptance criteria found in the current licensing basis. As previously indicated, in response to an NRC staff RAI regarding the fatigue evaluations performed for SSCs affected by the proposed MUR-PU, the licensee confirmed that all fatigue evaluations considered a 60-year plant life.

3.5.2.3.4 Pressurizer

The licensee evaluated the effects of the proposed power uprate on the structural integrity of the pressurizer, nozzles and the pressurizer surge line in Section IV.1.A.viii of Enclosure 2 to Reference 1. Table IV.1.D.1 of Enclosure 2 provides the design codes of record for the SSCs evaluated in support of the MUR-PU. The design code of record for the pressurizer at PINGP Unit 1 is the ASME B&PV Code, Section III, 1965 Edition with Summer 1966 Addenda. The design code of record for the pressurizer at PINGP Unit 2 is the ASME B&PV Code, Section III, 1965 Edition with Winter 1966 Addenda. The pressurizer surge lines were re-evaluated in accordance with Section III of the ASME B&PV Code to address NRC Bulletin 88-11, "Pressurizer Surge Line Thermal Stratification." NRC Bulletin 88-11 required licensees to evaluate pressurizer surge lines in order to demonstrate that these lines maintained acceptable stresses and fatigue usage factors due to thermal stratification and cycling.

With regards to the structural integrity of the pressurizer, the governing loading conditions occur when the RCS pressure is elevated and the hot and cold leg temperatures are low. The licensee reviewed the revised design conditions to determine the impact on the existing structural design basis analyses for the pressurizer. Given that the RCS pressure remains unchanged at the uprated conditions and the hot and cold leg temperature variations at these conditions remain bounded, the licensee concluded that the current design basis analyses remain bounding for the PINGP pressurizers. The licensee did note that the Unit 2 surge line and surge line nozzles experience higher thermal stresses than those currently accounted for in the fatigue analyses of record, yielding nonconservative fatigue usage factors. A subsequent analysis demonstrated that these components meet the applicable stress and fatigue usage acceptance criteria of the design code of record for current operation. For these components, the design conditions at current operation bound the conditions that exist at the proposed MUR power level. Therefore, for the pressurizer, pressurizer surge line and nozzles, the MUR-PU conditions are bounded by the current analyses of record and the allowable stress limits and fatigue usage factors for these components, provided by the applicable codes of record identified above, will continue to be satisfied following the implementation of the proposed power uprate. As previously indicated, in response to an NRC staff RAI regarding the fatigue evaluations performed for SSCs affected by the proposed MUR-PU, the licensee confirmed that all fatigue evaluations considered a 60-year plant life.

3.5.2.3.5 Reactor Core Support Structures and Reactor Pressure Vessel Internals

The licensee evaluated the effects of the proposed power uprate on the structural integrity of the reactor core support structures and RVIs in Section IV.1.A.ii of Enclosure 2 in Reference 1. Section 3.6 of the PINGP USAR details the mechanical design criteria, including the design codes of record, used in the design and analysis of the PINGP core support structures and RVIs. Mechanical and structural evaluations were performed by the licensee to determine any effects on the RVIs due to the conditions which would be present following the implementation of the proposed MUR power uprate. Following its assessment of the effects of the MUR power uprate on the thermal-hydraulic and structural loadings, including those due to FIV, imposed on the core support structures and RVIs, the licensee concluded that these components will not be adversely affected by the implementation of the proposed MUR power uprate and will continue to meet their structural design basis acceptance criteria requirements following the implementation of the proposed power uprate.

In response to an NRC staff RAI regarding the analyses of record for the core support structures and RVIs, the licensee indicated in Reference 2 that the structural evaluations and analyses performed for these components, in support of a prior license amendment request (Reference 6) to transition to Westinghouse 0.422-inch 14x14 Vantage+ fuel, remain bounding as they apply to the conditions at the proposed power level. In Reference 7, the licensee documented the results of its structural evaluations performed for the core support structures and RVIs. It was demonstrated in these responses that the core support structures and RVIs meet the design code of record acceptance criteria for stresses and fatigue. Additionally, the licensee demonstrated that the core support structures and RVIs also maintain acceptable margins with respect to the effects of FIV on the core support structures and RVIs. The staff subsequently documented its acceptance of the fuel transition LAR in Reference 8. As previously indicated, in response to an NRC staff RAI regarding the fatigue evaluations performed for SSCs affected by the proposed MUR power uprate, the licensee confirmed in Reference 2 that all fatigue evaluations considered a 60-year plant life.

The NRC staff has reviewed the licensee's evaluations related to the structural integrity of the reactor core structures and RVIs. For the reasons set forth above, which demonstrate that the reactor core structures and RVIs will continue to meet their design basis acceptance criteria under the conditions of the proposed MUR power level, the NRC staff concludes that the licensee has adequately addressed the effects of the proposed MUR on these components.

Based on the above, the NRC staff further concludes that the licensee has demonstrated that the reactor core structures and RVIs will continue to meet the applicable regulatory requirements, described above, following implementation of the proposed MUR. Therefore, the NRC staff finds the proposed MUR acceptable with respect to the structural integrity of the reactor core structures and RVIs.

3.5.2.3.6 Conclusion

The NRC staff has reviewed the licensee's evaluations related to the structural integrity of the RCS piping, components, and supports. For the reasons set forth above, which demonstrate that the RCS piping, components, and supports will continue to meet their design basis acceptance criteria under the conditions of the proposed MUR-PU power level, NRC staff concludes that the licensee has adequately addressed the effects of the proposed MUR-PU on these components. The NRC staff further concludes that the licensee has demonstrated that these SSCs will continue to meet the applicable regulatory requirements following implementation of the proposed MUR-PU. Therefore, the NRC staff finds the proposed MUR-PU to be acceptable with respect to the structural integrity of the RCS piping, components, and supports.

3.5.2.4 High-Energy Line Break Locations

The licensee stated that the current HELB analysis for the PINGP was reviewed in support of the proposed MUR-PU. The licensee's review of the PINGP HELB analysis was performed to ensure that the proposed MUR-PU does not impact the current HELB analysis of record as it relates to postulated HELB locations and the potential dynamic effects due to HELBs, including pipe whip, missile generation and corresponding compartment flooding and pressurizations. Section IV.1.B.vii of Enclosure 2 to Reference 1 states that all MS, SG blowdown, and CVCS line breaks are postulated and analyzed under no-load conditions that bound the conditions present at the proposed power level, which precludes the possibility of the proposed power uprate having any effect on these HELB analyses.

The licensee noted that the changes in the temperature and pressure in certain high-energy lines, namely the MS and main FW system piping runs, were considered in evaluating any pressure, temperature, and flooding effects on compartments containing portions of these high-energy lines. The mass flow rate of the FW decreases due to the decrease in density of the FW. The effects due to the increased FW enthalpy were also reviewed by the licensee to determine what effect the enthalpy rise may have on the aforementioned compartment analyses. It was determined that these analyses of record remain bounding under MUR-PU conditions. In Section IV.1.B.viii of Enclosure 2 to Reference 1, the licensee also confirmed that the effects of jet impingement, pipe whip and thrust loads at MUR-PU conditions remain bounded by the current licensing basis. Additionally, the licensee noted that no high-energy lines will be added in support of the proposed power uprate.

The NRC staff has reviewed the licensee's evaluations related to determinations of HELB rupture locations and their corresponding dynamic effects. For the reasons set forth above, which demonstrate that the HELB analyses of record will remain bounding under the proposed MUR-PU power level, the NRC staff concludes that the licensee has adequately addressed the effects of the proposed MUR-PU on these analyses. The NRC staff further concludes that the licensee has demonstrated that all of the regulatory requirements applicable to the HELB analyses will continue to be met following implementation of the proposed MUR-PU. Therefore, the NRC staff finds the proposed MUR-PU to be acceptable with respect to the determination of rupture locations and dynamic effects associated with the postulated rupture of piping.

3.5.2.5 Balance of Plant (BOP) Piping Systems

The licensee evaluated the effects of the proposed power uprate on the structural integrity of the BOP piping, including NSSS interface systems, safety-related cooling water systems and containment systems, in Section IV.1.A.v of Enclosure 2 to Reference 1. A separate discussion on the piping stresses of the BOP piping systems is included in Section IV.1.B.i of Enclosure 2 to Reference 1. As previously indicated, Chapter 11 of the PINGP USAR provides a detailed description of the design bases for the BOP piping systems at the PINGP. Table 11.1-2 indicates that the steam and power conversion system piping were designed in accordance with the USAS B31.1 Code for Power Piping, 1967 Edition. The licensee indicated in its submittal that the current licensing bases remain bounding under the proposed MUR conditions for the BOP piping systems affected by the power uprate. However, the licensee indicated that the analysis of record for the MS piping system was found to contain various discrepancies when it was reviewed in support of the proposed power uprate.

In response to an NRC staff RAI requesting clarification regarding the effects of these discrepancies on the analyses of record, the licensee indicated in Reference 4 that the discrepancies previously noted were errors involving the MS piping system model and certain seismic load inputs to the model. The identification of these discrepancies resulted in overstress conditions on the MS piping system in the upset loading condition. The licensee detailed the operability evaluations performed to evaluate the effects of these overstress conditions in order to justify continued plant operation. The operability evaluations were performed for Hot Zero Power (HZP) conditions and demonstrated that the overstress conditions existing in the MS piping system were operable but nonconforming. In Reference 4, the conditions at HZP are tabulated against the current conditions and the conditions that exist under the MUR-PU; the conditions at HZP are bounding with respect to the current and MUR conditions that demonstrate the bounding nature of the operability stress analyses performed as they relate to the stresses present under current and MUR conditions. The licensee also provided details regarding how each of the various loads on the MS piping system is affected by the proposed MUR-PU. The licensee noted that while the thermal, pressure, and MS safety valve (MSSV) thrust force loads increase slightly due to the power uprate, the analysis of record uses values which bound these slight increases. Additionally, it was stated that the MS pipe rupture loads are unaffected by the proposed uprate given that these loads are calculated based on HZP conditions, which bound the conditions at the proposed power level. The staff considers the licensee's assessment of the effects of the MUR-PU on the MS piping adequate based on the licensee's demonstration that the MS piping stresses present under current and MUR conditions remain bounded by the analyses of record.

The NRC staff has reviewed the licensee's evaluations related to the PINGP BOP piping systems. For the reasons set forth above, which demonstrate that the BOP piping systems will continue to meet their design basis acceptance criteria under the conditions of the proposed MUR power level, NRC staff concludes that the licensee has adequately addressed the effects of the proposed MUR-PU on these components. The NRC staff further concludes that the licensee has demonstrated that these SSCs will continue to meet the applicable regulatory requirements following implementation of the proposed MUR-PU. Therefore, the NRC staff finds the proposed MUR-PU to be acceptable with respect to the structural integrity of the BOP piping systems at the PINGP.

3.5.2.6 Safety-Related Valves

The NRC staff reviewed the licensee's safety-related valves analysis. The NRC's acceptance criteria for review are based on 10 CFR 50.55a, "Codes and Standards." Additional information is also provided by the plant-specific evaluations of Generic Letter (GL) 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance," GL 95-07, "Pressure Locking and Thermal

Binding of Safety-Related Power-Operated Gate Valves," and GL 96-05, "Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves."

The licensee reviewed the impact of the proposed MUR-PU conditions on the existing design basis analyses for the safety-related valves. In Sections IV.1.A.ix, IV.1.B.iv, IV.1.B.v, and IV.1.B.vi of Enclosure 2, "Summary of Measurement Uncertainty Recapture Power Uprate Evaluation," of Reference 1, the licensee reviewed the revised design and operating conditions resulting from the MUR-PU against the CLB for all safety-related valves and confirmed that the existing valve capacities, setpoints and operating conditions remain valid for the MUR-PU conditions. In Section IV.1.E related to motor-operated valves, the licensee reviewed the MUR-PU impact on the requirements of NRC Bulletin 85-03, "Motor-Operated Valve Common Mode Failures During Plant Transient Due to Improper Switch Settings," GL 89-10, GL 95-07, and GL 96-05. The evaluation determined that the MUR-PU conditions are bounded by the CLB for all safety-related valves, and no further evaluation is required. In Section IV.1.E for the Inservice Testing (IST) Program, the licensee stated that there are no modifications or replacement of ASME Pressure and Vessel Code Class components for the MUR-PU, and there are no new safety-related functions required of existing equipment to accommodate the MUP-PU. Therefore, the licensee confirmed that the current IST program for safety-related valves will remain valid after the MUR-PU is implemented.

The licensee confirms that there are insignificant changes in temperature, flow, and operating pressure, that the MUR-PU conditions are bounded by the CLB for all safety-related valves, and that none of the safety-related valves require a change to valve design or operating conditions as a result of the MUR-PU. The NRC staff finds the performance of existing safety-related valves and the current IST program acceptable with respect to the MUR-PU.

3.5.3 Conclusion

The NRC staff has reviewed the licensee's assessment of the impact of the proposed MUR-PU on NSSS and BOP systems and components with regard to stresses, CUFs, flow-induced vibration, HELB locations, and jet impingement and thrust forces. Based on this review, the staff concludes that the proposed MUR-PU will not have an adverse impact on the structural integrity of the piping systems, components, their supports, RVIs, CRDMs, or BOP piping.

3.6 Reactor Systems

The NRC staff in the Reactor Systems Branch reviewed the thermal-hydraulic aspects of UFM applications. All other aspects contributing to uncertainty, such as most transducer characteristics, physical dimensions, signal processing operations other than those that may affect velocity profile, and operator-display interfaces, were not part of the review. The staff's review was limited to the thermal hydraulic aspects of the CheckPlus™ UFM, including some aspects of the transducers that may influence the perceived flow profile, and consideration of the associated uncertainty.

3.6.1 Regulatory Evaluation

Early revisions of 10 CFR 50.46, and Appendix K to 10 CFR 50, required licensees to base their LOCA analysis on an assumed power level of at least 102-percent of the licensed thermal power level to account for power measurement uncertainty. The NRC later modified this requirement to permit licensees to justify a smaller margin for power measurement uncertainty. Licensees may apply the reduced margin to operate the plant at a level higher than the previously licensed power. The licensee proposed to use a Caldon LEFM CheckPlus™ system to decrease the uncertainty in the measurement of FW flow, thereby decreasing the power level measurement uncertainty from 2.0-percent to 0.36-percent.

The licensee developed its LAR consistent with the guidelines in NRC RIS 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications."

3.6.2 Technical Evaluation

The NRC staff reviewed the thermal-hydraulic aspects of the LEFM CheckPlus™ system installation, including its laboratory calibration, the effects of system changes such as transducer replacement, and the impact the system installation will have, if any, on the applicable plant safety analyses, as discussed in the following subsections.

3.6.2.1 FW Flow Measurement Device Installation

The Caldon LEFM CheckPlus™ Systems at the PINGP, Units 1 and 2, consist of two metering spool pieces and an electronics unit cabinet. The Unit 1 spool pieces are installed in the 16-inch Loop A and B FW flow lines downstream of the existing FW flow venturis. The Unit 2 spool pieces are installed in the 16-inch Loop A and B FW flow lines upstream of the existing flow venturis. Location differences between Units 1 and 2 are due to different piping arrangements in each unit.

The devices are installed in accordance with the requirements in the approved Caldon Topical Reports ER-80P and ER-157P related to the LEFM Check and LEFM CheckPlus™ Systems. After plant installation, the licensee compared the "as-installed" measurement uncertainties with that of the measurement uncertainties obtained during testing and calibration at Alden Labs and the measurement uncertainties were found to be consistent.

3.6.2.2 CheckPlus™ Inoperability

To operate above the current licensed thermal power of 1650 MWt, the licensee proposes to use the Caldon LEFM CheckPlus™ System in the normal and the "ALERT" modes. In the "FAIL" mode, the input for the calorimetric will not use the Caldon LEFM data. Instead, the input will revert to the original source from the venturis.

In the normal mode, the LEFM has a FW flow temperature measurement accuracy of approximately 0.3 percent. The MUR power uprate assumes a conservative 0.36 percent accuracy. The LEFM provides input to the plant computer program. This program makes the LEFM status and calorimetric data available for use for displays, trending, and various program uses as necessary.

The "ALERT" mode is entered when the LEFM internal checking system detects an alarm condition or other condition impacting a single flow plane on either LEFM meter. In this mode, the accuracy of the LEFM is 0.54 percent and the plant will derate power to 99.82 percent or

1674 MWt or less if the LEFM is not returned to normal status before the next scheduled daily Power Range Nuclear Instrumentation calibration.

The "FAIL" mode is entered when the LEFM internal checking system detects an alarm condition of at least one flow plane is not operable in each meter, if loss of the data link between the LEFM and the ERCS occurs, or if the reactor trip breakers are opened. The plant computer program automatically reverts to using the input from the FW flow venturis. The plant must derate power to equal or less than 1650 MWt or 98.38 percent if the LEFM is not returned to normal status before the next scheduled daily Power Range Nuclear Instrumentation calibration. Stated differently, the plant will be operated as though the CheckPlus™ was never installed and the power uprate was not in effect.

In the "FAIL" mode, the venturi input will revert to the last known acceptable correction factors. Operation with the last good correction factors derived from the LEFM will be limited to a maximum of seven days. If the LEFM system status is not returned to service in the allotted time, correction factors for the venturi FW flow and inline temperature inputs will be set to 1.0. These actions are to be covered in the TRM. The NRC staff finds that operation with an inoperable CheckPlus™ has been acceptably addressed.

3.6.2.3 Transducer Replacement

Uncertainty associated with transducer replacement was addressed in "Caldon Ultrasonics, Engineering Report: ER-551P Rev.3, LEFM✔ + Transducer Installation Sensitivity," dated March 2007 (Reference 9). Since the transducer replacement uncertainty is incorporated in the original and no additional uncertainty terms need to be applied whenever a transducer is replaced, the NRC staff finds that transducer installation variability has been acceptably addressed.

3.6.2.4 CheckPlus™ Calibration

CheckPlus™ calibration was accomplished at Alden Laboratories. The test configuration is provided in Reference 1. The NRC staff reviewed drawings and schematics provided in Reference 2 and confirmed that, insofar as configuration is concerned, the laboratory configuration largely matched the in situ configuration.

The calibration studies were completed to determine the meter factor, a calibration coefficient, and also parametric tests to determine the meter sensitivity to upstream hydraulics. The tests were completed using previously applied procedures and laboratory measurement elements traceable to the National Institute of Standards and Technology. The NRC staff finds that the licensee's laboratory calibration was sufficiently fabricated to provide meaningful data based on the modeling of piping geometry of the UFM at the PINGP.

3.6.2.5 Caldon Topical Reports Safety Evaluation Criteria

The NRC staff reviewed and approved Caldon Topical Reports ER-80P and ER-157P related to the LEFM CheckPlus™ Systems. In approving the Caldon Topical reports, the NRC staff established four criteria to be satisfied by each licensee as follows:

Criterion 1

Discuss maintenance and calibration procedures that will be implemented with the incorporation of the LEFM, including processes and contingencies for inoperable LEFM instrumentation and the effect on thermal power measurements and plant operation.

The licensee will perform maintenance on the unit-specific LEFM specific system during each refueling outage. PINGP will use site procedures developed in accordance with the vendor maintenance and troubleshooting manual. Maintenance will be performed by qualified personnel that have participated in formal vendor training.

The MUR power uprate is based on the use of the Caldon LEFM CheckPlus™ flowmeter. The licensee will derate to the power level 1674 MWt, if the Caldon LEFM CheckPlus ™ flowmeter is in the "ALERT" mode during the next scheduled daily power calorimetric. The licensee will derate to the current license power level (1650 MWt), if the Caldon LEFM CheckPlus ™ flowmeter is out of service in the "FAIL" mode beyond the allowed outage time (7 days). After the allowed outage time has expired and without an operable CheckPlus ™, the plant will be operated as though the CheckPlus ™ was never installed and the power uprate was not in effect.

The responses to a change in LEFM status will be captured in a new TRM section and the critical alarm response procedure will be updated to reflect the limiting power levels with the implementation of the MUR-PU.

Criterion 2

For plants that currently have LEFMs installed, provide an evaluation of the operational and maintenance history of the installed instrumentation and confirmation that the installed instrumentation is representative of the LEFM system and bounds the analyses and assumptions set forth in Reference 5

PINGP installed the LEFM's in Units 1 and 2 during the February 2008 and October 2008 outages, respectively. Following Unit 1 installation, the LEFM indicated a 1 percent higher power level than the calorimetric calculation based on flow venturis. The largest source of deviation was associated with flow indication in the Loop A FW flow. The licensee implemented corrective actions and the system has functioned normally. Unit 1 experienced an unplanned trip in July 2008 and May 2009. The LEFM remained in operation and responded as expected during the time offline and during the return to power. The venturi FW flow and temperature correction factors returned to within +/- 0.001 of the correction factor values that existed prior to the trip.

Following Unit 2 installation, the LEFM indicated a 0.5 percent higher power level than the calorimetric calculation based on flow venturis. The largest source of deviation was associated with flow indication in the Loop A FW flow. The licensee implemented corrective actions and the system has functioned normally.

Criterion 3

> Confirm that the methodology used to calculate the uncertainty of the LEFM in comparison to the current FW instrumentation is based on accepted plant setpoint methodology (with regard to the development of instrument uncertainty). If an alternative approach is used, the application should be justified and to both venturi and ultrasonic flow measurement instrumentation installations for comparison.

PINGP uses a methodology consistent with the approved topical reports to calculate the uncertainty of the Caldon LEFM CheckPlus™ system. Errors associated with the plant computer calculation of the secondary power calorimetric based on flow venturis use a sensitivity analysis to determine the relative impact of channel errors associated with the input value. Errors determined to be insignificant, generally less than 10 percent of the largest errors, were eliminated and the remaining errors were combined using the square root sum of the squares method as the errors are random and nearly normally distributed. Dependent errors, such as temperature and enthalpy errors, were mathematically combined to create random errors and then combined with the remaining errors in the same square root sum of the squares method. The LEFM uncertainty calculations have a 95 percent confidence interval, 95 percent probability flow measurement.

Criterion 4

> For plants where the ultrasonic meter (including LEFM CheckPlus™ System) was not installed and flow elements calibrated to a site-specific piping configuration (flow profiles and meter factors not representative of the plant-specific installation), additional justification should be provided for its use. The justification should show that the meter installation is either independent of the plant-specific flow profiles for the stated accuracy, or that the installation can be shown to be equivalent to known calibrations and plant configurations for the specific installation including the propagation of flow profile effects at higher Reynolds numbers. Additionally, for previously installed calibrated elements, confirm that the piping configuration remains bounding for the original LEFM CheckPlus™ System installation and calibration assumptions.

The Caldon LEFM CheckPlus™ system was calibrated at Alden Labs using plant-specific piping configurations. In its LAR, the licensee noted that the PINGP flow elements are installed in the same piping configuration as tested at Alden laboratories, with the exception that the installation of each LEFM in Unit 2 is 8 inches closer to the inlet of the FW venturis than in the Alden laboratories configuration. Post-installation commission testing of the Unit 2 LEFMs verified the actual plant installation remained bounded by the original LEFM installation and calibration assumptions.

Based on its review of the licensee's responses, the NRC staff determined that the licensee has addressed the four criteria specified in the NRC staff's evaluation of topical reports ER-80P and ER-157P and it is consistent with the guidelines of RIS 2002-03 with regards to the Reactor Systems Branch review responsibility.

3.6.2.6 Accident and Transient Analysis

The licensee provided sufficient information in its license amendment request to demonstrate that each accident and transient analysis contained in Chapter 14 of the PINGP USAR was performed with such power uncertainty treatment that the results are bounding of operation at the requested uprated power level. This was accomplished using analyses in three categories:

(1) Statistical analyses were performed assuming the plant nominal power level with a statistical allocation for power level uncertainty;

(2) Bounding analyses were performed assuming a power level of 1683 MWt or higher; and

(3) Certain analyses, like zero-power transients, are insensitive to core power level and remain bounding.

In all cases, the analytic results are bounding of facility operation at the uprated power level with reduced power uncertainty.

Regulatory Issue Summary 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications," states the following:

> *In areas (e.g., accident/transient analyses, components, systems) for which the existing analyses of record do bound plant operation at the proposed uprated power level, the staff will not conduct a detailed review.*

Since the licensee's accident and transient analyses all bound plant operation at the proposed uprated power level, the NRC staff did not conduct a detailed review. This is consistent with the guidance discussed in RIS 2002-03. The NRC staff finds the requested uprate acceptable with respect to the accident and transient analyses based on the bounding nature of the analyses.

3.6.3 Conclusion

The NRC staff reviewed the reactor systems and thermal-hydraulic aspects of the proposed license amendment request in support of implementation of a measurement uncertainty recapture. Based on the considerations discussed above, the NRC staff determined that the results of the licensee's analyses related to these areas continue to meet applicable acceptance criteria following implementation of the MUR-PU. Most of the current analyses of record are based on operation at 1683 MWt, which includes 2.0-percent measurement uncertainty. The proposed amendment is based on the use of a Caldon LEFM Check Plus™ system that would decrease the uncertainty in the FW flow, thereby decreasing the power level measurement uncertainty from 2.0-percent to 0.36-percent. In these cases, the proposed MUR-PU RTP of 1677 MWt is bounded by the current analyses of record.

3.7 <u>Reactor Pressure Vessel and Internals Integrity</u>

The NRC staff's review in the area of RPV and internals integrity focuses on the impact of the proposed MUR-PU on pressurized thermal shock (PTS) calculations; RPV pressure-temperature (P-T) limits; the integrity of the pressurizer shell and RPV internals; upper shelf energy (USE) evaluations; RPV surveillance capsule withdrawal schedules; and neutron fluence calculations. This review was conducted consistent with the guidance contained in RIS 2002-03 to verify that the results of licensee analyses related to these areas continue to meet the requirements of 10 CFR 50.60, 10 CFR 50.61, and Appendices G and H to 10 CFR Part 50, following implementation of the proposed MUR-PU.

3.7.1 Pressurized Thermal Shock

3.7.1.1 Regulatory Evaluation

The PTS evaluation provides a means for assessing the susceptibility of PWR RPV beltline materials to failure during a PTS event to assure that adequate fracture toughness exists during reactor operation. The staff's requirements, methods of evaluation, and safety criteria for PTS assessments are given in 10 CFR 50.61, *"Fracture Toughness Requirements for Protection Against Pressure Thermal Shock Events."* The NRC staff's review covered the PTS methodology and the calculations for the reference temperature for pressurized thermal shock (RT_{PTS}) at the expiration of the license, considering neutron embrittlement effects.

3.7.1.2 Technical Evaluation

The licensee noted that the PTS calculations were performed for PINGP, Units 1 and 2, using the latest procedures specified by the NRC in 10 CFR 50.61. Updated end of life (EOL) neutron fluence projections that considered the effect of the MUR-PU were developed. These projections are documented in WCAP-14781, *"Evaluation of Pressurized Thermal Shock, Revision 4"* for PINGP, Unit 1, dated September 2007, and WCAP-14638, *"Evaluation of Pressurized Thermal Shock, Revision 4"* for PINGP, Unit 2, dated September 2007, respectively.

The licensee provided in Reference 1 the RT_{PTS} values for the limiting beltline materials for Units 1 and 2 of the PINGP. In Table IV.1.C.1 of Enclosure 2, the following values were provided for Unit 1:

Material	RT_{PTS}
Nozzle Shell Forging B	89 °F
Intermediate Shell Forging C	125 °F
Weld Seam 2	157 °F
Weld Seam 3	156 °F
Lower Shell Forging D	92 °F

In Table IV.1.C.2 of Enclosure 2, the following values were provided for Unit 2:

Material	RT_{PTS}
Upper Shell Forging B	72 °F
Intermediate Shell Forging C	110 °F
Weld Seam 2	136 °F
Weld Seam 3	110 °F

Lower Shell Forging D	114 °F

The NRC staff verified that these values were equal to those provided in Tables 4.2-4 and 4.2-5, "PINGP Unit 1 RT_{PTS} at 54 EFPY" and "PINGP Unit 2 RT_{PTS} at 54 EFPY," respectively, contained in the technical and administrative information section (ADAMS Accession No. ML081130673) of the licensee's application for renewed operating license dated April 11, 2008 (ADAMS Accession No. ML081130666), and approved in the "Safety Evaluation Report Related to the License Renewal of Prairie Island Generating Plant, Units 1 and 2," dated October 16, 2009 (ADAMS Accession No. ML092890209). The NRC staff also verified that the neutron fluence values for the MUR-PU are equal to those reported and approved in the license renewal application.

Therefore, the staff concludes that the PINGP, Unit 1 and 2, RPVs will remain within limits for PTS after the MUR-PU. The staff concluded that the PINGP, Unit 1 and 2, RPV materials would continue to meet the PTS screening criteria of 10 CFR 50.61.

3.7.2 P-T Limits and USE

3.7.2.1 Regulatory Evaluation

The regulation at 10 CFR Part 50, Appendix G, provides fracture toughness requirements for ferritic (low alloy steel or carbon steel) materials in the RCPB, including requirements on the USE values used for assessing the safety margins of the RPV materials against ductile tearing and for calculating P-T limits for the plant. These P-T limits are established to ensure the structural integrity of the ferritic components of the RCPB during any condition of normal operation, including anticipated operational occurrences and hydrostatic tests. The NRC staff's P-T limits review covered the P-T limits methodology and the calculations for the number of effective full-power years (EFPY) specified for the proposed MUR power uprate considering neutron embrittlement effects.

3.7.2.2 Technical Evaluation

As part of its December 28, 2009, submittal, the licensee proposed to update the PTLR analysis methodology described in WCAP-14040-A, Revision 4. The licensee's current PTLR methodology is based on WCAP-14040-A, Revision 2. However, the licensee was not prepared to provide all of the information necessary to support the staff's review of this action at this time. As a result of discussions with the licensee in a telephone conference call on January 29, 2010, the licensee decided to withdraw the requested change to the PTLR analysis methodology and provided a response in a letter dated April 23, 2010 (ADAMS Accession No. ML101130449). The licensee will continue to use the existing PTLR methodology based on WCAP-14040-A, Revision 2. At a later date, the licensee will update to the WCAP-14040-A, Revision 4 methodology via a separate license amendment which will be reviewed by the NRC at that time.

For this LAR, the licensee performed a review of the applicability dates of the P-T limits in the existing PINGP, Unit 1 and 2 PTLR (PTLR, Revision 3 dated October 2002). The licensee noted that these limits are currently defined as being applicable to 35 EFPY of plant operation. The licensee's review of their applicability was performed by comparing the neutron fluence values used to generate the P-T limits to the uprated neutron fluence values for the RPV beltline materials determined using the provisions noted in Section 2.1 of this SE. The licensee's results showed that the updated projected neutron fluence values at 35 EFPY, which reflect the effects of the MUR-PU, were lower than the 35 EFPY values upon which the existing P-T limits are based. Since neutron fluence is a key factor in the adjusted reference temperature (ART) calculation, lower neutron fluence values will result in lower ART values and, in turn, the potential for less restrictive P-T limits. Therefore, the staff determined that the current P-T limits are more restrictive than P-T limits that would be consistent with the updated projected neutron fluence values which include the effects of the MUR power uprate, that the MUR power uprate conditions are bounded by the CLB, and that the existing P-T limits are still valid through 35 EFPY of operation.

The licensee also addressed the impact of the MUR power uprate on the PINGP, Unit 1 and 2 USE evaluations. Based on the current, pre-MUR power uprate analysis, all PINGP, Unit 1 and 2 RPV materials are expected to have a USE greater than 50 ft-lb through EOL as required by 10 CFR Part 50, Appendix G. The licensee re-calculated the EOL USE values for the PINGP, Unit 1 and 2 RPV beltline materials using updated EOL neutron fluence projections based on the provisions noted in Section 2.1 of this SE and procedures specified by the NRC in RG 1.99, Revision 2, *"Radiation Embrittlement of Reactor Vessel Materials."* Consistent with the results discussed above, the updated projected EOL neutron fluence values, which reflect the effects of the MUR power uprate, were lower than the values upon which the existing USE evaluations are based. Hence, the licensee demonstrated that the USE values for all PINGP, Unit 1 and 2 materials, based on the updated projected EOL neutron fluence values, would be continue to be greater than 50 ft-lb through EOL and are, in fact, bounded by the values currently in the CLB. Therefore, the NRC staff finds the proposed MUR power uprate to be acceptable with respect to the P-T limits and USE.

3.7.3 Reactor Pressure Vessel (RPV) Internals and Core Support Materials

3.7.3.1 Regulatory Evaluation

The RPV internals and core support structures include SSCs that perform safety functions or whose failure could affect safety functions performed by other SSCs. These safety functions include reactivity monitoring and control, core cooling, and fission product confinement (within both the fuel cladding and the RCPB). The NRC's acceptance criteria for RPV internals and core support materials are based on GDC 1 and 10 CFR 50.55a for material specifications, controls on welding, and inspection of RPV internals and core supports. Matrix 1 of NRC RS-001, Revision 0, "Review Standard for Extended Power Uprates," provides references to the NRC's approval of the recommended guidelines for RPV internals in Topical Reports WCAP-14577, Revision 1-A, "License Renewal Evaluation: Aging Management for Reactor Internals" (March 2001), and BAW-2248-A, "Demonstration of the Management of Aging Effects for the Reactor Vessel Internals" (March 2000).

3.7.3.2 Technical Evaluation

The licensee discussed the impact of the PINGP, Units 1 and 2 MUR-PU on the structural integrity of the RPV internal components in Enclosure, Section IV.1.A.ii of the submittal and the licensee's RAI response dated April 19, 2010 (Reference 2). The licensee concluded that the RPV internals and core support structures are not adversely affected either by the MUR-PU reactor coolant system conditions and transients, or by secondary effects on reactor thermal-hydraulic or structural performance. Therefore, the staff determined that, based on the above, the existing loads remain valid and the stresses and fatigue values also remain valid.

The RPV internals of PWR-designed light-water reactors may also be susceptible to the following aging effects:

- cracking induced by thermal cycling (fatigue-induced cracking), stress corrosion cracking (SCC), or irradiation assisted stress corrosion cracking (IASCC);

- loss of fracture toughness properties induced by radiation exposure for all stainless steel grades, or the synergistic effects of radiation exposure and thermal aging for cast austenitic stainless steel (CASS) grades;

- stress relaxation in bolted, fastened, keyed or pinned RPV internal components induced by irradiation exposure and/or exposure to elevated temperatures; and

- void swelling (induced by radiation exposure).

Matrix 1 of NRC RS-001, Revision 0 provides the staff's basis for evaluating the potential for extended power uprates to induce these aging effects. Note 1 to Matrix 1, states that guidance on the neutron irradiation-related threshold for IASCC for PWR RPV internals are given in BAW-2248-A and WCAP-14577, Revision 1-A. This Matrix 1 note further stated that for thermal and neutron embrittlement of CASS, SCC, and void swelling, licensees will need to provide plant-specific degradation management programs or participate in industry programs that investigate degradation effects and determine appropriate management programs.

In its RAI response dated April 19, 2010, the licensee discussed its plant-specific PWR RPV internals program. The program was developed as a License Renewal Aging Management Program (AMP). It is based on the EPRI *"Pressurized Water Reactor Internals Inspection and Evaluation Guidelines"* (MRP-227) and the ASME Code, Section XI Inservice Inspection, Subsections IWB, IWC, and IWD Program. The licensee noted that the MRP-227 guidelines consider various aging factors including neutron fluence exposure, temperature history, and representative stress levels for determining relative susceptibility of PWR internals to postulated aging mechanisms that include SCC, IASCC, wear, fatigue, thermal aging embrittlement, irradiation embrittlement, irradiation-enhanced stress relaxation and creep, and void swelling. The licensee's AMP will be reviewed by the NRC separately from this MUR power uprate application.

The NRC staff determined that, based on above, the licensee's development of an AMP for managing the degradation of RPV internals and core support structures is consistent with the staff guidance stated in RS-001, Revision 0, Matrix 1 and is, therefore, acceptable for the purpose of addressing this issue in its MUR power uprate application.

3.7.4 Surveillance Capsule Withdrawal Schedule

3.7.4.1 Regulatory Evaluation

The RPV material surveillance program provides a means for determining and monitoring the fracture toughness of the RPV beltline materials to support analyses for ensuring the structural integrity of the ferritic components of the RPV. Appendix H of 10 CFR Part 50 provides the staff's requirements for the design and implementation of the RPV material surveillance program.

3.7.4.2 Technical Evaluation

In its letter dated December 28, 2009 (Reference 1), the licensee noted that the current surveillance capsule withdrawal schedule for PINGP, Units 1 and 2 is based on American Standard for Testing of Materials (ASTM) E185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels." Per ASTM E185-82, the withdrawal of a capsule is to be scheduled at the nearest vessel refueling outage to the calculated EFPY established for the particular surveillance capsule withdrawal.

In its RAI response dated April 19, 2010 (Reference 2), the licensee noted that six surveillance capsules were initially installed in each RPV at the PINGP, Units 1 and 2, of which four capsules have been removed from each unit as required by their established withdrawal schedules. The last capsule to be removed from PINGP, Unit 1 was Capsule "S" at 18.12 EFPY and the last capsule removed from PINGP, Unit 2 was Capsule "P" at 17.24 EFPY. Two surveillance capsules remain installed in each RPV.

To determine the surveillance capsule withdrawal schedule, the licensee calculated ΔRT_{NDT} by using updated projected neutron fluence values. The neutron fluence projections accounted for the MUR power uprate and indicated that the ΔRT_{NDT} of the RPV beltline materials remains between 100 °F and 200 °F. It was concluded that both PINGP units will continue to maintain a four capsule minimum withdrawal schedule in accordance with ASTM E185-82.

Furthermore, in its RAI response, the licensee described plans for withdrawing an additional surveillance capsule from each unit. These plans were originally documented by letter dated November 12, 2008, (Letter Number L-PI-08-097, ADAMS No. ML083370202) in a response to RAI questions regarding the PINGP, Units 1 and 2 license renewal application (LRA). The licensee stated that one of the remaining capsules in each unit will be withdrawn in accordance with the requirements of ASTM E-185-82, Section 7.6.2 when its neutron fluence value exceeds the peak EOL RPV fluence, but prior to exceeding twice that neutron fluence level. This was based on a projected EOL neutron fluence value for the RPV at 54 EFPY associated with license renewal. Therefore, the neutron fluence value range for the next capsule from each unit is as follows:

- PINGP, Unit 1 - Between 5.162×10^{19} and 1.032×10^{20} n/cm^2 (E > 1.0 MeV)
- PINGP, Unit 2 - Between 5.196×10^{19} and 1.039×10^{20} n/cm^2 (E > 1.0 MeV)

The licensee's schedules for capsule removal for each unit are as follows, along with the lead factor and projected neutron fluence values for each capsule at the time of the next withdrawal:

- PINGP, Unit 1- Refueling outage 1R27, expected in spring 2011 at 31.6 EFPY

Withdraw: Capsule N, Lead Factor = 1.77, Projected Fluence = 5.893 x 10^{19} n/cm^2 (E > 1.0 MeV)
Standby: Capsule T, Lead Factor = 1.89, Projected Fluence = 6.292 x 10^{19} n/cm^2 (E > 1.0 MeV)

- PINGP, Unit 2 - Refueling Outage 2R27, expected in 2012 at 32.2 EFPY

Withdraw: Capsule S, Lead Factor = 1.72, Projected Fluence = 5.739 x 10^{19} n/cm^2 (E > 1.0 MeV)
Standby: Capsule N, Lead Factor = 1.72, Projected Fluence = 5.739 x 10^{19} n/cm^2 (E > 1.0 MeV)

A revised capsule withdrawal schedule consistent with this information was submitted for NRC approval in accordance with the requirements of 10 CFR Part 50, Appendix H in NSPM letter L-PI-10-029, "Request for Revision to Reactor Vessel Material Surveillance Capsule Withdrawal Schedule for PINGP," dated March 30, 2010 (ADAMS Accession No. ML100900089).

The NRC staff determined that since the neutron fluence values for the MUR-PU are bounded by the fluence values used to evaluate the license renewal EOL transition temperature shift values, and since the PINGP, Units 1 and 2 RPV surveillance programs will continue to meet the requirements of 10 CFR Part 50, Appendix H under the MUR-PU conditions, they are, therefore, acceptable.

3.7.5 RPV Neutron Fluence Calculations

3.7.5.1 Regulatory Evaluation

Licensee RPV neutron fluence calculations are expected to be performed based upon the use of a staff-approved neutron fluence calculational methodology which is consistent with the guidance in NRC Regulatory Guide 1.190 (RG 1.190), *"Calculation and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence."*

3.7.5.2 Technical Evaluation

In support of the MUR power uprate, the licensee provided updated neutron fluence projections for the RPV which account for the effects of the MUR power uprate. The licensee's updated neutron fluence projections were derived using the neutron fluence calculational methodology of WCAP-14040-A, Revision 4, "*Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves,*" which is consistent with the guidance in NRC RG 1.190. The staff concluded that the licensee's use of the neutron fluence calculational methodology of WCAP-14040-A, Revision 4 was acceptable for this application because it reflects a staff-approved updated calculational methodology when compared to the methodology in WCAP-14040-A, Revision 2 which defines the PINGP CLB.

The licensee's updated projected neutron fluence values reflected the effects of the MUR power uprate and used actual fuel enrichment and burnup history data in lieu of conservative assumptions regarding future fuel loads when compared to previous projected neutron fluence values calculated using the neutron fluence methodology of WCAP-14040-A, Revision 2.

3.7.6 Conclusion

The NRC staff has reviewed the licensee's proposed LAR to increase the rated core thermal power by approximately 1.64 percent and has evaluated the impact that the MUR-PU conditions will have on the structural integrity assessments for the RPV and RPV internals. The staff has determined that the changes identified in the proposed LAR will not impact the remaining safety margins required for the following structural integrity assessments: (1) RPV surveillance program; (2) RPV USE assessment; (3) P-T limits; (4) PTS assessment; and (5) RPV internals and core support structures. Based on the above, the NRC staff finds that sufficient information has been provided to support the requested MUR-PU.

3.8 Electrical Systems

The NRC staff reviewed the licensee evaluation of the impact of MUR power uprate on following electrical systems/components:

- Alternating Current (AC) Distribution System
- Power Block Equipment (Generator, Exciter, Transformers, Isolated-phase bus duct, Generator circuit breaker)
- Direct Current (DC) Distribution System
- Emergency Diesel Generators (EDGs)
- Switchyard
- Grid Stability
- Station Blackout (SBO)
- Environmental Qualification (EQ) Program

3.8.1 Regulatory Evaluation

The regulatory requirements which the staff applied in its review of the application include:

The regulation at 10 CFR 50.63, "Loss of all alternating current [AC] power," requires, in part, that all nuclear plants have the capability to withstand a loss of all AC power (i.e., SBO) for an established period of time, and to recover therefrom.

The regulation at 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants," requires licensees to establish programs to qualify electric equipment important to safety.

USAR GDC 24 and 39 require that sufficient alternative sources of power are provided and designed with adequate independence, redundancy, capacity, and testability to permit the functioning of the protection system in the event of loss of all offsite power. Also, both onsite and offsite power system need to provide the required capacity assuming the failure of a single active component in each power system.

3.8.2 Technical Evaluation

3.8.2.1 AC Distribution System

The AC distribution system is the source of power for the non-safety related buses and for the safety-related emergency buses. It consists of the 4.16 kilovolt (kV), 480 volt (V), 208 V and 120 V systems (not including the EDGs). The licensee stated that the MUR-PU will not require any major equipment modifications and it will only slightly increase each unit's station service loading due to increases in main FW and condensate pumps loading. Furthermore, the licensee stated that the emergency power buses will not be affected by the MUR power uprate since no additional loads will be added on them. Based on this information, the NRC staff concludes that the AC power distribution system will experience minor load changes, and that the AC distribution system has adequate capacity to operate the plant equipment within the design to support the uprated conditions.

3.8.2.2 Power Block Equipment (Generator, Exciter, Transformers, Iso-phase bus duct, Generator circuit breaker)

As a result of the power uprate, the rated thermal power will increase to 1677 MWt from the previously analyzed core power level of 1650 MWt.

The NRC staff requested the licensee provide the uprated loadings for each of the main generators. In its letter of April 19, 2010, the licensee stated that the main generators are rated at 659,000 kilovolt ampere (KVA), 20 kV and 0.9 power factor. The licensee stated that the main generator is capable of approximately 592 MWe. This is above the requested power uprate of 584 MWe (9 MWe increase). The licensee evaluated components and subsystems of the generator and concluded that these components are capable of operating at MUR-PU conditions.

In its response to the NRC staff's RAI, the licensee provided the winter net and gross electrical generation values before and after the MUR-PU. The winter electrical generation values are greater than the summer electrical values and, therefore, they bound them. The licensee stated that the winter net and gross electrical generation values are 551 MWe (Unit 1), 545 MWe (Unit 2) and 575 MWe (Unit 1 and 2), respectively. Also, the licensee stated that the post MUR-PU winter net and gross electrical generation values are 560 MWe (Unit 1), 554 MWe (Unit 2) and 584 MWe (Unit 1 and 2), respectively. Based on this information, the staff finds that the main generators are capable of operation at uprated conditions.

The NRC staff requested the licensee to provide the nameplate ratings and uprated loadings for the isolated phase bus duct. In its April 19, 2010, letter, the licensee stated that the iso-phase bus is rated for 20,000 amperes (A), and that it is within the generator capability curve for the MUR-PU. The generator is rated at 19,024 A for Unit 1 and 19,023 A for Unit 2, which is below the 20,000 A rating of the iso-phase bus. Therefore, the staff finds that the iso-phase bus is capable of operation at uprated conditions, since the increase from the MUR power uprate remains below the iso-phase bus rating.

The NRC staff requested the licensee to provide the nameplate ratings and uprated loadings for the generator step up transformers and plant service transformers. In its April 19, 2010, letter, the licensee stated that the generator step-up transformer is rated at 600 megavolt-amperes (MVA) at a 55°C rise and 672 MVA at a 65°C rise. The generator step-up transformer is operated below the 55°C rating for the great majority of the time and only operates between the 55°C and 65°C rating less than 2 percent of the time. This condition of operation at higher demand duties is acceptable due to it being short-term and infrequent. The licensee stated that the plant service transformers ratings are 24/32/40 MVA at 55°C and 26.9/35.3/44.8 MVA at 65°C. The current total station service load for each unit is between 28 - 31 MVA and that the MUR-PU is expected to add approximately 0.095 MVA. This small increase is within the capacity of the main auxiliary transformers. Therefore, the NRC staff finds that the generator step up transformers and plant service transformers are capable of operation at uprated conditions.

3.8.2.3 DC Distribution System

The 125V DC distribution system supplies power to the instrumentation, control, and safety-related equipment. The 125V DC power source in each unit consists of two independent and redundant subsystems. Each subsystem consists of one 125 VDC battery a battery chargers and the associated distribution equipment.

The licensee stated that the MUR-PU does not affect any DC-powered systems since no loads were added and existing power supplies were unaffected. The NRC staff reviewed the LAR and USAR and confirmed that the power uprate does not impact DC system loads. Based on this information, the staff finds that the analyses for DC system bound MUR-PU conditions.

3.8.2.4 Emergency Diesel Generators

The EDG system is a source of reliable emergency AC power for each PINGP unit. The two EDGs for each unit are capable of sequentially starting and supplying the power requirements to the engineered safeguards electrical buses. Each EDG is sized to start and carry the safety features required for a Design-Basis Accident and loss of offsite power.

The licensee reviewed the emergency onsite power system and found no impact on it as a result of the MUR-PU. Hence, the EDG system is bounded by the PINGP current licensing basis.

Based on the above, the NRC staff finds that the power uprate does not impact EDG system loads. The staff finds that the analyses for the EDG system are bounded by the MUR-PU conditions, and that the onsite power system will continue to meet the requirements of GDC 24 and 39.

3.8.2.5 Switchyard

The switchyard equipment and associated components are classified as non-safety related. The switchyard serves four 345-kV lines and one 161-kV line. The primary function of the switchyard and distribution system is to connect the station electrical system to the transmission grid.

The licensee stated that the switchyard is designed to support operation within the generator capability curves. The electrical current supplied to the switchyard is bounded by the main transformer capability. The small increase in plant output does not significantly impact the switchyard equipment. Therefore, the NRC staff concludes with the analyses that the PINGP switchyard system would not be adversely impacted by the MUR-PU conditions.

3.8.2.6 Grid Stability

In its response to the NRC staff's RAIs, the licensee provided information associated with the following grid studies performed in order to support the MUR-PU:

> Midwest Independent System Operator (MISO) Study (March 2006): The study evaluated the impact of a 38 MWe (18 MWe for MUR and 20 for other anticipated system load increases) addition to the PINGP. MISO performed a system stability analysis, steady-state analysis, and a supplemental analysis in order to complete their transmission interconnection study. The study used the post MUR-PU summer generation values of a net 537 MWe (Unit 1) and 532 MWe (Unit 2), and a gross 569 MWe (Unit 1) and 565 MWe (Unit 2). The study determined that the MUR-PU will have no adverse impact on the transmission system stability, a negligible impact on the transmission facility loadings and bus voltages, and that no new thermal, voltage or stability criteria violations would be produced.
>
> MISO Study (May 2006): This study used the same MUR-PU conditions, but was based on winter generation values. The study used the post MUR-PU winter generation values of a net 568 MWe (Unit 1) and 565 MWe (Unit 2). Similar to the March 2006 study, the May 2006 study determined that the MUR-PU will have no adverse impact on the transmission system stability, a negligible impact on the transmission facility loadings and bus voltages, and that no new thermal, voltage or stability criteria violations would be produced.
>
> Institute of Electrical and Electronics Engineers (IEEE)-765 Study (November 2009): This transient stability study analyzed the strength of the transmission system following system disturbances and the voltage level at PINGP buses after these disturbances. The study used the post MUR-PU summer generation values of a net 537 MWe (Unit 1) and 532 MWe (Unit 2). The study determined that all the PINGP bus voltages were within the acceptable range for all disturbances evaluated. This study was re-evaluated using the post MUR-PU winter generation values of a net 568 MWe (Unit 1) and 565 MWe (Unit 2). The previous summer results have been confirmed and the re-evaluated study determined that all the PINGP bus voltages were within the acceptable range for all disturbances evaluated.

The licensee stated that the minimum voltages for the PINGP grid interface location are based on the equipment required to safely shut down the plant after a dual-unit trip. These voltages are assured on a real-time basis by computer software that monitors grid configuration and conditions. Also, computer models are run for various grid events, including a dual unit trip.

The NRC staff reviewed the grid stability studies and concludes that grid stability and post-trip voltages requirements for the plant are maintained as a result of the MUR-PU.

3.8.2.7 Station Blackout

The regulations in 10 CFR 50.63 require that each light-water cooled nuclear power plant be able to withstand and recover from a loss of all AC power, referred to as an SBO.

The PINGP's SBO coping duration is 4 hours. This is based on the licensee's evaluation of the offsite power design characteristics, emergency AC power system configuration, and EDG reliability, in accordance with the evaluation procedure outlined in NUMARC 87-00 and RG 1.155. The offsite power design characteristics include the expected frequency of a grid-related loss of offsite power, the estimated frequency of loss of offsite power from severe and extremely severe weather, and the independence of offsite power.

The licensee stated that in case of an SBO, one of the EDGs of the non-affected unit is configured to act as an alternate AC (AAC) power source. The plant is licensed to have an AAC source available within 10 minutes, which is capable of providing power to the shutdown buses throughout the 4-hour coping duration. The licensee stated that the heating, ventilation, and air conditioning (HVAC), containment isolation, heat tracing of SBO coping equipment, and cooling to the reactor coolant pump seals will not be affected by the MUR power uprate due to the availability of AAC. The licensee has performed two studies to analyze the required volume for the Condensate Storage Tank (CST). Both studies have shown that the required CST volume for the power uprate or above (102 percent) will be maintained per the technical specification surveillance requirements. Based on this information, the staff finds that the MUR power uprate will have no impact on PINGP's SBO coping duration. Therefore, the NRC staff finds that the PINGP will continue to meet the requirements of 10 CFR 50.63 under power uprate conditions.

3.8.2.8 Environmental Qualification Program

In its LAR submittal, the licensee stated that the small temperature change inside containment due to the MUR-PU is acceptable for the electrical loads. Furthermore, the normal pressures, temperatures, and humidity in EQ areas outside of containment are unaffected by the power uprate. The licensee stated that the environmental qualification of qualified equipment was performed at a core power of 4100 MWt for Westinghouse equipment and 1683 MWt for non-Westinghouse equipment. Both of these qualifications bound the post MUR thermal power of 1677 MWt.

Based on this information, the NRC staff finds that the current EQ parameters remain bounding for the MUR-PU. The staff concludes that the MUR-PU will have no impact on the PINGP's EQ program and that the requirements of 10 CFR 50.49 will continue to be met.

3.8.3 Overall Electrical Systems

Based on the technical evaluation provided above, the NRC staff finds that the PINGP will continue to meet GDC 24 and 39, 10 CFR 50.63, and 10 CFR 50.49. Therefore, the staff finds the MUR-PU to be acceptable.

3.9 Instrumentation and Controls

This power uprate is based on a reduced measurement uncertainty of core thermal power resulting from the installation of a Cameron (formerly Caldon) LEFM CheckPlus™ System to measure feedwater flow and temperature at PINGP. The licensee's submittal referenced Caldon Topical Report ER-80P, Revision 0, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the System," issued March 1997, and its supplement, Topical Report ER-157P, Revision 5, "Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM √™ or LEFM CheckPlus™ System," issued October 2001. These topical reports, which are generically applicable to nuclear power plants, document the ability of the Caldon LEFM Check and √+ Systems to increase the accuracy of flow measurement. The NRC approved Topical Report ER-80P and its supplement, Topical Report ER-157P, in safety evaluation reports (SERs) dated March 8, 1999, and December 20, 2001, respectively.

Topical Report ER-80P describes the LEFM technology, includes calculations of power measurement uncertainty using a Caldon LEFM Check System in a typical two-loop PWR or two-FW-line BWR, and provides guidelines and equations for determining the plant-specific power calorimetric uncertainties. Its supplement, Topical Report ER-157P, describes the Caldon LEFM √+ System and lists nonproprietary results of a typical PWR or BWR thermal power measurement uncertainty calculation using either the Caldon LEFM Check or LEFM √+ System. Together, these two reports provide a generic basis for a measurement uncertainty recapture power uprate.

The LAR also provides several enclosures (proprietary), which describe the plant-specific bases for the proposed uprate at Prairie Island, including:

- Cameron Engineering Report ER-532, Revision 1, "Uncertainty Analysis for Thermal Power Determination at Prairie Island Unit 1 Using the LEFM √+ System," issued February 2008

- Cameron Engineering Report ER-533, Revision 2, "Uncertainty Analysis for Thermal Power Determination at Prairie Island Unit 2 Using the LEFM √+ System," issued July 2008

- Cameron Engineering Report ER-583, Revision 0, "LEFM √+ Meter Factor Calculation and Accuracy Assessment for Prairie Island Unit 1 Nuclear Power Station," issued February 2008

- Cameron Engineering Report ER-553, Revision 2, "LEFM √+ Meter Factor Calculation and Accuracy Assessment for Prairie Island Unit 2 Nuclear Power Station," issued July 2008

3.9.1 Regulatory Evaluation

Topical Report ER-80P and its supplement, Topical Report ER-157P, describe the Caldon LEFM CheckPlus™ system for the measurement of FW flow and provide a basis for the proposed 1.64 percent power uprate of the licensed RTP. The NRC staff also considered the guidance of RIS 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture

Power Uprate Applications," dated January 31, 2002, in its review of the licensee's submittals for the proposed power uprate.

3.9.2 Technical Evaluation

3.9.2.1 LEFM Technology and Measurement

Both the Caldon LEFM Check and LEFM CheckPlus™ systems use transit time methodology to measure fluid velocity. The basis of the transit time methodology for measuring fluid velocity and temperature is that ultrasonic pulses transmitted through a fluid stream travel faster in the direction of the fluid flow than opposite the flow. The difference in the upstream and downstream traversing times of the ultrasonic pulse is proportional to the fluid velocity in the pipe, and the temperature is determined using a correlation between the mean propagation velocity of the ultrasound pulses in the fluid and the fluid pressure.

Both systems use multiple diagonal acoustic paths, instead of a single diagonal path, allowing velocities measured along each path to be numerically integrated over the pipe cross-section to determine the average fluid velocity in the pipe. This fluid velocity is multiplied by a velocity profile correction factor, the pipe cross-section area, and the fluid density to determine the FW mass flow rate in the piping. The mean fluid density may be obtained using the measured pressure and the derived mean fluid temperature as an input to a table of thermodynamic properties of water. The velocity profile correction factor is derived from calibration testing of the LEFM in a plant-specific piping model at a calibration laboratory.

The Caldon LEFM Check System consists of a spool piece with eight transducers, two on each of the four acoustic paths in a single plane of the spool piece. The velocity measured by any one of the four acoustic paths is the vector sum of the axial and the transverse components of fluid velocity as projected onto the path. The Caldon LEFM CheckPlus™ system uses 16 transducers, eight each in two orthogonal planes of the spool piece. In the Caldon LEFM CheckPlus™ system, when the fluid velocity measured by an acoustic path in one plane is averaged with the fluid velocity measured by its companion path in the second plane, the transverse components of the two velocities are canceled and the result reflects only the axial velocity of the fluid. This makes the numerical integration of four pairs of averaged axial velocities and computation of volumetric flow inherently more accurate than a result obtained using four acoustic paths in a single plane. Also, since there are twice as many acoustic paths and there are two independent clocks to measure the transit times, errors associated with uncertainties in path length and transit time measurements are reduced.

The NRC staff's review in the area of instrumentation and controls (I&C) covers the proposed plant-specific implementation of the FW flow measurement technique and the power increase gained as a result of implementing this technique, in accordance with the guidelines (A through H) provided in Section I of Attachment 1 to RIS 2002-03. The NRC staff conducted its review to confirm that the licensee's implementation of the proposed FW flow measurement device is consistent with the staff-approved Topical Reports ER-80P and ER-157P, and that the licensee adequately addressed the four additional requirements listed in the staff's SER (Section 3.9.2.2, Item D will discuss these four requirements in more detail). The NRC staff also reviewed the power measurement uncertainty calculations to ensure that (1) the conservatively proposed uncertainty value of 0.36 percent correctly accounts for all uncertainties associated with power level instrumentation errors and (2) the uncertainty calculations meet the relevant requirements of Appendix K to 10 CFR Part 50.

The licensee provided the information described below regarding the Caldon LEFM CheckPlus™ system FW flow measurement technique and its implementation at PINGP, Units 1 and 2.

The PINGP Unit 1 and Unit 2 LEFM systems contain an individual LEFM metering spool piece on each of the two FW lines and an electronics unit cabinet for each unit. Each LEFM is installed in accordance with the requirements of Caldon Topical Reports ER-80P and ER-157P and vendor guidelines in accordance with the PINGP modification process.

The Unit 1 LEFM electronics unit cabinet is installed in the Train A event monitoring room, and the Unit 2 LEFM electronics unit cabinet is installed in the Train B event monitoring room. The Train A and B event monitoring rooms are temperature controlled and provide mild environments. The output from the unit-specific LEFM cabinet is connected to the unit-specific plant emergency response computer system (ERCS) via an Ethernet connection. The ERCS provides the LEFM outputs to the secondary calorimetric program (CALM), control room thermal power monitor (TPM) display, LEFM status displays, and a critical computer alarm if the LEFM status changes.

The systems are designed with internal monitoring and checking devices to ensure the system parameters are within design limits during system operation. If system parameters exceed pre-established limits, a system alarm occurs that results in a control room critical ERCS alarm indicating that a change in LEFM status has occurred. Along with the computer alarm, the TPM displays and LEFM status displays reflect the change in LEFM status and allowable power based on the LEFM status.

3.9.2.2 LAR Compliance to RIS 2002-03, Attachment 1, Section I Guidance A through H

Items A through C

Items A, B, and C in Section I of Attachment 1 to RIS 2002-03 guide licensees in identifying the approved topical reports, providing references to the NRC's approval of the measurement technique, and discussing the plant-specific implementation of the guidelines in the topical report and the NRC staff's approval of the FW flow measurement technique, respectively.

In its LAR, the licensee identified Topical Reports ER-80P and ER-157P as applicable to the Caldon LEFM CheckPlus™ system. The licensee also referenced NRC SERs for Topical Reports ER-80P, dated March 8, 1999, and ER-157P, dated December 20, 2001. The licensee also cited an NRC SER dated July 5, 2006, which reexamined the performance of the Caldon systems and confirmed the validity of the previously referenced Caldon topical reports.

The licensee installed Caldon LEFM CheckPlus™ systems in PINGP Units 1 and 2 in February and October 2008, and incorporated the appropriate information into the site design, procedures, and maintenance programs. The licensee has changed the CALM program to incorporate the LEFM FW flow and temperature inputs to calculate an additional independent secondary calorimetric (LCALM) in addition to the existing venturi calorimetric (VCALM).

Based on its review of the licensee's submittals as discussed above, the NRC staff finds that the licensee has sufficiently addressed the plant-specific implementation of the Caldon LEFM CheckPlus™ system using proper topical report guidelines. Therefore, the licensee's description of the FW flow measurement technique and implementation of the power uprate using this technique follows the guidance in Items A through C of Section I of Attachment 1 to RIS 2002-03.

Item D

Item D in Section I of Attachment 1 to RIS 2002-03 guides licensees in addressing four criteria when implementing the FW flow measurement uncertainty technique. The staff's SERs on Topical Reports ER-80P and ER-157P both include these four plant-specific criteria to be addressed by a licensee referencing these topical reports for power uprate. The licensee's submittals address each of the four criteria as follows:

(1) The licensee should discuss the maintenance and calibration procedures that will be implemented with the incorporation of the LEFM. These procedures should include processes and contingencies for an inoperable LEFM and the effect on thermal power measurement and plant operation.

Licensee Response:

The licensee states that maintenance of the unit-specific LEFM system is performed during each refueling using site-specific procedures developed in accordance with the guidelines established in the vendor maintenance and troubleshooting manual. Proper maintenance is ensured through both automatic and manual checks of the system.

Qualified personnel use site procedures to perform calibration, maintenance, and corrective actions. PINGP I&C personnel have received formal onsite vendor training in LEFM operation and maintenance.

Based on its review of the licensee submittals, the NRC staff concludes that the licensee adequately addressed Criterion 1.

(2) For plants that currently have LEFMs installed, provide an evaluation of the operational and maintenance history of the installed installation and confirmation that the installed instrumentation is representative of the LEFM system and bounds the analysis and assumptions set forth in Topical Report ER-80P.

Licensee Response:

The licensee installed the PINGP Unit 1 LEFM during the February 2008 outage. It installed the Unit 2 LEFM during the October 2008 outage. In installing the Unit 1 and Unit 2 LEFMs, the licensee has considered industry operating experience associated with the LEFMs and incorporated the appropriate information into the site design, procedures, and maintenance programs.

Since turnover of the Unit 1 LEFM, the system has operated normally and no maintenance has been required. The licensee has completed normal system inspections in accordance with approved station procedures. During an unplanned trip of Unit 1 in July 2008 and May 2009, the LEFM remained in operation and responded as expected during the time offline and during the return to power.

Since turnover of the Unit 2 LEFM, the system has operated normally and no maintenance has been required. The licensee has completed normal system inspections in accordance with approved station procedures.

In addition, the licensee stated that PINGP operating and maintenance procedures for the LEFM have been developed to ensure that the assumptions and requirements of the uncertainty calculation remain valid.

Based on the review of the licensee submittals, the NRC staff finds the licensee's response adequate to address Criterion 2.

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

Licensee Response:

The licensee stated that the calculation of plant thermal power uncertainty is consistent with American Society of Mechanical Engineers, ASME Pressure Test Code (PTC) 19.1, "Test Uncertainty," 1990, and Instrument Society of America, (ISA) 67.04, "Setpoints for Nuclear Safety-Related Instrumentation," September 1994, as approved in Caldon Topical Reports ER-80P and ER-157P. Dependent errors such as temperature and enthalpy errors are mathematically combined to create random errors. Then the errors within instrument loops are combined using the square root sum of squares method. The LEFM uncertainty calculations have been performed to achieve a 95-percent confidence interval, 95-percent probability flow measurement. The methodology for determining the error associated with the ERCS thermal power calculation is unchanged.

Based on the above discussion, the NRC staff concludes that the licensee adequately addressed Criterion 3.

(4) For plant installation where the ultrasonic meter (including LEFM) was not installed with flow elements calibrated to a site-specific piping configuration (flow profiles and meter factors are not representative of the plant-specific installation), licensees should provide additional justification for its use. The 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 calibrations and plant configurations for the specific installation, including the propagation of flow profile effects at higher Reynolds numbers. Additionally, for previously installed calibrated elements, licensees

should confirm that the piping configuration remains bounding for the original LEFM installation and calibration assumptions.

Licensee Response:

The PINGP LEFM flow elements were calibrated at Alden Labs in a plant-specific piping configuration. The flow elements were installed in the same piping configuration as that at Alden Labs; however, because of an incorrect measurement shown on an existing plant drawing, and with the approval of Cameron, the plant installation of each Unit 2 LEFM flow element was 8 inches closer to the inlet of the FW venturis than during the testing at Alden Labs. Post-installation commission testing of the Unit 2 LEFMs verified that the actual plant installation remained bounded by the original LEFM installation and calibration assumptions.

In response to questions raised by the NRC during the NRC evaluation of the LEFM Check and CheckPlus™ systems, concerning the impact of changing or replacing flow transducers on the LEFM Check and CheckPlus™ system uncertainty analysis, Cameron performed testing to evaluate the effect on the system uncertainty when transducers were changed or replaced. Cameron Engineering Report ER-551P, Revision 1, "LEFM √+ Transducer Installation Sensitivity," dated March, 2007 (ADAMS Accession Nos. ML071500358 and ML071500360), reports the results of these tests. The total power measurement uncertainty calculations performed by Cameron for PINGP Units 1 and 2 include the uncertainty associated with transducer replacement.

Based on the foregoing, the NRC staff concludes that the licensee adequately addressed Criterion 4.

Item E

Item E in Section I of Attachment 1 to RIS 2002-03 guides licensees in the submittal of a plant-specific total power measurement uncertainty calculation, explicitly identifying all parameters and their individual contribution to the power uncertainty.

To address Item E of RIS 2002-03, the licensee provided Cameron Engineering Reports ER-532, Revision 1, and ER-533, Revision 2. In addition, the licensee provided a table listing each contribution factor and values for the overall thermal power calorimetric uncertainty. The NRC staff reviewed the calculations and determined that the licensee identified all the parameters associated with the thermal power measurement uncertainty, provided individual measurement uncertainties, and calculated the overall thermal power uncertainty.

The licensee's calculations arithmetically summed uncertainties for parameters that are not statistically independent and statistically combined with other parameters. The licensee combined random uncertainties using the square root sum of squares approach and added systematic biases to the result to determine the overall uncertainty. This methodology is consistent with the vendor determination of the Caldon LEFM CheckPlus™ system uncertainty, as described in the referenced topical reports, and is consistent with the guidelines in Regulatory Guide 1.105, "Setpoints for Safety-Related Instrumentation," Revision 3, issued December 1999.

The NRC staff finds that the licensee has provided calculations of the total power measurement uncertainty at the plant, explicitly identifying all parameters and their individual contribution to the power uncertainty. Therefore, the licensee has adequately addressed the guidance in Item E of Section I of Attachment 1 to RIS 2002-03.

Item F

Item F in Section I of Attachment 1 to RIS 2002-03 guides licensees in providing information to address the specified aspects of the calibration and maintenance procedures related to all instruments that affect the power calorimetric.

In the LAR, the licensee addressed each of the five aspects of the calibration and maintenance procedures listed in Item F of RIS 2002-03 related to all instruments that affect the power calorimetric as follows:

(1) Maintaining Calibration

The licensee stated that the LEFM is maintained in accordance with the guidelines established in the vendor maintenance and troubleshooting manual. Proper maintenance is ensured through both automatic and manual checks of the system. Manual checks are performed using site-specific procedures developed from the vendor maintenance and troubleshooting manual. The response to Item D.1 also addresses the preventive maintenance program.

(2) Controlling Hardware and Software Configuration

The Caldon LEFMs are designed and manufactured in accordance with the vendor's quality assurance program, which meets the requirements of Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50. The licensee stated that the LEFM hardware configuration is controlled on site by PINGP's configuration control program. The LEFM software is controlled on site by PINGP's software quality assurance program.

(3) Performing Corrective Actions

Qualified maintenance personnel use controlled plant procedures to perform corrective actions under the site work control process. The licensee documents and evaluates any conditions that are adverse to quality under the site corrective action program.

(4) Reporting Deficiencies to the Manufacturer

Equipment problems for all plant systems, including the LEFM equipment, fall under the site work control process or the corrective action process. The licensee documents and evaluates conditions adverse to quality under the corrective action program and subsequently transmits them to the vendor as appropriate.

(5) Receiving and Addressing Manufacturer Deficiency Reports

Cameron's quality assurance program includes the PINGP LEFMs. Cameron maintains procedures for user notification of significant deficiencies and processes them through its customer information bulletins. The licensee would screen any deficiency reports coming to PINGP for the LEFM via a Cameron customer information bulletin or industry operating experience and address them under the operating experience assessment process.

The NRC staff's review of the above information found that the licensee addressed the calibration and maintenance aspects of the Caldon LEFM CheckPlus™ system and all other instruments affecting the power calorimetric. Thus, the licensee meets the guidance in Item F of Section I of Attachment 1 to RIS 2002-03.

Items G and H

Items G and H in Section I of Attachment 1 to RIS 2002-03 guide licensees to provide a proposed allowed outage time (AOT) for the instrument and to propose actions to reduce power if the AOT is exceeded.

In response to the NRC staff's RAI, the licensee described the three LEFM system mode indications as follows and listed all conditions in both "Alert" and "Fail" modes:

- Normal: An LEFM system "Normal" status is displayed when all the FW flow, temperature, and header pressure signals for FW loops A and B are normal and operating within design limits. Calculated power level error associated with the LEFM flow measuring system in this condition is 0.36 percent.

- Alert: An LEFM system "Alert" alarm indicates a loss of redundancy, and the calculated power level error associated with the LEFM flow measuring system in this condition is 0.54 percent.

- Fail: An LEFM system "Fail" alarm indicates a loss of function and the power level error reverts to the 2-percent error associated with the venturi flow meters.

With an LEFM "Alert" or "Fail" status, the licensee stated that operation at 100-percent uprated licensed core thermal power (1677 MWt) may be continued until the next daily surveillance of the nuclear instrumentation system, as required by TS Surveillance Requirement (SR) 3.3.1.2. Operation at uprated licensed core power until the next scheduled calorimetric is permissible since the nuclear instruments will still be operating within their TS-required daily calibration window.

As described below, if the LEFM system is not restored to "Normal" operating status at the next scheduled daily calorimetric calibration, a reactor power reduction is required. Power must be reduced by an amount commensurate with the increase in uncertainty associated with the condition of the FW flow measurement system (i.e., either 0.54 percent or 2.0 percent).

The licensee discussed the LEFM operating modes and proposed AOTs as described below.

Normal Operation

"Normal" operation of the system includes normal functioning of the LEFM meter in each FW loop, wherein each Caldon LEFM CheckPlus™ meter contains two flow monitoring planes, and each plane includes four ultrasonic flowpaths. The power measurement uncertainty of operating in the "Normal" mode with both planes in each meter operable is calculated to be 0.36 percent.

Alert Mode

If any one plane in either or both LEFM meters is inoperable, the "Alert" alarm is received. This condition represents a loss of redundancy, and the power measurement uncertainty is increased from 0.36 percent to 0.54 percent.

Operator response actions in this condition are as follows:

- Operation at licensed (uprated) core power (1677 MWt, as indicated on the TPM and based on calorimetric heat balance calculations using FW flow measurements from the LEFM planes still in operation, LCALM), may continue until the next scheduled daily power calorimetric surveillance. The TPM computer program will change the allowable power level from 1677 MWt (100 percent) to 1674 MWt (99.82 percent) and will display an alarm if the calculated current power level is greater than this allowable power level.

  Justification: Operation in this condition is allowable because the LEFM condition does not affect the power range nuclear instruments. The instruments are considered operable for providing required reactor protection until their next daily calorimetric surveillance as required by TS SR 3.3.1.2. The alarm condition alerts operators to the potential need to reduce power if the LEFM is not restored to "Normal" status by the next daily calorimetric heat balance surveillance.

- If the LEFM is not restored to "Normal" operating conditions by the next scheduled daily calorimetric calibration, and if the LEFM remains in the "Alert" mode, then power level will be reduced to less than or equal to 1674 MWt (99.82 percent), as determined by the calorimetric heat balance calculated using the LEFM (LCALM). This condition may continue indefinitely until the LEFM is restored.

Justification: With the LEFM in "Alert," the LEFM system measurement uncertainty increases from 0.36 percent to 0.54 percent. Reducing allowable power to less than or equal to 1674 MWt (analyzed power level of 1683 MWt – 0.54 percent [9 MWt] = 1674 MWt) corresponds to the increase in the LEFM measurement uncertainty during operation in the "Alert" mode (operating with only one plane of flow sensors in either or both FW loops). The LEFM performs self-checks and does not exhibit drift, as illustrated by the fact that the Caldon LEFM system can operate without calibration, with either one or two flow measurement planes. This condition of a single measurement plane represents the configuration of a Caldon Check ultrasonic flow measurement system. Operating indefinitely with the LEFM in the "Alert" mode, in the reduced power condition, is equivalent to operating with a Caldon Check system, as has been used in other nuclear plant flow monitoring systems.

"Fail" Mode

If both LEFM planes in either or both FW loop meters are inoperable, the "Fail" alarm is received. This condition represents a loss of LEFM function. The power measurement calorimetric heat balance computer program automatically changes to use of the VCALM, and the assumed power measurement uncertainty is increased to 2 percent, which is the uncertainty associated with the current venturi-based flow measurement system.

Operator response actions in this condition are as follows:

- Operation at licensed core power (1677 MWt, as indicated on the TPM and based on calorimetric heat balance calculations using FW flow measurements from VCALM) may continue until the next scheduled daily power calorimetric. The TPM will change the allowable power level from 1677 MWt (100 percent) to 1650 MWt (98.38 percent) and will display an alarm if the current calculated power level is greater than this allowable power level.

  Justification: Operation in this condition is allowable because the LEFM condition does not affect the power range nuclear instruments. The instruments are considered operable for providing required reactor protection until their next daily calorimetric calibration, as required by TS SR 3.3.1.2. The alarm condition alerts operators to the potential need to reduce power if the LEFM is not restored to "Normal" status by the next daily calorimetric heat balance.

- If the LEFM remains in the "Fail" mode by the next scheduled daily calorimetric calibration, then power level will be reduced to less than or equal to 1650 MWt. This condition may continue for up to 7 days with the last known valid correction factors until the LEFM is restored. If the LEFM is restored to the "Alert" condition, then actions appropriate for that condition are taken as described above.

Justification: With the LEFM in "Fail," the calorimetric automatically reverts to using the VCALM with the original assumed measurement uncertainty of 2 percent. Operation at less than or equal to 1650 MWt is consistent with the accuracy of the remaining operable flow measurement system and is also consistent with operations before the measurement uncertainty recapture power uprate.

The thermal power measurement correction factors are used to normalize the power levels determined from the venturi-based FW flow measurements to the more accurate LEFM flow measurements. The correction factors are essentially continuous calibration factors for the venturi flow measurement system.

Operation for up to 7 days while using the last known good correction factors (before LEFM failure) in the VCALM provides assurance that the licensed thermal power level is not exceeded when the calorimetric power calculation program automatically shifts from the LEFM-based FW flow measurements (LCALM) to VCALM. The 7-day period provides time to restore the LEFMs to their "Normal" status.

After 7 days, if the LEFM is not restored to service, correction factors for the VCALM FW flow and temperature inputs will be reset to 1.0. If the correction factors are greater than 1.0, the power level will be reduced an additional amount equivalent to the combined average of FW flow correction factors, as described in the Enclosure 2, pages 9 and 10, of the LAR. These actions will limit the deviation between actual core power and calculated core power determined using the venturi-based FW flow measuring system.

Instrument Drift and Nozzle Fouling Considerations

Evaluation and trending of correction factors allows identification of changes in deviations between the LEFM and venturi FW flow measurement systems and could indicate instrument drift or nozzle fouling. As explained in Enclosure 2 to the LAR, item I.D.1 (page 9), plant-specific trending of the venturi FW flow and temperature correction factors indicates that potential drift of the FW venturi flow and resistance temperature detector (RTD) instrumentation is less than 0.1 percent within 7 days. Trending of correction factors reflects all factors that could contribute to differences between the LEFM and venturi flow measurement systems, such as individual instrument drift, instrument bias, or environmental effects. Attributing the 0.1-percent deviation in correction factors to instrument drift is conservative and bounds drift values based on the manufacturer's data for the venturi flow measurement components. For the same 7-day time interval, flow measurement drift based on instrument vendor data is calculated to be approximately 0.012 percent.

As noted above, the LEFM system performs self-checks and is not subject to drift. The LEFM CheckPlus™ system has been installed at PINGP Unit 1 since the February 2008 outage, and at PINGP Unit 2 since the October 2008 outage. The licensee states that no drift or calibration issues regarding the LEFMs have occurred. The licensee and the LEFM vendor (Cameron) have reviewed the LEFM and venture FW flow data for trends since installation of the Unit 1 and Unit 2 LEFM systems. This review has not identified any indications between the LEFM and venturi data that would represent nozzle fouling or defouling events.

The NRC staff reviewed the licensee's submittals and found that the licensee provided sufficient justification for the proposed AOTs and the proposed actions to reduce power level if the AOT is exceeded. Therefore, the licensee has followed the guidance in Items G and H of Section I of Attachment 1 to RIS 2002-03.

3.9.3 Conclusion

The NRC staff reviewed the licensee's proposed plant-specific implementation of the FW flow measurement device and the power uncertainty calculations and determined that the licensee's proposed amendment is consistent with the approved Topical Report ER-80P and its supplement Topical Report ER-157P. The staff has also determined that the licensee adequately accounted for all instrumentation uncertainties in the reactor thermal power measurement uncertainty calculations and demonstrated that the calculations meet the relevant requirements of 10 CFR Part 50, Appendix K.

Therefore, the staff finds the instrumentation and controls aspects of the 1.64 percent proposed thermal power uprate to be acceptable.

3.10 Plant Systems

3.10.1 Regulatory Evaluation

The NRC staff's review in the area of plant systems covers the impact of the proposed MUR-PU on the NSSS interface systems, containment systems, safety-related cooling water systems, spent fuel pool (SFP) storage and cooling, radioactive waste systems, and ESF HVAC systems. The staff's review is based on the guidance in SRP Chapters 6 and 9, and RIS 2002-03, Attachment 1, Sections II, III, and VI. The licensee evaluated the effect of the MUR-PU on the plant systems. This evaluation is reflected in Section VI of Enclosure 2 of the licensee's application dated December 28, 2009.

The staff reviewed the licensee's LAR for compliance with the following regulations. GDC 16, containment design, requires that the containment shall provide an essentially leak tight barrier against the uncontrolled release of radioactivity to the environment. GDC 19, control room, requires that the control room must provide the operators with the capability to operate the nuclear power units safely under normal conditions and maintain the reactor in a safe condition under accident conditions including a LOCA. GDC 38, containment heat removal, requires that the containment heat removal systems are capable of rapidly reducing the containment temperature and pressure following a LOCA, and maintaining them at an acceptably low level. GDC 50, containment design basis, requires that the containment accommodate the pressure and temperature conditions resulting from a LOCA without exceeding the design leakage rate. GDC 60, control of releases of radioactive materials to the environment, requires the nuclear power unit to have means to control the release of radioactive materials in gaseous and liquid effluents during normal operation and anticipated operational occurrences.

3.10.2 Technical Evaluation

NSSS Interface Systems

The NSSS interface systems at PINGP include the MS system, the steam dump subsystem, the condensate and FW systems, the auxiliary FW (AFW) system), and the steam generator blowdown system.

The licensee evaluated the major components of the MS system (SG steam safety valves, SG power-operated atmospheric relief valves (ARVs), the MS isolation valves (MSIVs), and associated check and bypass valves. The licensee determined that the MUR-PU conditions for the SG safety valves, the SG ARVs, and the MSIVs are bounded by the CLB. The licensee also determined that the MUR-PU has no impact on the interface requirements for the MSIV bypass valves. Therefore, the MS system components are capable of supporting the MUR-PU.

The steam dump sub-system, consisting of the condenser steam dump system and the atmospheric dump system, is provided to remove energy for the steam generators downstream of the MS isolation and non-return valves. The licensee evaluated the operation of the steam dump sub-system and determined that the plant operability analysis for the MUR-PU verified that the current load rejection capability (40 percent of plant rated electrical load without a plant trip) remains at MUR-PU conditions without change to the NSSS control system setpoints and time constants. Therefore, the steam dump sub-system is capable of supporting the MUR-PU.

The condensate and FW systems provide FW to the SG from the condenser hotwell during normal operation. The FW system isolates during accidents. The licensee evaluated the FW control valves and the FW pumps using the NSSS design parameters for the MUR-PU. The licensee determined that the MUR-PU conditions (FW flow and density) are bounded by the CLB. Therefore, the condensate and FW systems are capable of supporting the MUR-PU.

The AFW system provides FW to the secondary side of the SGs when the normal FW system is not available, thereby maintaining the heat sink of the SGs. The licensee evaluated the effects of the MUR-PU on the AFW system and determined that the minimum flow requirements at MUR-PU conditions are bounded by the current safety analyses and the CLB. In addition, the licensee evaluated whether there was sufficient condensate available in the condensate storage tank to allow the AFW pumps to fulfill their design function during accident or transient conditions. The licensee determined that the MUR-PU conditions are bounded by the CLB. Therefore, the AFW system is capable of supporting the MUR-PU.

The SG blowdown system controls the chemical composition of the SG secondary side water within the specified limits, and also controls the buildup of solids in the SG secondary-side tubesheet. The licensee reviewed the blowdown required to control secondary chemistry and SG solids at the uprated condition. The licensee determined that, based on the no-load steam pressure and the minimum full-load steam pressure, the MUR-PU will not impact blowdown flow control. Therefore, the SG blowdown system is capable of supporting the MUR-PU.

The NRC staff reviewed the licensee's evaluations and concurs with the results. The licensee determined that there is no adverse impact on the NSSS interface systems from the MUR-PU because there is sufficient operating margin to produce an additional 1.64 percent power, and all equipment will be operated within its design limits. The staff does not anticipate that an MUR-PU will challenge the NSSS interface systems, and all systems have been shown to be operating within design. Therefore, the NRC staff concludes that the NSSS systems are acceptable for the MUR uprate.

Containment System

The NRC staff reviewed the following areas of containment design and analysis for the proposed PINGP MUR-PU: short-term LOCA containment response analyses, long-term LOCA containment response analyses, containment response to a main steamline break (MSLB) inside containment, net positive suction head evaluation for safety injection and containment spray pumps, impact of the MUR-PU on containment isolation, and the impact of the MUR-PU on combustible gas control.

The licensee stated that the existing containment integrity evaluations remain valid for MUR-PU conditions because the analyzed values for reactor vessel average temperature, design flow, and the reactor coolant pressure are the same for both current and for MUR-PU levels. Therefore, the energy that can be released from the reactor at MUR-PU conditions is essentially the same as in the current analysis. The current long-term mass and energy release evaluation assumed 102-percent of 1650 MWt (i.e., 1683 MWt) and therefore considers the operating conditions at MUR-PU levels.

The short-term LOCA mass and energy releases and containment sub-compartment response analyses are affected by the fact that the cold-leg temperatures are less than the current analysis. Therefore, the mass flux into the sub-compartments may possibly increase for a cold-leg break. In a response to request for additional information RAI, the licensee stated that calculations performed show that the difference between cold-leg and hot-leg fluid temperatures for current and MUR-PU conditions is not significant. Therefore, the mass and energy releases resulting from a large hot-leg or cold-leg break are essentially the same at MUR-PU and current operating conditions. Also, the short-term LOCA containment response analysis for the MUR-PU conditions shows that the peak LOCA containment pressure and temperature conditions are bounded by the current analysis, and therefore the sub-compartment pressurization loads are bounded by the current loads.

The licensee stated that the current containment response analyses of the MSLB accident inside the containment performed at the core power of 1683 MWt is conservative and bounding for the MUR-PU. This is the expected result and the NRC staff concurs.

The licensee stated that the required heat removal capability of the containment spray (CS) system and the portion of the primary containment ventilation system that is required to operate during post-accident conditions were reviewed for potential impact by the MUR-PU. The LOCA and MSLB containment analyses establish the system and component design requirements for these systems. Since the current licensing basis analyses bound the MUR-PU conditions, the CS system and the portions of the ventilation system required for post-accident containment heat removal are not impacted by the PU. The NRC staff concurs with the licensee's conclusion that no further evaluation for these systems is required.

The licensee stated that the MUR-PU does not add or remove any containment isolation valves. The ability to close or operate containment isolation valves and position indication capability is independent of the reactor power level except for the possibility of overpressurization between redundant closed containment isolation valves addressed in GL 96-06. Since the predicted containment accident temperatures following the design-basis LOCA and MSLB accident inside containment are bounded by those of the current licensing basis, this issue is acceptably addressed for the MUR-PU.

Safety-Related Cooling Water Systems

The safety-related cooling water systems include the safety injection (SI) system/CS system, the residual heat removal (RHR) system, and the component cooling water (CC) system and the cooling water (CL) system.

The required volume, duration, and heat rejection capability of the SI system and CS system flows in the event of a break is determined based on the analytical and empirical models that simulate reactor and containment conditions subsequent to the postulated RC system and MS system breaks. From these analyses, the system and component criteria necessary to demonstrate compliance with regulatory requirements at the uprated conditions are established. The licensee reviewed the SI and CS systems for impacts of the uprated power conditions and determined that the MUR-PU conditions are bounded by the CLB. Therefore, no further evaluation is required.

The MUR-PU increases the heat generated in the core during normal cooldown, refueling operations, and accident conditions. This provides a higher heat load on the RHR heat exchangers during cooldown, and also during refueling outages. The licensee evaluated these conditions with respect to the MUR-PU. The licensee stated that the CLB analyses associated with demonstrating the TS cooldown time limits are satisfied and are performed based on 102 percent of the current licensed power level of 1650 MWt. The licensee also stated that the current analysis of record demonstrates continued compliance with TS cooldown requirements at the MUR-PU condition. The licensee also stated that the current Appendix R cooldown analysis will continue to bound the power level and plant conditions at the MUR-PU conditions.

The CC system is designed to remove heat from major components in the NSSS under normal conditions, and from all components associated with removal of reactor core decay heat under accident conditions. The licensee evaluated the impact of the MUR-PU on the CC system. The licensee stated that the changes to normal plant operating parameters which impact the CC system are negligible. The licensee also stated that the accident analyses relevant to the CC system are performed assuming a core thermal power of at least 102 percent of the current rated value, which bound the MUR-PU operating conditions. Therefore, the NRC staff concludes that the CC system is capable of safely supporting the MUR-PU.

The CL system is designed to provide redundant cooling water supplies with isolation valves to auxiliary FW pumps, Unit 1 diesel generators, air compressors, CC heat exchangers, containment fan-coil units, and the Auxiliary Building unit coolers. The licensee evaluated the impact of the MUR-PU on the CL system. The licensee stated that the MUR-PU has a negligible impact, if any, on the normal CL system and component duties, and changes in flow rates and operating limits are within the existing system design. The licensee also stated that the safety-related heat loads on the CL system are based on 102 percent of the current licensed core power, which bound the MUR-PU conditions. Therefore, the NRC staff concludes that the CL system is capable of safely supporting the MUR-PU.

The NRC staff reviewed the licensee's evaluation of safety-related cooling water systems. Based upon the licensee's determination that the existing analyses for these systems were evaluated for 102 percent RTP, the NRC staff finds there is reasonable assurance that the systems are acceptable for the MUR-PU.

Spent Fuel Pool Storage and Cooling System

The principal function of the spent fuel pool storage and cooling (SFPC) system is to provide storage and cooling of the spent fuel. The primary impact of a power uprate would be to the decay heat of the fuel recently discharged from the core. The licensee stated that the power level assumed in the SFPC system calculation already assumes decay heat associated with 102 percent of the current operating power and the maximum expected core operating time. Therefore, the NRC staff concurs with the licensee's conclusion that the SFPC system will not be impacted by the power uprate.

Radioactive Waste Systems

The waste processing systems provide the means to sample, collect, process, store/hold, re-use, and/or release gaseous and liquid low-level effluents and solids. The licensee evaluated the normal annual radiological effluents for the uprated power level. The licensee determined that the liquid and gaseous radwaste system design will be capable of maintaining normal operational offsite releases and doses within regulatory requirements and effluents will remain bounded by the Final Environmental Study estimates.

The licensee stated that they do not expect the volume of solid waste to increase proportionally to the increase in core power because the MUR-PU does not appreciably affect installed equipment and does not require changes in system operation.

The NRC staff does not expect a 1.64 percent increase in power to result in a significant change to the operation of the radioactive systems. Therefore, based on the licensee's assessment, the staff concludes that the radioactive waste systems will function adequately for the MUR-PU.

Engineered Safety Features Heating, Ventilation and Air Conditioning Systems

The licensee considered the safeguards chilled water system for impact by the MUR-PU. The licensee determined that there will be only a minor increase in the heat load on the system as a result of the MUR-PU, and that adequate capacity is available in the current design to accommodate this increase. Based on the licensee's evaluation, the NRC staff concludes that no further evaluation is required.

The licensee considered the control room area ventilation system for impact by the MUR-PU. Since the accidents affecting the control room habitability are based on 102-percent of the current licensed power which bounds the MUR-PU conditions, no further evaluation is required. Based on the licensee's evaluation, the NRC staff concludes that no further evaluation is required.

The licensee considered the auxiliary building special ventilation system for impact by the MUR-PU. The licensee stated that the component design parameters are bounded by the original design ratings and for accident response are designed for power operation at 102-percent of the current licensed thermal power. Therefore, the MUR-PU conditions are bounded by the current design conditions. Based on the licensee's evaluation, the NRC staff concludes that no further evaluation is required.

The licensee considered the spent fuel pool special ventilations sub-system for impact by the MUR-PU. The licensee determined that the sub-system has no power dependent piping or equipment in its service area. The heat released from the pool will not change because the pool temperature will not change with the MUR-PU, and therefore, the MUR PU conditions are bounded by the current licensing basis. Based on the licensee's evaluation, the NRC staff concludes that no further evaluation is required.

The licensee concluded that the safety functions of these systems are not impacted by the power uprate. The NRC staff reviewed the licensee's evaluation and, based on the licensee's assessment, the staff concludes that the ESF HVAC systems are acceptable for the MUR-PU.

3.10.3 Conclusion

In summary, the licensee reviewed the design and operation of the plant systems and determined that the proposed MUR-PU does not adversely impact any of the systems. For the reasons noted above, the NRC staff concludes that the plant systems will be acceptable for the MUR-PU.

3.11 Changes to Facility Operating License and TSs

3.11.1 Regulatory Evaluation

This LAR revises the licensed power level identified in Condition 2.C of the Facility Operating Licenses and the TS definition of Rated Thermal Power.

Licensees may revise the TS content provided that plant-specific review supports a finding of continued adequate safety because: (1) the change is editorial, administrative or provides clarification (i.e., no requirements are materially altered), (2) the change is more restrictive than the licensee's current requirement, or (3) the change is less restrictive than the licensee's current requirement, but nonetheless, still affords adequate assurance of safety when judged against current regulatory standards. The detailed application of this general framework, and additional specialized guidance, are discussed in this section in the context of the specific proposed changes.

3.11.2 Technical Evaluation

The submittal includes TS requirements that would demonstrate compliance with 10 CFR 50.36, "Technical specifications," for plant operating conditions related to the requested authorization for a power level increase. The plant modifications will improve the accuracy of the plant power calorimetric measurement based on the Caldon LEFM CheckPlus™ system UFM instrumentation.

3.11.2.1 Facility Operating License and Definitions – Rated Thermal Power

The licensee proposed to revise paragraph 2.C of the Facility Operating License and TS 1.1, Definitions – RTP to reflect the authorized power level increase. The TS RTP will limit the maximum reactor core heat transfer rate to the reactor coolant to 1677 MWt. The NRC staff finds that this change meets 10 CFR 50.36 and is acceptable because the TS limit for operation is derived from the analyses and evaluation included in the safety analysis report (SAR) as accepted by the SE for the requested power level increase discussed herein.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change 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 amendments involve 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 amendments involve no significant hazards consideration, and there has been no public comment on such finding (75 FR 26291). Accordingly, the amendments meet 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 amendments will not be inimical to the common defense and security or to the health and safety of the public.

7.0 LICENSEE COMMITMENTS

The following table identifies regulatory commitments made by the licensee in the application dated December 28, 2009.

Commitment Number	Commitment	Implementation Schedule
1	The PINGP Technical Requirements Manual (TRM) will be revised to include LEFM administrative controls.	Prior to operating above 1650 MWt
2	Revise ERCS Alarm Response Procedure to reflect any changes in LEFM status such as outage time and power limits.	Prior to operating above 1650 MWt
3	Revise CHECWORKS models to incorporate flow and process system conditions that are determined for the MUR PU conditions.	Prior to operating above 1650 MWt
4	Revise Emergency and Abnormal Operating Procedures that are power dependent.	Prior to operating above 1650 MWt
5	Recalibrate BOP system alarms due to small process condition changes.	Prior to operating above 1650 MWt
6	Revise ERCS and Simulator Calorimetric and TPM programs with new administrative power limits based on LEFM status. Other core power dependent ERCS and Simulator programs such as Xenon, NIS Power, and Boron Concentration will be revised to reflect a core power of 1677 MWt.	Prior to operating above 1650 MWt
7	Revise Operator Training Program to include changes to plant procedures and alarm responses in addition to Operator Training regarding the implementation of the allowable at-power administrative limits and new TRM governing LEFM out-of-service time.	Prior to operating above 1650 MWt
8	Re-scale applicable control and protection instrumentation consistent with the increase in 100 percent nominal core power from 1650 MWt to 1677 MWt.	Prior to operating above 1650 MWt
9	Revise ERCS TPM and CALM Programs to adjust the allowable licensed thermal power values used in these programs. Alarms will require evaluation and re-calibration, as required, to reflect small process changes in certain BOP systems. Other core power dependent ERCS programs such as Xenon, NIS Power, and Boron Concentration will be revised to reflect a core power of 1677 MWt.	Prior to operating above 1650 MWt

10	As part of the ERCS TPM program changes, the time greater than 100 percent power incremental monitoring levels will be based on a power measurement uncertainty of 0.36 percent.	Prior to operating above 1650 MWt

8.0 REFERENCES

1) Letter from M. A. Schimmel, Northern States Power Company - Minnesota, to NRC Document Control Desk, "Prairie Island Nuclear Generating Plant Units 1 and 2, Dockets 50-282 and 50-306, License Nos. DPR-42 and DPR-60, License Amendment Request for Measurement Uncertainty Recapture - Power Uprate," dated December 28, 2009 (ADAMS Accession No. ML093650061).

2) Letter from M. A. Schimmel, Northern States Power Company - Minnesota, to NRC Document Control Desk, "Prairie Island Nuclear Generating Plant Units 1 and 2, Dockets 50-282 and 50-306, License Nos. DPR-42 and DPR-60, Supplement to License Amendment Request for Measurement Uncertainty Recapture - Power Uprate – Response to Requests for Additional Information (TAC Nos. ME3015 and ME3016)," dated April 19, 2010 (ADAMS Accession No. ML101090498).

3) Letter from M. A. Schimmel, Northern States Power Company - Minnesota, to NRC Document Control Desk, "Prairie Island Nuclear Generating Plant Units 1 and 2, Dockets 50-282 and 50-306, License Nos. DPR-42 and DPR-60, Supplement to License Amendment Request for Measurement Uncertainty Recapture – Power Uprate, Withdrawal of Proposed Change to Analysis Methodology for Pressure Temperature Limits Report (TAC Nos. ME3015 and ME3016)," dated April 23, 2010 (ADAMS Accession No. ML101130449).

4) Letter from M. A. Schimmel, Northern States Power Company - Minnesota, to NRC Document Control Desk, "Prairie Island Nuclear Generating Plant Units 1 and 2, Dockets 50-282 and 50-306, License Nos. DPR-42 and DPR-60, Supplement to License Amendment Request for Measurement Uncertainty Recapture - Power Uprate – Additional Information Regarding Main Steam System Stress Analyses and Electrical Output (TAC Nos. ME3015 and ME3016)," dated June 17, 2010 (ADAMS Accession No. ML101690068).

5) Letter from M. D. Wadley, Nuclear Management Company, LLC, to NRC Document Control Desk, "Prairie Island Nuclear Generating Plant Units 1 and 2, Dockets 50-282 and 50-306, License Nos. DPR-42 and DPR-60, Application for Renewed Operating Licenses," dated April 11, 2008 (ADAMS Accession No. ML081130663).

6) Letter from M. D. Wadley, Northern States Power Company - Minnesota, to NRC Document Control Desk, "Prairie Island Nuclear Generating Plant Units 1 and 2, Dockets 50-282 and 50-306, License Nos. DPR-42 and DPR-60, License Amendment Request for Technical Specifications Changes to Allow Use of Westinghouse 0.422-inch OD 14x14 VANTAGE+ Fuel," dated June 26, 2008 (ADAMS Accession No. ML081820137).

7) Letter from M. D. Wadley, Northern States Power Company - Minnesota, to NRC Document Control Desk, "Prairie Island Nuclear Generating Plant Units 1 and 2, Dockets 50-282 and 50-306, License Nos. DPR-42 and DPR-60, Response to Request for Additional Information Regarding License Amendment Request for Technical Specifications Changes to Allow Use of Westinghouse 0.422-inch OD 14x14 VANTAGE+ Fuel (TAC Nos. MD9142 and MD9143)," dated March 12, 2009 (ADAMS Accession No. ML081820137).

8) Letter from T. J. Wengert, NRC, to M. D. Wadley, Northern States Power Company – Minnesota, "Prairie Island Nuclear Generating Plant, Units 1 and 2 – Issuance of Amendments Re: Technical Specifications Changes to Allow Use of Westinghouse 0.422-inch OD 14x14 VANTAGE+ Fuel," dated July 1, 2009. (ADAMS Accession No. ML091460809).

9) "Caldon Ultrasonics, Engineering Report: ER-551P Rev.1, LEFM✔ + Transducer Installation Sensitivity," dated March 2007 (ADAMS Accession No. ML071500360 (proprietary) and ML072740228 (non-proprietary)).

Principal Contributors: W. Rautzen, J. Parillo, N. Iqbal, B. Parks, D. Woodyatt, W. Lyon, Y. Huang, T. McLellan, S. Basturescu, P. Chung, W. Jessup, A. Sallman, A. Obodoako, K. Martin, T. Wengert, T. Beltz

Date: August 18, 2010

August 18, 2010

Mr. Mark A. Schimmel
Site Vice President
Prairie Island Nuclear Generating Plant
Northern States Power – Minnesota
1717 Wakonade Drive East
Welch, MN 55089

SUBJECT: PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2 – AMENDMENT RE: MEASUREMENT UNCERTAINTY RECAPTURE POWER UPRATE (TAC NOS. ME3015 AND ME3016)

Dear Mr. Schimmel:

The U.S Nuclear Regulatory Commission has issued the enclosed Amendment No. 197 to Facility Operating License No. DPR-42 and Amendment No. 186 to Facility Operating License No. DPR-60 for the Prairie Island Nuclear Generating Plant, Units 1 and 2, respectively. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated December 28, 2009, as supplemented by letters dated April 19, April 23, and June 17, 2010.

The amendments revise the license and TSs to reflect a 1.64 percent increase in the licensed rated thermal power (RTP) from 1650 megawatts thermal (MWt) to 1677 MWt. The RTP increase is based on reduced uncertainty in the RTP measurement achieved by installation of a Caldon Leading Edge Flow Meter CheckPlus™ System used to measure FW flow and temperature.

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

Sincerely,
/RA/
Thomas J. Wengert, Senior Project Manager
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-282 and 50-306
Enclosures:
1. Amendment No. 197 to DPR-42
2. Amendment No. 186 to DPR-60
3. Safety Evaluation

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ADAMS Accession No. ML102030573 * via memorandum

OFFICE	DCI/CPTB/BC (A)	DCI/CSGB/BC	DCI/CVIB/BC	DCI/CPNB/BC	DE/EEEB/BC
NAME	STingen *	RTaylor *	MMitchell *	TLupold	RMathew (A) *
DATE	04/08/10	04/26/10	05/26/10	08/09 /10	06/29/10
OFFICE	DE/EICB/BC	DE/EMCB/BC	DIRS/IHPB/BC	DIRS/ITSB/BC	DRA/AADB/BC
NAME	WKemper *	MKhanna *	UShoop *	RElliott	TTate *
DATE	05/04/10	06/25/10	05/27/10	08/04/10	05/17/10
OFFICE	DRA/AFPB/BC	DSS/SCVB/BC	DSS/SRXB/BC	DSS/SBPB/BC	DSS/SNPB/BC
NAME	AKlein *	RDennig *	AUlses *	GCasto	AMendiola
DATE	05/07/10	05/06/10	05/28/10	08/03/10	08/03/10
OFFICE	LPL3-1/PM	LPL3-1/LA	OGC NLO	LPL3-1/BC	DORL/D
NAME	TWengert	BTully */THarris for*	MDreher	RPascarelli	JGiitter
DATE	08/17/10	08/17/10	08/16 /10	08/17/10	08/18/10

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Letter to Mark A. Schimmel from Thomas J. Wengert dated August 18, 2010

SUBJECT: PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2 – AMENDMENT RE: MEASUREMENT UNCERTAINTY RECAPTURE POWER UPRATE (TAC NOS. ME3015 AND ME3016)

PUBLIC	LPL3-1 R/F	RidsNrrDorl
RidsNrrPMPrairieIsland	RidsOgcRp	RidsNrrLABTully
RidsAcrsAcnwMailCenter	RidsNrrDorlDpr	RidsRgn3MailCenter
RidsNrrDorlLpl3-1	RidsNrrDciCsgb	
RidsNrrDciCvib	RidsNrrDciCptb	RidsNrrDprPgcb
RidsNrrDciCpnb	RidsNrrDeEeeb	RidsNrrDeEicb
RidsNrrDeEmcb	RidsNrrDirsIolb	RidsNrrDirsItsb
RidsNrrDraAadb	RidsNrrDraAfpb	RidsNrrDssSnpb
RidsNrrDssSrxb	RidsNrrDssSbpb	B. Parks, NRR
T. Alexion, NRR	A. Obodoako, NRR	S. Bagley, OEDO
W. Rautzen, NRR	P. Chung, NRR	A. Sallman, NRR
Y. Huang, NRR	C. Schulten, NRR	A. Johnson, NRR
T. McLellan, NRR	N. Iqbal, NRR	A. Travers, NRR
W. Jessup, NRR	S. Basturescu, NRR	K. Martin, NRR
D. Woodyatt, NRR	T. Beltz, NRR	W. Lyon, NRR